Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion®

July 30, 2012

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Serial No. 12-458
LIC/JG/R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 254: PROPOSED TECHNICAL SPECIFICATIONS TO ADOPT TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (KPS). The proposed amendment would modify TS requirements regarding steam generator tube inspections and reporting as described in Technical Specification Task Force (TSTF) 510, Revision 1, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination and environmental considerations for the proposed changes. Attachment 2 provides the current TS pages marked up to show the proposed changes. Attachment 3 provides the proposed TS pages incorporating the proposed changes (Attachment 3 includes unaffected TS sections whose pagination was changed as a result of this amendment). Attachment 4 provides the marked-up TS Bases pages (for information).

The KPS Facility Safety Review Committee has approved the proposed change and a copy of this submittal has been provided to the State of Wisconsin in accordance with 10 CFR 50.91(b).

Approval of the proposed amendment is requested by July 2013. Once approved, the amendment shall be implemented within 60 days.

A001
NRR

If you have any questions or require additional information, please contact Mr. E. Thomas Shaub at (804) 273-2763.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering of Dominion Energy Kewaunee, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 30TH day of July, 2012.

My Commission Expires: May 31, 2014.

Vicki L. Hull
Notary Public

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

Attachments:

1. Discussion of Change, Technical Analysis, Significant Hazards Determination and Environmental Considerations
2. Marked up Technical Specifications Pages
3. Proposed (Clean) Technical Specifications Pages
4. Marked up Technical Specifications Bases Pages

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ATTACHMENT 1

LICENSE AMENDMENT REQUEST 254
PROPOSED TECHNICAL SPECIFICATIONS TO ADOPT TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

DISCUSSION OF CHANGE, SAFETY EVALUATION, SIGNIFICANT HAZARDS DETERMINATION AND ENVIRONMENTAL CONSIDERATIONS

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

The proposed change revises Specification 5.5.7, "Steam Generator (SG) Program" and 5.6.5, "Steam Generator Tube Inspection Report." The proposed changes are needed to address implementation issues associated with the inspection periods, and address other administrative changes and clarifications. For consistency, additional administrative changes are being made to Specification 3.4.17 "Steam Generator (SG) Tube Integrity."

The proposed amendment is consistent with TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

Dominion Energy Kewaunee (DEK) has reviewed TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," (ADAMS Accession No. ML110610350) and the model safety evaluation dated October 19, 2011 (ADAMS Accession No. ML112101513) as identified in the Federal Register Notice of Availability, dated October 27, 2011 (76 FR 66763). As described in the subsequent paragraphs, DEK has concluded that the justifications presented in TSTF-510 and the model safety evaluation prepared by the NRC staff are applicable to Kewaunee Power Station (KPS) and justify this amendment for the incorporation of the changes to the KPS Technical Specifications (TS).

2.2 Optional Changes and Variations

DEK is not proposing any technical variations or deviations from the TS changes described in the TSTF-510, Revision 2, or the applicable parts of the NRC staff's model safety evaluation. However, DEK is proposing the following administrative variations from the TS changes described in the TSTF-510, Revision 2, or the applicable parts of the NRC staff's model safety evaluation dated October 19, 2011.

The KPS TS numbering system is different than the Improved Technical Specifications (ITS) on which TSTF-510 was based. Specifically, the "Steam Generator (SG) Program" in the KPS TS is numbered 5.5.7 rather than 5.5.9 and the "Steam Generator Tube Inspection Report" is numbered 5.6.5 rather than 5.6.7. These differences are administrative and do not affect the applicability of TSTF-510 to the KPS TS.

In addition, the proposed change corrects an administrative inconsistency in TSTF-510, Paragraph d.2 of the Steam Generator Tube Inspection Program. In Section 2.0, "Proposed Change," TSTF-510 states that references to "tube repair criteria" in

Paragraph d.2 is revised to "tube plugging [or repair] criteria." However, in the following sentence in Paragraph d.2, this change was inadvertently omitted,

> "If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated" (emphasis added).

