Dominion Energy Kewaunee, Inc.
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Dominion®

January 12, 2006

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 05-562
KPS/LIC/GR: R4
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Dominion Energy Kewaunee, Inc. (DEK) is submitting a request for an amendment to the technical specifications (TS) for Kewaunee Power Station (Kewaunee).

The proposed amendment would revise the TS requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

Attachment 1 provides a description of the proposed change and confirmation of applicability. Attachment 2 provides a description of the variations necessary for the Kewaunee Custom TS to incorporate the TS changes described in TSTF 449, Revision 4. Attachment 3 provides the existing TS pages marked-up to show the proposed change. Attachment 4 provides the proposed TS pages. Attachments 5 and 6 provide the marked-up and proposed TS bases pages, respectively, for information only.

DEK requests approval of the proposed license amendment by June 30, 2006, to facilitate scheduling of the fall 2006 refueling outage, with the amendment being implemented within 90 days.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Wisconsin Official.

A001

If you should have any questions regarding this submittal, please contact Mr. Gerald Riste at 920-388-8424.

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering

Attachments:
1. Description and Assessment
2. Variations due to Custom TS
3. Marked Up Technical Specification Pages
4. Proposed Technical Specification Pages
5. Marked Up Technical Specification Bases Pages
6. Proposed Technical Specification Bases Pages

Commitments made in this letter: Correct deviations from EPRI Primary-to-Secondary Leakage Guidelines Rev. 3, Final Report December 2004, prior to implementation of license amendment 218 regarding SG Tube Integrity.

cc: Regional Administrator
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President – Nuclear Engineering of Dominion Energy Kewaunee, Inc. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 12th day of January, 2006.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

DESCRIPTION AND ASSESSMENT

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

License Amendment Request 218
Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity
Description And Assessment

1.0 INTRODUCTION

The proposed license amendment revises the requirements in the Kewaunee Power Station (Kewaunee) Technical Specifications (TS) related to steam generator tube integrity. The changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this technical specification improvement was announced in the Federal Register on May 6, 2005 as part of the consolidated line item improvement process (CLIIP).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include:

- New TS 1.0.t - New TS definition of Leakage
- Revised TS 3.1.d - "RCS Operational Leakage"
- New TS 3.1.g - "Steam Generator Tube Integrity"
- New TS 4.18 – "RCS Operational Leakage"
- New TS 4.19, "Steam Generator (SG) Tube Integrity," replacing existing TS 4.2.b "Steam Generator Tubes"
- New TS 6.9.b.3 - "Steam Generator Tube Inspection Report"
- New TS 6.22 - "Steam Generator (SG) Program"

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4 is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

5.0 TECHNICAL ANALYSIS

Dominion Energy Kewaunee, Inc. (DEK) has reviewed the safety evaluation (SE) published on March 2, 2005, (70 FR 10298) as part of the CLIIP Notice for Comment. This included the NRC staffs SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. DEK has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to Kewaunee Power Station (Kewaunee) and justify this amendment for the incorporation of the changes to the Kewaunee TS.

Kewaunee's TS Basis will not mirror standard TS Basis exactly due to differences in approved accident analysis. Kewaunee's assumed post-accident primary-to-secondary leakage is 150 gpd. This is the same as the operational leakage limit described in TS 3.1.d. This is considered acceptable because Kewaunee is committed to implement the Electric Power Research Institute guidelines for primary-to-secondary leakage monitoring and corrective actions (Reference 10.3). Procedures are in place to implement these guidelines. As a result of a recent self-assessment, some deviations from the guidelines were identified and corrective actions were initiated to resolve them. These deviations will be corrected prior to the implementation of this license amendment.

At Kewaunee, installed Radiation Monitoring Systems (RMSs) provide continuous on-line monitoring of primary-to-secondary leakage to plant operators. Kewaunee operating procedure E-0-14, Steam Generator Tube Leak, provides actions to take when a small primary-to-secondary steam generator tube leak exists. A small tube leak is defined as one that is greater than 5 gallons per day in any steam generator. The procedure requires confirmation and monitoring of the leak rate to determine if the leak

has stabilized. Operations, Engineering, and Radiation Protection are notified of the condition and participate in the evaluation and monitoring of the situation.

If the leak rate increases to 30 gallons per day, E-0-14 directs the operators to place the secondary radiation monitors on continuous trend, monitor every 15 minutes, and verify the secondary radiation monitors alarm setpoints. E-0-14 directs chemistry to increase the grab sampling frequency, determine which steam generator is leaking, and determine the new leakrate.

If primary-to-secondary leakage is 75 gpd or greater for greater than one hour, the operators place the secondary radiation monitors on continuous trend and monitor every 15 minutes. Actions are initiated to perform a normal plant shutdown and achieve the Hot Shutdown condition (reactor shutdown and RCS Tavg greater than or equal to 540 °F) within 24 hours.

If primary-to-secondary leakage is 100 gpd or greater, a rapid plant shutdown is initiated. E-0-14 directs the operators to reduce plant power to less than 50% within one hour and requires that the plant be placed in the Hot Shutdown condition within the next two hours.

6.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

Although Kewaunee generally conforms to the regulatory requirements published in the May 6, 2005, NRC Notice of Availability, the Kewaunee plant was licensed to design requirements that were in effect prior to the adoption of 10CFR50 Appendix A, "General Design Criteria."

The US Atomic Energy Commission (AEC) issued their Safety Evaluation (SE) of the Kewaunee Power Station (Kewaunee) on July 24, 1972, with supplements dated December 18, 1972, and May 10, 1973. In the AEC's SE, section 3.1, "Conformance with AEC General Design Criteria," the staff described the conclusions the AEC reached associated with the General Design Criteria in effect at the time. The AEC stated:

> *The Kewaunee plant was designed and constructed to meet the intent of the AEC's General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 50% complete and the Final Safety*

Analysis Report (Amendment No. 7) had been filed with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant or resubmit the FSAR. However, our technical review did assess the plant against the General Design Criteria now in effect and we are satisfied that the plant design generally conforms to the intent of these criteria.

As such, the applicable design criteria Kewaunee is licensed to from the Final Safety Analysis Report (Amendment 7), which has been updated and now titled the Updated Safety Analysis Report (USAR), are listed below.

Criterion 1 - Quality Standards

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability or codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Criterion 9 - Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 33 - Reactor Coolant Pressure Boundary Capability

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant.

Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures.

