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Dominion®

May 31, 2006

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Serial No. 06-001
NLOS/PRW Rev. 0
Docket Nos. 50-336/423
License Nos. DPR-65
NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNITS 2 AND 3
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT
REGARDING STEAM GENERATOR TUBE INTEGRITY (CLIIP)

In accordance with the provisions of 10 CFR 50.90 Dominion Nuclear Connecticut, Inc. (DNC) is submitting a request for an amendment to the technical specifications (TS) for Millstone Power Station Units 2 and 3 (MPS2&3).

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). In addition, the definitions of the various types of LEAKAGE have been consolidated under one heading entitled "LEAKAGE," to align with NUREG 1431, Rev. 3, "Standard Technical Specifications for Westinghouse Plants," and NUREG 1432, Rev. 3, "Standard Technical Specifications for Combustion Engineering Plants."

Attachments 1 and 5 contain the description and justification for MPS2 and MPS3, respectively. Likewise, Attachments 2 and 6 contain the associated marked up technical specification pages. Attachments 3 and 7 contain the proposed amendment pages. Attachments 4 and 8 contain the marked up bases pages and are provided for information only. The changes to the affected TS bases pages will be incorporated in accordance with the TS Bases Control Program.

The Site Operations Review Committee has reviewed and concurred with the determinations.

DNC requests approval of the proposed license amendment by November 30, 2006, with the amendment to be implemented within 180 days. Approval of this request for MPS2 is contingent upon concurrent NRC approval of DNC's submittal for implementation of an alternate source term, which is being provided under separate cover within the next 30 days.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

Should you have any questions in regard to this submittal, please contact Mr. Paul R. Willoughby at (804) 273-3572.

Sincerely,

Eugene S. Grecheck
Vice President – Nuclear Support Services

Attachments:

1. Evaluation of Proposed License Amendment, MPS2
2. Marked Up Pages, MPS2
3. Proposed Amendment Pages, MPS2
4. Marked Up Bases Pages MPS2
5. Evaluation of Proposed License Amendment, MPS3
6. Marked Up Pages, MPS3
7. Proposed Amendment Pages, MPS3
8. Marked Up Bases Pages MPS3

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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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 Eugene S. Grecheck, who is Vice President - Nuclear Support Services of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 31st day of May, 2006.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

EVALUATION OF PROPOSED LICENSE AMENDMENT

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY EVALUATION OF PROPOSED LICENSE AMENDMENT

1.0 INTRODUCTION

The proposed license amendment revises the requirements in the Millstone Power Station Unit 2 (MPS2) 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). In addition, this proposed amendment groups the TS definitions of leakages under one definition of LEAKAGE, similar to that found in the standard technical specifications (NUREG 1432, Rev. 3.0).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

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

- Revise MPS2 TS INDEX
- Replace MPS2 TS 1.14, "IDENTIFIED LEAKAGE" with new MPS2 TS 1.14, "LEAKAGE" (includes ""CONTROLLED LEAKAGE," "IDENTIFIED LEAKAGE," "PRESSURE BOUNDARY LEAKAGE," "UNIDENTIFIED LEAKAGE")
- Delete MPS2 TS 1.15 "UNIDENTIFIED LEAKAGE"
- Delete MPS2 TS 1.16 "PRESSURE BOUNDARY LEAKAGE"
- Delete MPS2 TS 1.17 "CONTROLLED LEAKAGE"
- Revise MPS2 TS 3.4.5 and rename it, "Steam Generator (SG) Tube Integrity"
- Replace existing MPS2 SR 4.4.5.0 and 4.4.5.1 with new MPS2 TS SR 4.4.5.1 and 4.4.5.2
- Delete Table 4.4.5 and Table 4.4.6
- Rename MPS2 TS 3.4.6.2, "Reactor Coolant System Operational LEAKAGE"
- Revise MPS2 TS 3.4.6.2.c.
- Delete redundant wording from MPS2 TS 3.4.6.2.d.
- Revise MPS2 TS 3.4.6.2 ACTIONS a. and b.
- Revise MPS2 TS SR 4.4.6.2.1 and SR 4.4.6.2.2
- Delete MPS2 TS 6.9.1.5.b
- Insert new MPS2 TS 6.9.1.9, "Steam Generator Tube Inspection Report"

- Delete MPS2 TS 6.9.2 j.
- Add new MPS2 TS 6.26, "Steam Generator (SG) Program"
- Replace Bases of MPS2 B3/4.4.5, "Steam Generators" with Bases B3/4.4.5, "Steam Generator Tube Integrity"
- Replace Bases of MPS2 B 3/4.4.6.2, "Reactor Coolant System LEAKAGE with Bases B 3/4.4.6.2, "Reactor Coolant System Operational LEAKAGE."