The underlined phrase should state "tube plugging [or repair] criteria," consistent with the other changes made in TSTF-510. DEK is changing the phrase to "tube plugging criteria." This change is administrative and should not result in this application being removed from the Consolidated Line Item Improvement Process.

This administrative error was indentified in a February NRC-TSTF meeting and documented in a letter from the TSTF to the NRC dated March 28, 2012 (TSTF letter No. 12-09)

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

DEK requests adoption of an approved change to the standard technical specifications (STS) into the Kewaunee technical specifications (TS), to revise the Specification 5.5.7, "Steam Generator (SG) Program," 5.6.5, "Steam Generator Tube Inspection Report," and LCO 3.4.17, "Steam Generator (SG) Tube Integrity," to address inspection periods and other administrative changes and clarifications.

As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change revises the Steam Generator (SG) Program to modify the frequency of verification of SG tube integrity and SG tube sample selection. A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. The proposed SG tube inspection frequency and sample selection criteria will continue to ensure that the SG tubes are inspected such that the probability of a SGTR is not increased. The consequences of a SGTR are bounded by the conservative assumptions in the design basis accident analysis. The proposed change will not cause the consequences of a SGTR to exceed those assumptions. Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes to the Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs or their method of operation. In addition, the proposed change does not impact any other plant system or component.

   Therefore, it is concluded that this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

   Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change will continue to require monitoring of the physical condition of the SG tubes such that there will not be a reduction in the margin of safety compared to the current requirements.

   Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, DEK concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Applicable regulatory Requirements/Criteria

The Kewaunee Nuclear Power Plant was designed, constructed, and is being operated to comply with Wisconsin Public Service Corporation's (WPSC) *understanding of the intent* of the Atomic Energy Commission (AEC) General Design Criteria (GDC) for

Nuclear Power Plant Construction Permits, as *proposed* on July 10, 1967. However, the AEC Safety Evaluation Report (SER), issued July 24, 1972, acknowledged that the AEC staff assessed the plant, as described in the FSAR (Amendment No. 7), against the Appendix A design criteria and "...are satisfied that the plant design generally conforms to the intent of these criteria." Specifically, Section 1.8 of the UFSAR discusses the design of the station relative to the design criteria published in 1971. The following information demonstrates compliance with GDC 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Specifically, the GDC state that the Reactor Coolant Pressure Boundary (RCPB) shall have "an extremely low probability of abnormal leakage . . . and gross rupture" (GDC 14/USAR 1.8, Criterion 9), "shall be designed with sufficient margin" (GDCs 15/USAR 1.8, Criterion 33 and 31/USAR 1.8, Criterion 35), shall be of "the highest quality standards practical" (GDC 30/USAR 1.8, Criterion 1 and 16), and shall be designed to permit "periodic inspection and testing . . . to assess . . . structural and leak tight integrity" (GDC 32/USAR 1.8, Criterion 36). Structural integrity refers to maintaining adequate margins against burst, and collapse of the SG tubing.

The TS repair limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions. The reactor coolant pressure boundary is designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. Reactor coolant pressure boundary components have provisions for the inspection testing and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. Structural integrity refers to maintaining adequate margins against burst, and collapse of the SG tubing. Leakage integrity refers to limiting primary to secondary leakage to within acceptable limits during all plant conditions.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements of this appendix apply to all activities affecting the safety related functions of these components. These requirements are described in Criteria IX, XI, and XVI of Appendix B and include control of special processes, inspection, testing, and corrective action.

Under 10 CFR 50.65, the Maintenance Rule, licensees classify SGs as risk significant components because they are relied upon to remain functional during and after design basis events. SGs are to be monitored under 10 CFR 50.65(a)(2) against industry established performance criteria. Meeting the performance criteria of NEI 97-06, Revision 3, provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary.

4.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

Serial No. 12-458

ATTACHMENT 2

LICENSE AMENDMENT REQUEST 254
PROPOSED TECHNICAL SPECIFICATIONS TO ADOPT TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

MARKED-UP TECHNICAL SPECIFICATIONS PAGES:
TS 3.4.17-1
TS 3.4.17-2
TS 5.5-5
TS 5.5-6
Insert A for TS 5.5.7
TS 5.6-4

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

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

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

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

ACTIONS

--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~~ plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the 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 not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

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

5.5 Programs and Manuals (continued)

5.5.7 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 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 in-service 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.