Criterion 36 - Reactor Coolant Pressure Boundary Surveillance

Reactor Coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the boundary components during service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

6.1 Verification and Commitments

The following information is provided to support the NRC staff's review of this amendment application:

Plant Name, Unit No.	Kewaunee Power Station (KPS)
Steam Generator Model:	Westinghouse Model 54-F
Effective Full Power Years (EFPY) of service for currently installed SGs	3.4 EFPY through December 31, 2005
Tubing Material	Inconel Alloy 690 Thermally Treated
Number of tubes per SG	3592
Number and percentage of tubes plugged in each SG	SG A – 0 (0.0%) SG B – 0 (0.0%)
Number of tubes repaired in each SG	SG A – 0 (0.0%) SG B – 0 (0.0%)
Degradation mechanism(s) identified	No degradation mechanisms are currently active.
Current primary-to-secondary leakage limits:	per SG: 150 gpd Total: 300 gpd Leakage rate is at room temperature.
Approved Alternate Tube Repair Criteria (ARC):	None
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	150 gpd/SG, 300 gpd total SG leakage Leakage rate is at room temperature. *Primary-to-secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions.*

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

7.1 Incorporation of TSTF-449, Revision 4

DEK has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. DEK has concluded that the proposed determination presented in the notice is applicable to Kewaunee and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91 (a).

7.2 Conversion of Kewaunee Power Station custom Technical Specifications to Improved Standard Technical Specification Format

In order to incorporate the CLIIP license amendment request, several changes are needed to the Kewaunee Power Station custom technical specifications. These changes include:

1) Add a new definition, TS 1.0.t, for LEAKAGE,
2) Modify the wording of the current TS 3.1.d,
3) Add new TS 4.18,
4) Make related Bases changes to be consistent with NUREG-1431, Revision 3.

These changes are necessary to make the current Kewaunee TS compatible with the proposed changes of TSTF-449, Revision 4.

A significant hazards consideration determination has been performed for these TS changes to facilitate incorporation of the changes described in TSTF-449, Revision 4. The proposed changes do not involve a significant hazards determination because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change involves rewording of certain Technical Specification sections to be consistent with NUREG-1431, Revision 3. These modifications involve no technical changes to the existing Technical Specifications. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change involves rewording of certain Technical Specification sections to be consistent with NUREG-1431, Revision 3. The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements from those already approved in the CLIIP. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed change involves rewording of certain Technical Specification sections to be consistent with NUREG-1431, Revision 3. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

8.0 ENVIRONMENTAL EVALUATION

DEK has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. DEK has concluded that the staff's findings presented in that evaluation are applicable to Kewaunee Power Station, and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. In general, DEK is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298). However, since Kewaunee has custom TS, as opposed to Improved Standard TS (ISTS), the changes proposed by the CLIIP have been implemented such that they are consistent with the existing Kewaunee TS format requirements. Specifically, the variations from TSTF-449, Revision 4, are provided in Attachment 2. These variations do not conflict with the applicability of the NRC's model safety evaluation to the proposed change. The variations are primarily TS format or terminology differences due to Kewaunee's custom TS format and wording.

10.0 REFERENCES

Federal Register Notices:

10.1 Notice for Comment published on March 2, 2005 (70 FR 10298)

10.2 Notice of Availability published on May 6, 2005 (70 FR 24126)

10.3 Electric Power Research Institute PWR Primary-To-Secondary Leak Guidelines – Revision 3, Final Report, December 2004

ATTACHMENT 2

LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

VARIATIONS FROM THE TS CHANGES DESCRIBED IN TSTF-449, REVISION 4 FOR KEWAUNEE POWER STATION CUSTOM TS

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

LICENSE AMENDMENT REQUEST 218

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

DEK is proposing minor variations and/or deviations from the TS changes described in TSTF-449, Revision 4, to provide consistent terminology and format within Kewaunee's custom TS. For example, Kewaunee TS separate limiting conditions for operation (LCOs) and surveillance requirements (SRs) into different TS sections (3 and 4, respectively). In addition, Kewaunee TS do not use the improved standard technical specification (ISTS) MODE terminology convention for reactor operating conditions. Kewaunee TS use specific definitions for each operating condition instead. However, the reactor operating MODEs specified in the CLIIP are consistent with the defined reactor operating conditions used in the Kewaunee license amendment request. The minor variations and/or deviations from the specific wording/format provided in the CLIIP do not change the meaning, intent or applicability of the CLIIP.

Kewaunee Power Station Technical Specifications item 1.0.j, "MODES," defines the stations operating modes. The following table lists these operating modes.

KEWAUNEE MODE	REACTIVITY Δk/k	COOLANT TEMP T_{avg} °F	FISSION POWER %
OPERATING	< 0.25%	~T_{oper}	≥ 2
HOT STANDBY	< 0.25%	~T_{oper}	< 2
HOT SHUTDOWN	(1)	≥ 540	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
COLD SHUTDOWN	≤ -1%	≤ 200	~0
REFUELING	≤ -5%	≤ 140	~0
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		

(1) Refer to the required SHUTDOWN MARGIN as specified in the Core Operating Limits Report.

For comparison with the Operating Modes of Kewaunee Power Station custom Technical Specifications, the Operating Modes of NUREG 1431, Revision 3 are provided below.

ISTS MODE	TITLE	CONDITION (KEFF)	THERMAL POWER [(A)]	REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ [350]
4	Hot Shutdown [(b)]	< 0.99	NA	[350] > Tavg >[200]
5	Cold Shutdown [(b)]	< 0.99	NA	≤ [200]
6	Refueling [(c)]	NA	NA	NA

(a) Excluding decay heat.
(b) All reactor vessel head closure bolts fully tensioned.
(c) One or more reactor vessel head closure bolts less than fully tensioned.

A summary of the minor variations and/or deviations from the TS changes described in TSTF-449, Revision 4 is provided as follows.

Variations from the TS changes described in TSTF-449, Revision 4 For Kewaunee Power Station Custom TS			
ISTS Section	CLIIP/TSTF 449 TS Revision	Kewaunee TS Section	Inclusion of Proposed Change into Kewaunee Custom TS
1.1 Definition	Revises the LEAKAGE definition in TS to include the parenthetical phrase "(primary-to-secondary LEAKAGE)" in item a.3 and item c and deletes the term "(SG)" in both items.	1.0.t	Kewaunee TS do not currently include a definition for LEAKAGE. The proposed change incorporates a definition for LEAKAGE into the Kewaunee TS that is identical to the ISTS Definition including the proposed TSTF change.
B3.4.4	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODES 1 and 2 LCO Bases section.	N/A	Kewaunee TS do not include this TS/phrase; therefore, no change is required.
B3.4.5	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODE 3 LCO Bases section.	N/A	Kewaunee TS do not include this TS/phrase; therefore, no change is required.