   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 150 gallons per day per SG.

   3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

c. Provisions for SG tube ~~repair~~ plugging 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~~ plugging 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 integrity is maintained until the next SG inspection. A~~n~~ degradation assessment ~~of degradation~~ shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG ~~replacement~~ installation.

Replace with Insert A

2. ~~Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.~~

3. If crack indications are found in any SG tube, then the next inspection for each affected or potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections ~~is less~~). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

Insert A for TS 5.5.7 - SG Program (690)

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least 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 a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the 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 and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   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

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Post Accident Monitoring Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," 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.

5.6.5 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 the Specification 5.5.7, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. ~~Active d~~Degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

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

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

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

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

~~h. The effective plugging percentage for all plugging in each SG~~.

Serial No. 12-458

ATTACHMENT 3

LICENSE AMENDMENT REQUEST 254
PROPOSED TECHNICAL SPECIFICATIONS TO ADOPT TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

PROPOSED (CLEAN) TECHNICAL SPECIFICATIONS PAGES:
TS 3.4.17-1
TS 3.4.17-2
TS 5.5-5 through TS 5.5.16
TS 5.6-4

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

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

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

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

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the 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 not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

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

5.5 Programs and Manuals (continued)

5.5.7 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:

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 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 in-service 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), 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.

   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 150 gallons per day per SG.

   3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

c. Provisions for SG tube plugging 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 plugging 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 integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least 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 a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the 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 and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

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.

3. If crack indications are found in any SG tube, then the next inspection for each affected or potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals (continued)

5.5.8 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of safety related filter ventilation systems in accordance with Regulatory Positions C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, ANSI N510-1975, ATSM D3803-1989, and AG-1.

The test described in Specification 5.5.9.a shall be performed once per 18 months and after each complete or partial replacement of the high efficiency particulate air (HEPA) filter bank and any maintenance on the system that could affect the HEPA bank bypass leakage.

The test described in Specification 5.5.9.b shall be performed after each complete or partial replacement of a charcoal adsorber bank or maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

The test described in Specification 5.5.9.c shall be performed once per 18 months for filters in a standby status or after 720 hours of filter operation, and following painting, fire, or chemical release in any ventilation zone communicating with the system.

The test described in Specification 5.5.9.d shall be performed once per 18 months.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

The test described in Specification 5.5.9.e shall be performed after any maintenance or testing that could affect the air distribution within the systems.

a. Demonstrate for each of the safety related systems listed below that an inplace test of the HEPA filters shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Position C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%.

Safety Related System	Flow Rate (cfm)
Shield Building Ventilation System (SBVS)	5700
Auxiliary Building Special Ventilation (ASV) System	9000
Control Room Post Accident Recirculation (CRPAR) System	2500

b. Demonstrate for each of the safety related systems listed below that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Position C.5.d of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified below ± 10%.

Safety Related System	Flow Rate (cfm)
SBVS	5700
ASV System	9000
CRPAR System	2500

c. Demonstrate for each of the safety related systems listed below that a laboratory test of a sample of the charcoal adsorber shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 and AG-1 at a temperature of 30°C (86°F) and relative humidity of 95%.

Safety Related System	Penetration
SBVS	≤ 2.5%
ASV System	≤ 2.5%
CRPAR System	≤ 5%

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each of the safety related systems listed below that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1975 at the system flowrate specified below ± 10%.

Safety Related System	Combined Delta P (in. wc)	Flow Rate (cfm)
SBVS	< 6.3	5700
ASV System	< 6.3	9000
CRPAR System	< 2.4	2500

e. Demonstrate for each of the safety related systems listed below that when tested at the system flowrate specified below (± 10%) the air distribution is uniform within ± 20%.