Variations from the TS changes described in TSTF-449, Revision 4 For Kewaunee Power Station Custom TS			
ISTS Section	CLIIP/TSTF 449 TS Revision	Kewaunee TS Section	Inclusion of Proposed Change into Kewaunee Custom TS
B3.4.6	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODE 4 LCO Bases section.	N/A	Kewaunee TS do not include this TS/phrase; therefore, no change is required.
B3.4.7	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODE 5, Loops Filled LCO Bases section.	N/A	Kewaunee TS do not include this TS/phrase; therefore, no change is required.
LCO 3.4.13	Revises RCS Operational LEAKAGE for primary-to-secondary LEAKAGE to ≤ 150 gallons per day primary-to-secondary LEAKAGE through any one SG. Includes primary-to-secondary LEAKAGE in the CONDITIONS column of the LCO ACTIONS. RCS Operational LEAKAGE TS SURVEILLANCE REQUIREMENTS - Be in Mode 3 within 6 hours and in Mode 5 in 36 hours.	3.1.d.1 through .3.1.d.3	Current Kewaunee TS primary-to-secondary leakage limit is ≤ 150 gpd through any one steam generator. Kewaunee TS 3.1.d.1 through TS 3.1.d.4 LCOs and ACTIONS associated with RCS Operational LEAKAGE have been replaced with TS 3.1.d.1 through .3 specifications consistent with the revised ITS Section 3.4.13. Kewaunee's TS format and MODE terminology is retained vs. the format and MODE terminology used in ISTS. Specifically, MODE 3 is changed to HOT SHUTDOWN and MODE 5 is changed to COLD SHUTDOWN. This requirement has been included in Kewaunee TS 3.1.d.3. (Kewaunee's TS format and Mode terminology is retained vs. the format and Mode terminology used in

Variations from the TS changes described in TSTF-449, Revision 4 For Kewaunee Power Station Custom TS			
ISTS Section	CLIIP/TSTF 449 TS Revision	Kewaunee TS Section	Inclusion of Proposed Change into Kewaunee Custom TS
			ISTS.) The verbiage has changed but the requirement is still to achieve cold shutdown within 36 hours of the condition not being met. Renumbered TS 3.1.d.5 to 3.1.d.4.
SR 3.4.13	Added new note indicating SR not applicable to primary-to-secondary LEAKAGE. Revised the SR to verify primary-to-secondary LEAKAGE every 72 hours. Added a Note stating "Not required to be performed until 12 hours after establishment of steady state operation."	4.18	The revised ISTS SR 3.4.13.1 has been included in new Kewaunee TS 4.18 for performance of RCS water inventory balance every 72 hours. TS 4.18 also includes the associated ISTS notes as revised by the TSTF. New Kewaunee TS 4.18 includes the revised ISTS SR 3.4.13.2 to verify once every 72 hours that primary-to-secondary LEAKAGE is ≤150 gallons per day through any one SG, as well as the new note.
B3.4.13	Revise the Bases for the RCS Operational LEAKAGE TS to address TSTF-449, Rev. 4 changes.	3.1.d 4.18	The existing TS Basis section for Kewaunee TS 3.1.d is being replaced with the ISTS B3.4.13 Bases wording as revised by TSTF-449 as appropriate for Kewaunee. ISTS TS 3.4.13 bases is divided into two parts to address Kewaunee TS format. The LCO portion is included in TS 3.1.d, and the SRs portion is included in TS 4.18. Consequently, B3.4.13 has been divided between the two

Variations from the TS changes described in TSTF-449, Revision 4 For Kewaunee Power Station Custom TS			
ISTS Section	CLIIP/TSTF 449 TS Revision	Kewaunee TS Section	Inclusion of Proposed Change into Kewaunee Custom TS
			Kewaunee TS sections accordingly. The Background, Applicable Safety Analyses, Limiting Conditions for Operation, Applicability, Actions and References sections were included in the TS 3.1.d Basis, and the Surveillance Requirements and References (repeated) were included in the TS 4.18 Basis.
LCO 3.4.20	New TS added for SG tube integrity requires surveillance frequency in accordance with TS 5.5.9, Steam Generator Program. Frequency is dependent upon tubing material, the previous inspection results and the anticipated defect growth rate.	3.1.g	New Kewaunee TS 3.1.g, SG Tube Integrity, has been added and is consistent with ITS TS 3.4.20. (Note: Kewaunee TS LCOs and SRs are contained in different TS sections.)
SR 3.4.20	SG Tube Integrity – SR 3.4.20.1 requires that tube integrity be verified in accordance with the Steam Generator Program.	4.19	New TS 4.19, SG Tube Integrity, which includes the surveillance requirement that tube integrity be verified in accordance with the Steam Generator Program, has replaced existing Kewaunee TS 4.2.b and TS Table 4.2-2 in their entirety. The new TS 4.19 SRs are consistent with ISTS TS 3.4.20 SRs. The ISTS phrase "prior to entering Mode 4" has been changed to "prior to entering INTERMEDIATE SHUTDOWN" for consistency with Kewaunee TS format.

Variations from the TS changes described in TSTF-449, Revision 4 For Kewaunee Power Station Custom TS			
ISTS Section	CLIIP/TSTF 449 TS Revision	Kewaunee TS Section	Inclusion of Proposed Change into Kewaunee Custom TS
B3.4.20	New Bases for the new SG Tube Integrity TS in accordance with TSTF-449, Rev. 4.	3.1.g 4.19	TS 3.4.20 is divided into two parts to address Kewaunee TS format. The LCO portion is included in TS 3.1.g, and the SRs are included in TS 4.19 as discussed above. Consequently, the B3.4.20 has been divided between the two Kewaunee TS sections accordingly. The Background, Applicable Safety Analyses, Limiting Conditions for Operation, Applicability, Actions and References sections were included with the TS 3.1.g Basis as appropriate, and the Surveillance Requirements and References (repeated) were included in the TS 4.19 Basis as appropriate.
5.5.9	New Steam Generator (SG) Program description/criteria	6.22	New TS 6.22, Steam Generator Program, has been incorporated into Kewaunee TS. The TS 6.22 text is identical to ITS TS 5.5.9 text including the proposed TSTF change.
5.6.9	New Steam Generator Tube Inspection Report description/criteria.	6.9.b.4	New Kewaunee TS 6.9.b.4, Steam Generator Tube Inspection Report, has been incorporated into Kewaunee TS and replaces the reporting requirements contained in TS 4.2.b. The TS 6.9.b.4 text is identical to the revised ISTS 5.6.9 text with the exception of the use of the term "after the initial entry into MODE 4" since Kewaunee's TS do not use the MODE 1-6 plant condition terminology. This phrase has been revised to "after the initial entry into INTERMEDIATE SHUTDOWN" for consistency with Kewaunee TS reactor operation mode terminology.

ATTACHMENT 3

LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

MARKED UP PROPOSED TECHNICAL SPECIFICATION PAGES

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

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Section	Title	Page

Section	Title	Page

LIST OF TABLES

t. LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified Leakage

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary Leakage

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

d. ~~Leakage of Reactor Coolant~~RCS Operational LEAKAGE

1. When the average RCS temperature is > 200°F, RCS operational leakage shall be limited to:

   A. No pressure boundary LEAKAGE,

   B. 1 gpm unidentified LEAKAGE,

   C. 10 gpm identified LEAKAGE, and

   D. 150 gallons per day primary to secondary LEAKAGE through any one SG.

2. If the limits contained in TS 3.1.d.1 for identified or unidentified LEAKAGE are exceeded, then reduce the LEAKAGE to within their limits within 4 hours.

3. If the limits contained in TS 3.1.d.1 for pressure boundary or primary to secondary LEAKAGE are exceeded, or the time limit contained in TS 3.1.d.2 is exceeded, then initiate action to:

   - Achieve HOT SHUTDOWN within 6 hours, and

   - Achieve COLD SHUTDOWN within an additional 30 hours.

~~1. Any Reactor Coolant System leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 four hours of the indication. Any indicated leak shall be considered to be a real leak until it is determined that no unsafe condition exists. If the Reactor Coolant System leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, then the reactor shall be placed in the HOT SHUTDOWN condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 48 hours, then the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.~~

~~2. Reactor coolant-to-secondary leakage through the steam generator tubes shall be limited to 150 gallons per day through any one steam generator. With tube leakage greater than the above limit, reduce the leakage rate within 4 four hours or be in COLD SHUTDOWN within the next 36 hours.~~

~~3. If the sources of leakage other than that in 3.1.d.2 have been identified and it is evaluated that continued operation is safe, then operation of the reactor with a total Reactor Coolant System leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, then the reactor shall be placed in the HOT SHUTDOWN condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the COLD SHUTDOWN condition utilizing normal operating procedures.~~

~~4. If any reactor coolant leakage exists through a non-isolable fault in a Reactor Coolant System component (exterior wall of the reactor vessel, piping, valve body, relief valve leaks, pressurizer, steam generator head, or pump seal leakoff), then the reactor shall be shut down; and cooldown to the COLD SHUTDOWN condition shall be initiated within 24 hours of detection.~~

~~4~~5. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is OPERABLE.

g. Steam Generator Tube Integrity

1. When the average reactor coolant system temperature is > 200°F the following shall be maintained:

   A. SG Tube integrity shall be maintained, and

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

Note: Separate entry condition is allowed for each SG tube.

2. If the requirements of TS 3.1.g.1.B can not be met, then:

   A. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

   B. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering INTERMEDIATE SHUTDOWN following the next refueling outage or SG tube inspection.

3. If the requirements of TS 3.1.g.2.A or TS 3.1.g.1.A can not be met, then initiate action:

   - Achieve HOT SHUTDOWN within 6 hours

   - Achieve COLD SHUTDOWN within an additional 30 hours.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

b. Deleted~~Steam Generator Tubes~~

~~Examinations of the steam generator tubes shall be in accordance with the in-service inspection program described herein. The following terms are defined to clarify requirements of the inspection program.~~

~~Imperfection is a deviation from the dimension, finish, or contour required by a design drawing or specification.~~

~~Degradation means service-induced cracking, wastage, wear or corrosion of a tube wall.~~

~~% Degradation is the amount in percent of tube wall thickness affected or removed by degradation.~~

~~Degraded Tube means a tube containing degradation that is ≥ 20% of nominal wall thickness.~~

~~Defect means an imperfection that violates criteria used to determine acceptability of a tube for continued use in operation.~~

~~Tube Inspection means the detailed examination of a steam generator tube from the point of entry (e.g., hot leg side) around the U-bend to the level of the top tube support plate of the opposite leg (cold leg).~~

~~Tube is a single hollow metal cylinder that is an element of an array of similar cylinders inside each steam generator, through which Reactor Coolant flows, and by which heat is transferred from the Reactor Coolant to the secondary system feedwater. Taken as a whole, steam generator tubes form a major portion of the reactor coolant pressure boundary.~~

~~Plugged Tube is a tube that has been removed from service by installing a mechanical device in each end of the tube to seal the tube in a manner that isolates it from the reactor coolant system.~~

~~1. Steam Generator Sample Selection and Inspection~~

~~In-service inspection of steam generators may be limited to one steam generator per inspection period on an alternating basis. The tubes shall be selected for inspection as set forth in TS 4.2.b.2.a, provided that previous inspections indicate the two steam generators are performing in an acceptably similar manner.~~

~~2. Steam Generator Tube Sample Selection and Inspection~~

~~Each in-service inspection:~~

~~Shall include a number of tubes that is at least equal to 3% of the total number of non-plugged tubes contained in both steam generators. Tubes shall be selected for inspection on a random basis except as noted in TS 4.2.b.2.b.~~

~~Shall concentrate the inspection by selecting at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.~~

~~Shall include all non-plugged tubes in which previous inspections revealed degradation that exceeded 20% of nominal wall thickness. For these tubes, only the area previously identified as degraded must be inspected, unless their inspection is also performed to satisfy requirements of TS 4.2.b.2.a and TS 4.2.b.2.b above.~~

~~May not require inspection of the full length of each tube during the second and third sample inspections but may concentrate the inspection only on those portions of the tubes previously found degraded.~~

~~Shall perform a tube inspection on each selected tube. If the eddy current inspection probe will not pass through the entire length of a tube, including the U-bend, it shall be so recorded and the tube shall be characterized as degraded. An adjacent tube shall also be inspected.~~

~~Shall classify sample inspection results as belonging to one of the following three categories, and actions shall accordingly be taken as described in Table TS 4.2-2.~~

~~Category~~ ~~Inspection Results~~

~~C-1~~ ~~Less than 5% of the total tubes inspected are degraded tubes, and none of the inspected tubes are defective.~~

~~C-2~~ ~~Between 5% and 10% of the total tubes inspected are degraded tubes, or one or more tubes, but not more than 1% of the total tubes inspected, are defective.~~

~~C-3~~ ~~More than 10% of the total tubes inspected are degraded tubes, or more than 1% of the inspected tubes are defective.~~

~~**NOTE:**~~ ~~For all inspections, previously degraded tubes must exhibit significant (>10%) added wall penetration to be included in the above percentage calculations.~~

~~3. Inspection Frequency~~

~~In-service inspection of steam generator tubes shall be performed at the following intervals:~~

~~In-service inspections may be performed during refueling outages, but shall be performed at intervals not to exceed 24 calendar months, except that the inspection interval may be extended to a maximum of 40 months if:~~