Safety Related System	Flow Rate (cfm)
SBVS	5700
ASV System	9000

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radioactive Waste Disposal System, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radioactive Waste Disposal System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

5.5 Programs and Manuals

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tank's contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Waste Disposal System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.11 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits;

   2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil; and

   3. A clear and bright appearance with proper color or a water and sediment content within limits;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 92 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5 Programs and Manuals (continued)

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

 1. A change in the TS incorporated in the license; or

 2. A change to the updated USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

d. Proposed changes that meet the criteria of Specification 5.5.12.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

 1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

 2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

 3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

 4. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable; and

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.13.b.1 and 5.5.13.b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 44.6 psig. The containment design pressure is 46 psig.

c. The maximum allowable containment leakage rate, L_a, at 46 psig (Peak Test Pressure), shall be 0.2% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

2. Air lock door seal leakage testing acceptance criteria for each door seal is a leakage rate of < 0.005 L_a when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing program.

5.5.15 Battery Monitoring and Maintenance Program

This Program provides controls for battery restoration and maintenance. The program shall be in accordance with IEEE Standard (Std) 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by Regulatory Guide 1.129, Revision 2 (RG), with RG exceptions and program provisions as identified below:

a. The program allows the following RG 1.129, Revision 2 exceptions:

1. Battery temperature correction may be performed before or after conducting discharge tests.

2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.

3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery after each performance of SR 3.8.6.5."

4 In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration", the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

5.5 Programs and Manuals

5.5.15 Battery Monitoring and Maintenance Program (continued)

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

5.5.16 Setpoint Control Program

This program shall establish the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analysis. The program shall ensure that testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verify that instrumentation will function as required.

a. The program shall list the Functions in the following specifications to which it applies:

1. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation";

2. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions";

3. LCO 3.3.5, "Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation";

4. LCO 3.3.6, "Containment Purge and Vent Isolation Instrumentation"; and

5. LCO 3.3.7, "Control Room Post Accident Recirculation (CRPAR) Actuation Instrumentation."

b. The program shall list the value of the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) for each Function described in Paragraph a. The NRC staff has not approved processing changes to Kewaunee Power Station instrumentation setpoints under 10 CFR 50.59 using an approved setpoint methodology as described in Option B of TSTF-493. NRC approval using

5.5 Programs and Manuals

5.5.16 Setpoint Control Program (continued)

10 CFR 50.90 is required to change the listed value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in Paragraph a.

c. The program shall establish methods to ensure that Functions described in Paragraph a. will function as required by verifying the as-left and as-found settings are consistent with the list of values established by Paragraph b. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.

d. The program shall identify the Functions described in Paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.

   1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.

   2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

   3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.

   4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).

e. The program shall be specified in the Technical Requirements Manual.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Post Accident Monitoring Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," 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.

5.6.5 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 the Specification 5.5.7, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG;

b. Degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

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

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; and

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

ATTACHMENT 4

LICENSE AMENDMENT REQUEST 254
PROPOSED TECHNICAL SPECIFICATIONS TO ADOPT TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES:
TS B 3.4.17-2
TS B 3.4.17-4
TS B 3.4.17-5
TS B 3.4.17-6
Insert B for TS Bases 3.4.17

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

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 3.4.13, "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 largely contained within the secondary side of the affected SG and released to the atmosphere via safety valves and secondary system leakage.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE of 150 gallons per day per SG or is assumed to increase to 150 gallons per day per SG 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 LCO 3.4.16, "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 10 CFR 50.67 (Ref. 2) 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 5.5.7, "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.

BASES

LCO (continued)

LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

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 Program. The SG ~~repair~~ plugging 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.

BASES

ACTIONS (continued)

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.

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 (Ref. 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 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

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. 5). 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 5.5.7 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

$\boxed{\text{Insert B}}$

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ plugging criteria is removed from service by plugging. The tube ~~repair~~ plugging criteria delineated in Specification 5.5.7 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~~ plugging 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.

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~~ plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50.67.
3. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
4. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
5. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Insert B for TS Bases - Surveillance Requirements

If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.7 until subsequent inspections support extending the inspection interval.