~~1. two consecutive inspections following service under AVT conditions, not including the pre-service inspection, yield results that fall into the C-1 category, or~~

~~2. two consecutive inspections demonstrate that previously documented degradation sites have not continued to deteriorate and no new degradation is found.~~

~~**NOTE:**~~ ~~A one-time inspection interval extension of a maximum of once per 40 months is allowed following the inspection performed during the spring 2003 inspection. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category.~~

~~If the result of a steam generator in-service inspection conducted in accordance with Table TS 4.2-2 falls into Category C-3, the inspection interval shall be reduced to 20 months. The 20 month interval shall apply until a subsequent inspection meets the conditions set forth in TS 4.2.b.3.a for extending the interval to 40 months.~~

~~Additional, unscheduled in-service inspections of each steam generator shall be performed using the criteria set forth in Table 4.2-2 for a "1st SAMPLE INSPECTION" during shutdowns consequent to:~~

~~1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of TS 3.1.d and TS 3.4.d, or~~

~~2. A seismic event having a magnitude greater than the Operating Basis Earthquake, or~~

~~3. A loss-of-coolant accident requiring actuation of engineered safeguards, where the Reactor Coolant System cooldown rate exceeded 100°F/hr, or~~

~~4. A main steam line or feedwater line break, where the Reactor Coolant System cooldown rate exceeded 100°F/hr.~~

~~If there is a significant change in steam generator chemistry control methodology, the steam generators shall be operated at power for three months while using the new treatment and shall then be inspected during the next outage of sufficient duration.~~

~~4. Plugging Limit Criteria~~

~~Any tube with tube wall degradation of 50% or more shall be plugged before returning the steam generator to service. If significant general tube thinning occurs, this criterion is reduced to 40% wall degradation.~~

~~5. Deleted~~

~~6. Deleted~~

~~7. Reports~~

~~Following each in-service inspection of steam generator tubes during which tubes are plugged, the number of tubes plugged shall be reported to the Commission within 30~~60~~ days.~~

~~The results of each steam generator tube in-service inspection shall be included in the Annual Operating Report for the reporting period that included completion of the inspection. The report shall include:~~

~~1. Number of tubes inspected and extent of inspection.~~

~~2. Location of each tube wall degradation and its percent of wall penetration.~~

~~3. Identification of tubes plugged.~~

—

~~If a steam generator tube inspection result falls into Category C-3, the Commission shall be promptly (within 4 hours) notified according to requirements of 10 CFR 50.72(b)(2~~3~~)(ii). A Licensee Event Report shall then be filed with the Commission as described by Specification 4.2.b.7.a and as set forth in 10 CFR 50.73(a)(2)(ii).~~

4.18 RCS Operational LEAKAGE

APPLICABILITY

Applies to the surveillance requirements for RCS operational LEAKAGE.

OBJECTIVE

To assure that the RCS operational LEAKAGE requirements are verified in a sufficient periodicity.

SPECIFICATION

Note 1: LEAKAGE surveillances are not required to be performed until 12 hours after establishment of steady state operation.

Note 2: TS 4.18.a is not applicable to primary to secondary LEAKAGE

a. Verify RCS operational LEAKAGE, except for primary to secondary LEAKAGE, is within limits by performance of RCS water inventory balance each 72 hours.

b. Verify primary to secondary LEAKAGE is < 150 gallons per day through any one SG each 72 hours.

4.19 Steam Generator Tube Integrity

APPLICABILITY

Applies to the surveillance requirements for Steam Generator Tube Integrity.

OBJECTIVE

To assure that the Steam Generator Tube Integrity requirements are verified in a sufficient periodicity.

SPECIFICATION

a. Verify SG tube integrity in accordance with the Steam Generator Program.

b. Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering INTERMEDIATE SHUTDOWN following a SG tube inspection.

TABLE TS 4.2-2

STEAM GENERATOR TUBE INSPECTION

TS Table 4.2-2 has been deleted

~~1ST SAMPLE INSPECTION~~			~~2ND SAMPLE INSPECTION~~		~~3RD SAMPLE INSPECTION~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of S Tubes per S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Plug defective tubes and inspect additional 2S tubes in this S.G. (2)~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Plug defective tubes and inspect additional 4S tubes in this S.G. (2)~~	~~C-1~~	~~None~~
					~~C-2~~	~~Plug defective tubes~~
					~~C-3~~	~~Perform action for C-3 result of first sample~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~	~~N/A~~	~~N/A~~
	~~C-3~~	~~Inspect all tubes in this S.G., (2) plug defective tubes and inspect 2S tubes in the other S.G. (2)~~ ~~Prompt notification of the Commission. (1)~~	~~The other S.G. is C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~Other S.G. is C-2~~	~~Perform action for C-2 result of second sample~~	~~N/A~~	~~N/A~~
			~~Other S.G. is C-3~~	~~Inspect all tubes in other S.G. and plug defective tubes. Prompt notification of the Commission. (1) (2)~~	~~N/A~~	~~N/A~~

~~S = 6%/n Where n is the number of steam generators inspected during an inspection.~~

~~Notes: 1. Refer to Specification 4.2.b.7.c~~

~~2. As allowed by TS 4.2.b.2.d, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those portions of the tubes where imperfections were previously found.~~

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b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

4. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into INTERMEDIATE SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.22, 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,

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

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induced indications.

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

f. Total number and percentage of tubes plugged to date.

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

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

6.22 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 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in TS 3.1.d, "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 integrity is maintained until the next SG inspection. An 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.

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 SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever 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.

ATTACHMENT 4

LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

PROPOSED TECHNICAL SPECIFICATION PAGES

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

Section	Title	Page

Section	Title	Page

LIST OF TABLES

t. LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank.
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified Leakage

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary Leakage

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

d. RCS Operational LEAKAGE

1. When the average RCS temperature is > 200°F, RCS operational leakage shall be limited to:

   A. No pressure boundary LEAKAGE,

   B. 1 gpm unidentified LEAKAGE,

   C. 10 gpm identified LEAKAGE, and

   D. 150 gallons per day primary to secondary LEAKAGE through any one SG.

2. If the limits contained in TS 3.1.d.1 for identified or unidentified LEAKAGE are exceeded, then reduce the LEAKAGE to within their limits within 4 hours.

3. If the limits contained in TS 3.1.d.1 for pressure boundary or primary to secondary LEAKAGE are exceeded, or the time limit contained in TS 3.1.d.2 is exceeded, then initiate action to:

   - Achieve HOT SHUTDOWN within 6 hours, and

   - Achieve COLD SHUTDOWN within an additional 30 hours.

4. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be in operation with one of the two systems sensitive to radioactivity. Either system may be out of operation for up to 12 hours provided at least one system is OPERABLE.

g. Steam Generator Tube Integrity

1. When the average reactor coolant system temperature is > 200°F the following shall be maintained:

   A. SG Tube integrity shall be maintained, and

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

Note: Separate entry condition is allowed for each SG tube.

2. If the requirements of TS 3.1.g.1.B can not be met, then:

   A. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

   B. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering INTERMEDIATE SHUTDOWN following the next refueling outage or SG tube inspection.

3. If the requirements of TS 3.1.g.2.A or TS 3.1.g.1.A can not be met, then initiate action:

   - Achieve HOT SHUTDOWN within 6 hours

   - Achieve COLD SHUTDOWN within an additional 30 hours.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

b. Deleted

4.18 RCS Operational LEAKAGE

APPLICABILITY

Applies to the surveillance requirements for RCS operational LEAKAGE.

OBJECTIVE

To assure that the RCS operational LEAKAGE requirements are verified in a sufficient periodicity.

SPECIFICATION

Note 1: LEAKAGE surveillances are not required to be performed until 12 hours after establishment of steady state operation.

Note 2: TS 4.18.a is not applicable to primary to secondary LEAKAGE

a. Verify RCS operational LEAKAGE, except for primary to secondary LEAKAGE, is within limits by performance of RCS water inventory balance each 72 hours.

b. Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG each 72 hours.

4.19 Steam Generator Tube Integrity

APPLICABILITY

Applies to the surveillance requirements for Steam Generator Tube Integrity.

OBJECTIVE

To assure that the Steam Generator Tube Integrity requirements are verified in a sufficient periodicity.

SPECIFICATION

a. Verify SG tube integrity in accordance with the Steam Generator Program.

b. Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering INTERMEDIATE SHUTDOWN following a SG tube inspection.

TABLE TS 4.2-2

STEAM GENERATOR TUBE INSPECTION

TS Table 4.2-2 has been deleted

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

   A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

   Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

   A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

      (1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

4. Steam Generator Tube Inspection Report

   A report shall be submitted within 180 days after the initial entry into INTERMEDIATE SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.22, 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,

   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 number and percentage of tubes plugged to date,

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

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

6.22 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 gpd per SG.

   3. The operational LEAKAGE performance criterion is specified in TS 3.1.d, "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 integrity is maintained until the next SG inspection. An 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.

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 SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever 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.

ATTACHMENT 5

LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

Marked Up Technical Specification Bases Pages

For Information Only
KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

Leakage of Reactor Coolant (TS 3.1.d)[16]

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 TS requirement is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This TS requirement specifies the types and amounts of LEAKAGE.

KPS USAR, GDC Criterion 16 – "Monitoring Reactor Coolant Pressure Boundary," [17], states that means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage. USAR section 6.5 describes the capabilities of the leakage monitoring indication 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 that is 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% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

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

APPLICABLE Safety Analysis

-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 that primary to secondary LEAKAGE from the steam generators (SGs) is 150 gallons per day per steam generator [18][19][20][21]. The TS requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting

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[16] USAR Sections 6.5, 11.2.3, 14.2.4
[17] Kewaunee Power Station Updated Safety Analysis Report (USAR), Section 1.8, Criteria 16.
[18] USAR Section 14.2.4, "Steam Generator Tube Rupture.
[19] USAR Section 14.1.8, Locked Rotor
[20] USAR Section 14.2.5, Main Steam Line Break
[21] Westinghouse Calculation CN-CRA-00-70, Rod Ejection Accident

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 radiological accident ~~F~~-analysis [22] for SGTR assumes the contaminated secondary fluid is released to the environment from the ruptured and the intact steam generators. The release from the ruptured SG occurs until 30 minutes after the reactor trip and the release from the intact SG occurs until 24 hours after the reactor trip when RHR is placed in service. The 150 gpd per SG primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is less limiting for site radiation releases. The safety analysis for the SLB accident assumes 150 gpd primary to secondary LEAKAGE through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50.67 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

APPLICABILITY

When the RCS average temperature is > 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

TS REQUIREMENT

TS 3.1.d.1

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 TS requirement could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

B. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this TS requirement could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

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[22] Westinghouse Calculation CN-CRA-99-36, Steam Generator Tube Rupture

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, then the yearly whole body dose resulting from this activity at the SITE BOUNDARY, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the SITE BOUNDARY would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

C. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this TS requirement could result in continued degradation of a component or system.

D. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day limit per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines [23]. The Steam Generator Program operational LEAKAGE performance criteria in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that resulted in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

[23] NEI 97-06, "Steam Generator Program Guidelines."

TS 3.1.d.2

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the TS requirement limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

TS 3.1.d.3

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to HOT SHUTDOWN~~MODE 3~~ within 6 hours and COLD SHUTDOWN~~MODE 5~~ within an additional 306 hours after achieving HOT SHUTDOWN. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN~~MODE 5~~, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

~~TS (TS 3.1.d.1)~~

~~Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:~~

~~If the reactor coolant activity is 91/$\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, then the yearly whole body dose resulting from this activity at the SITE BOUNDARY, using an annual average X/Q = 2.0 x 10^{-6} sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.~~

~~With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the SITE BOUNDARY would be 0.09 rem/yr as given above.~~

~~Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).~~

~~Twelve hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.~~

~~TS 3.1.d.2~~

~~Limiting the leakage through any single steam generator to 150 gpd ensures that tube integrity is maintained during a design basis main steam line break or loss-of-coolant accident. Remaining within this leakage rate provides reasonable assurance that no single tube flaw will sufficiently enlarge to create a steam generator tube rupture as a result of stresses caused by a Loss of Coolant Accident (LOCA) or a main steam line break accident within the time allowed for detection of the accident condition and resulting commencement of plant shutdown. This operational leakage rate is less than the condition assumed in design basis safety analyses and conforms to industry standards established by the Nuclear Energy Institute through its NEI 97-06, "Generic Steam Generator Program Guidelines."~~

~~TS 3.1.d.3~~

~~When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the plant operating staff and will be documented in writing and approved by either the Plant Manager or his designated alternate. Under these conditions, an allowable Reactor Coolant System leak rate of 10 gpm has been established. This explained leak rate of 10 gpm is within the capacity of one charging pump as well as being equal to the capacity of the Steam Generator Blowdown Treatment System.~~

~~TS 3.1.d.4~~

~~The provision pertaining to a non-isolable fault in a Reactor Coolant System component is not intended to cover steam generator tube leaks, valve bonnets, packings, instrument fittings, or similar primary system boundaries not indicative of major component exterior wall leakage.~~

TS 3.1.d.4~~5~~

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A~~.~~ and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

If leakage is to another closed system it will be detected by the area and process radiation monitors and/or inventory control.

In the event that the limits as provided in the COLR are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit as provided in the COLR. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits as provided in the COLR are not developed, then the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits as specified in the COLR. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

BASIS – Steam Generator Tube Integrity (TS 3.1.g)

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by TS 3.4. "Steam and Power Conversion" when the RCS average temperature is greater than 350 F," and TS 3.1.a.2, "Decay Heat Removal Capability," when the RCS temperature is less than or equal to 350 F.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.22, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.22, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.22. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident

conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines.

APPLICABLE SAFETY ANALYSIS

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 TS 3.1.d, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube.

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 from all SGs of 300 gallons per day. 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 TS 3.1.c, "Maximum Coolant 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 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).

TS Requirement

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

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 6.22, "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 TS.

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.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and Draft Regulatory Guide 1.121.

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 150 gallons per day per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in TS 3.1.d, "RCS Operational 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 the OPERATING, HOT STANDBY, HOT SHUTDOWN, or INTERMEDIATE SHUTDOWN MODES.

RCS conditions are far less challenging in the COLD SHUTDOWN or REFUELING SHUTDOWN MODES than during the OPERATING, HOT STANDBY, HOT SHUTDOWN, or INTERMEDIATE SHUTDOWN MODES. In the COLD SHUTDOWN or REFUELING SHUTDOWN MODES, 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.

TS 3.1.g.2

This TS 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 TS 4.19. 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 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, TS 3.1.g.3 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 TS 3.1.g.2.B 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 INTERMEDIATE SHUTDOWN 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.

TS 3.1.g.3

If the Required Actions and associated Completion Times ~~of~~ are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within an additional 30 hours after achieving HOT SHUTDOWN.

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.

BASIS

Kewaunee ~~Nuclear~~ Power ~~Plant~~ Station (KPS) design was not designed to Section XI of the ASME Code; therefore, 100% compliance may not be practically achievable. However, the design process did consider access for in-service inspection, and made modifications within design limitations to provide maximum access. To the extent practical, ~~NMC~~ Dominion Energy Kewaunee, Inc. performs inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components in accordance with Section XI of the ASME Code. If an inspection required by the Code is impractical, ~~NMC~~ Dominion Energy Kewaunee, Inc. requests Commission approval for deviation from the requirement.

The basis for surveillance testing of the Reactor Coolant System pressure isolation valves identified in Table TS 3.1-2 is contained within "Order for Modification of License" dated April 20, 1981.

Technical Specification 4.2.b (Deleted)

~~These Technical Specifications provide inspection and plugging requirements for Kewaunee Nuclear Power Plant~~KPS ~~steam generator tubes. Fulfilling these requirements assures that KNPP~~ KPS ~~steam generator tubes are inspected and maintained in a manner consistent with current NRC regulations and guidelines including the General Design Criteria of 10 CFR Part 50, Appendix A.~~

~~General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," require the reactor coolant pressure boundary to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Also, GDC 15, "Reactor Coolant System Design," requires the Reactor Coolant System and associated auxiliary, control, and protection systems to be designed with sufficient margin to ensure that design limits of the reactor coolant pressure boundary are not exceeded during normal operation, including during anticipated operational transients. Furthermore, GDC 32, "Inspection of Reactor Coolant System Pressure Boundary," requires components that are part of the reactor coolant pressure boundary to be designed to permit periodic inspection and testing of critical areas in order to assess their structural and leak tight integrity.~~

~~The NRC has developed guidance for steam generator tube inspection and maintenance including Regulatory Guides 1.83 and 1.121. Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," forms the basis for many of the requirements in this section and should be consulted before revising them. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines steam generator tube minimum wall thickness.~~

Technical Specification 4.2.b.1 (Deleted)

~~If the steam generators are performing in an adequately similar manner, it is appropriate to limit the inspection to one steam generator per inspection interval on an alternating basis. This offers economic savings as well as reduction of radiation exposure and outage duration.~~

Technical Specification 4.2.b.2 (Deleted)

~~Inspection of the steam generator tubes provides evaluation of their service condition. Operational experience has shown that certain types of steam generators are susceptible to generic degradation mechanisms. It has also revealed site-specific steam generator tube degradation mechanisms. The Kewaunee inspection program assesses both generic and site-specific tube degradations.~~

~~Kewaunee uses various eddy current (EC) testing methodologies to inspect steam generator tubes. EC technology has improved considerably since Kewaunee began commercial operation in 1974, and NMC Dominion Energy Kewaunee, Inc. is committed to use advanced EC methods and technology, as appropriate, to assure accurate assessment of steam generator tube service condition.~~

Technical Specification 4.2.b.3 (Deleted)

~~Kewaunee Nuclear Power Plant Station steam generator tube inspections are typically conducted during refueling outages. Criteria used to select tubes for inspection are based, in part, on tube service condition determined during previous inspections, and on operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tubes results in expansion of the current inspection as well as increased frequency of subsequent inspections. In this manner, steam generator tube surveillance remains consistent with tube service condition.~~

~~Several operational events or transients require consequent steam generator tube inspections. These inspections must be performed after occurrence of excessive primary-to-secondary leakage or after transients that impose large mechanical and thermal stresses on the tubes~~

Technical Specification 4.2.b.4 (Deleted)

~~Procedures, calculations, and analyses found in WCAP-15325,[1] combined with conservative allowances, such as general corrosion and measurement error, are the bases for the tube plugging criteria set forth in TS 4.2.b.4. Tubes that exceed the limits established by these criteria must be removed from service by plugging.~~

~~Steam generator tube plugging is a common method of preventing excessive primary-to-secondary steam generator tube leakage. This method is relatively uncomplicated and isolates a defective tube from the reactor coolant system by installing mechanical devices to block its hot and cold leg tubesheet openings.~~

Technical Specification 4.2.b.5 (Deleted)

Technical Specification 4.2.b.6 (Deleted)

Technical Specification 4.2.b.7 (Deleted)

~~Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2~~3~~)(ii) and 10 CFR 50.73(a)(2)(ii).~~

~~[1] WCAP 15325, "Regulatory Guide 1.121 Analysis for the Kewaunee Replacement Steam Generators."~~

BASIS – RCS Operational Leakage (TS 4.18)

TS 4.18.a

Verifying RCS LEAKAGE to be within the TS LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). This surveillance is modified by two notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in TS 3.1.d.4.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory.

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

TS 4.18.b

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day 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 ~~b~~not met, compliance with TS 3.1.g, "Steam Generator Tube Integrity,"" should be evaluated. The 150 gallons per day 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 surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The surveillance frequency of 72 hours 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 ~~f~~grab samples in accordance with the EPRI guidelines[1].

[1] EPRI, " Pressurized Water Reactor Primary to Secondary Leak Guidelines"

BASIS – Steam Generator Tube Integrity (TS 4.19)

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines, 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.

TS 4.19.a

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, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of TS 4.19.a. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines [1]. 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 6.22 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

TS 4.19.b

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.22 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). NEI 97-06, "Steam Generator Program Guidelines," provides guidance for

[1] EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

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 INTERMEDIATE SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

ATTACHMENT 6

LICENSE AMENDMENT REQUEST 218
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

Proposed Technical Specification Bases Pages

For Information Only

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

Leakage of Reactor Coolant (TS 3.1.d)[16]

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 TS requirement is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This TS requirement specifies the types and amounts of LEAKAGE.

KPS USAR, GDC Criterion 16 – "Monitoring Reactor Coolant Pressure Boundary," [17], states that means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage. USAR section 6.5 describes the capabilities of the leakage monitoring indication 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 that is 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% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

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

APPLICABLE Safety Analysis

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 that primary to secondary LEAKAGE from the steam generators (SGs) is 150 gallons per day per steam generator [18][19][20][21]. The TS requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting

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[16] USAR Sections 6.5, 11.2.3, 14.2.4
[17] Kewaunee Power Station Updated Safety Analysis Report (USAR), Section 1.8, Criteria 16.
[18] USAR Section 14.2.4, "Steam Generator Tube Rupture.
[19] USAR Section 14.1.8, Locked Rotor
[20] USAR Section 14.2.5, Main Steam Line Break
[21] Westinghouse Calculation CN-CRA-00-70, Rod Ejection Accident

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 radiological accident analysis [22] for SGTR assumes the contaminated secondary fluid is released to the environment from the ruptured and the intact steam generators. The release from the ruptured SG occurs until 30 minutes after the reactor trip and the release from the intact SG occurs until 24 hours after the reactor trip when RHR is placed in service. The 150 gpd per SG primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is less limiting for site radiation releases. The safety analysis for the SLB accident assumes 150 gpd primary to secondary LEAKAGE through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50.67 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

APPLICABILITY

When the RCS average temperature is > 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

TS REQUIREMENT

TS 3.1.d.1

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 TS requirement could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

B. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this TS requirement could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

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[22] Westinghouse Calculation CN-CRA-99-36, Steam Generator Tube Rupture

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, then the yearly whole body dose resulting from this activity at the SITE BOUNDARY, using an annual average X/Q = 2.0×10^{-6} sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the SITE BOUNDARY would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

C. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this TS requirement could result in continued degradation of a component or system.

D. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day limit per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines [23]. The Steam Generator Program operational LEAKAGE performance criteria in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that resulted in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

[23] NEI 97-06, "Steam Generator Program Guidelines."

TS 3.1.d.2

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the TS requirement limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

TS 3.1.d.3

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within an additional 30 hours after achieving HOT SHUTDOWN. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

TS 3.1.d.4

If leakage is to the containment, it may be identified by one or more of the following methods:

A. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are dependent upon the presence of corrosion product activity.

B. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to > 10 gpm.

C. Humidity detection provides a backup to A and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

   If leakage is to another closed system it will be detected by the area and process radiation monitors and/or inventory control.

In the event that the limits as provided in the COLR are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit as provided in the COLR. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits as provided in the COLR are not developed, then the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits as specified in the COLR. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

BASIS – Steam Generator Tube Integrity (TS 3.1.g)

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by TS 3.4. "Steam and Power Conversion" when the RCS average temperature is greater than 350 F," and TS 3.1.a.2, "Decay Heat Removal Capability," when the RCS temperature is less than or equal to 350 F.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.22, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.22, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.22. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident

conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines.

APPLICABLE SAFETY ANALYSIS

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 TS 3.1.d, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube.

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 from all SGs of 300 gallons per day. 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 TS 3.1.c, "Maximum Coolant 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 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).

TS Requirement

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

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 6.22, "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 TS.

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.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and Draft Regulatory Guide 1.121.

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 150 gallons per day per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in TS 3.1.d, "RCS Operational 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 the OPERATING, HOT STANDBY, HOT SHUTDOWN, or INTERMEDIATE SHUTDOWN MODES.

RCS conditions are far less challenging in the COLD SHUTDOWN or REFUELING SHUTDOWN MODES than during the OPERATING, HOT STANDBY, HOT SHUTDOWN, or INTERMEDIATE SHUTDOWN MODES. In the COLD SHUTDOWN or REFUELING SHUTDOWN MODES, 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.

TS 3.1.g.2

This TS 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 TS 4.19. 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 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, TS 3.1.g.3 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 TS 3.1.g.2.B 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 INTERMEDIATE SHUTDOWN 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.

TS 3.1.g.3

If the Required Actions and associated Completion Times are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within an additional 30 hours after achieving HOT SHUTDOWN.

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.

BASIS

Kewaunee Power Station (KPS) design was not designed to Section XI of the ASME Code; therefore, 100% compliance may not be practically achievable. However, the design process did consider access for in-service inspection, and made modifications within design limitations to provide maximum access. To the extent practical, Dominion Energy Kewaunee, Inc. performs inspection of ASME Code Class 1, Class 2, Class 3, and Class MC components in accordance with Section XI of the ASME Code. If an inspection required by the Code is impractical, Dominion Energy Kewaunee, Inc. requests Commission approval for deviation from the requirement.

The basis for surveillance testing of the Reactor Coolant System pressure isolation valves identified in Table TS 3.1-2 is contained within "Order for Modification of License" dated April 20, 1981.

Technical Specification 4.2.b (Deleted)

Technical Specification 4.2.b.1 (Deleted)

Technical Specification 4.2.b.2 (Deleted)

Technical Specification 4.2.b.3 (Deleted)

Technical Specification 4.2.b.4 (Deleted)

Technical Specification 4.2.b.5 (Deleted)

Technical Specification 4.2.b.6 (Deleted)

Technical Specification 4.2.b.7 (Deleted)

BASIS – RCS Operational Leakage (TS 4.18)

TS 4.18.a

Verifying RCS LEAKAGE to be within the TS LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). This surveillance is modified by two notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in TS 3.1.d.4.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory.

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

TS 4.18.b

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day 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 TS 3.1.g, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day 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 surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The surveillance frequency of 72 hours 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 samples in accordance with the EPRI guidelines[1].

[1] EPRI, " Pressurized Water Reactor Primary to Secondary Leak Guidelines"

BASIS – Steam Generator Tube Integrity (TS 4.19)

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines, 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.

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The Steam Generator Program defines the Frequency of TS 4.19.a. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines [1]. 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 6.22 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

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During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.22 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). NEI 97-06, "Steam Generator Program Guidelines." provides guidance for

[1] EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

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 INTERMEDIATE SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.