Proposed revisions to the TS bases are included in this application for information only. 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 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.

5.0 TECHNICAL ANALYSIS

Dominion Nuclear Connecticut, Inc. (DNC) 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 staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. DNC has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to MPS2 and justify this amendment for the incorporation of the changes to the MPS2 TS.

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.

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.	Millstone Power Station Unit 2 (MPS2)
Steam Generator Model(s)	Babcock & Wilcox Replacement Steam Generators, 2-loop.
Effective Full Power Years (EFPY) of service for currently installed SGs	7.0 EFPY at last inspection in May 2005
Tubing Material	690TT
Number of tubes per SG	SG A – 8522. R57L156 hot leg tube sheet was plugged, cold leg was not drilled, tube was not installed. SG B – 8523.
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	None
Degradation mechanism(s) identified	•Outer diameter (OD) wear at fan bar intersections •OD wear due to transient foreign objects •Neither of these mechanisms meets the definition of an active mechanism.
Current primary-to-secondary leakage limits	Limit / TS / Admin. Control Per SG: 0.035 gpm* / 0.035 gpm* Total: n/a / n/a *at room temperature
Approved Alternate Tube Repair Criteria (ARC)	None
Approved SG Tube Repair Methods	Sleeving
Current performance criteria for accident leakage	0.035 gpm per SG leakage at room temperature

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

7.1 Incorporation of TSTF-449, Revision 4

Dominion Nuclear Connecticut, Inc. (DNC) has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. DNC has concluded that the proposed determination presented in the notice is applicable to Millstone Power Station Unit 2 (MPS2) and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91 (a).

7.2 Changes to Address Improved Technical Specifications Format

DNC is proposing minor variations and/or deviations from the Technical Specification (TS) changes described in TSTF-449, Revision 4, to provide consistent terminology and format within the MPS2 TS. For example, the improved standard TS (STS) wording for the definition for LEAKAGE, as modified by the CLIIP, is being incorporated into the MPS2 TS in this license amendment request. In addition, contrary to STS, MPS2 TS separate limiting conditions for operation (LCOs) and action statements from surveillance requirements (SRs) by placing them in different TS sections (3 and 4, respectively). MPS2 TS use specific definitions for each operating condition, e.g., POWER OPERATION, HOT SHUTDOWN, HOT STANDBY, REACTOR CRITICAL, COLD SHUTDOWN and REFUELING SHUTDOWN that, while not identical, are generally consistent with the reactor operating MODEs specified in the CLIIP. The minor variations and/or deviations from the specific wording/format provided in the proposed MPS2 TS do not change the meaning, intent or applicability of the CLIIP.

In addition, the leakage limit from any one steam generator will be 75 gallons per day, which is more conservative than the limit proposed by the CLIIP (i.e., 150 gallons per day). This more conservative limit is imposed to ensure the radiological limits imposed by 10 CFR Part 50.67 guidelines, and the radiological limits to control room personnel imposed by GDC-19, and other NRC approved licensing basis (e.g., alternate source term) are not exceeded. 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. DNC is submitting a proposed license amendment request to the NRC within the next 30 days under separate cover which would implement an alternate source term for MPS2. Implementation of the proposed alternate source term for MPS2 is required to support the revised steam generator leakage limits referenced in this letter.

A significant hazards consideration determination has been performed for the TS changes associated with terminology and format differences between the MPS2 TS and the STS to facilitate incorporation of the changes described in TSTF-449, Revision 4. DNC has concluded that 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 changes involve adding a new definition and rewording the existing TS to be consistent with NUREG-1432, Revision 3. In addition, the requested change for MPS2 incorporates a more conservative leakage limit of 75 gallons per day per steam generator as opposed to the CLIIP specified limit of 150 gallons per day per steam generator. These changes do not involve any physical plant modifications or changes in plant operation; consequently, no technical changes are being made to the existing TS. 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. However, without the separate license amendment action on the alternate source term for MPS2, the CLIIP cannot be implemented.

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

The proposed changes involve adding a new definition and rewording the existing TS to be consistent with NUREG-1432, Revision 3. These administrative changes do not involve 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. 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 changes involve adding a new definition and rewording the existing TS to be consistent with NUREG-1432, 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. Also, since these changes are administrative in nature, no margin of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

8.0 ENVIRONMENTAL EVALUATION

DNC has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. DNC has concluded that the NRC staff's findings presented in that evaluation are applicable to MPS2, and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. DNC is not proposing significant variations and/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, the TS changes proposed by the CLIIP would be implemented such that they are consistent with the existing MPS2 TS format requirements. These minor variations and/or deviations do not conflict with the applicability of the NRC's model safety evaluation to the proposed changes. As noted earlier, this proposed TS cannot be acted on until the alternate source term is dispositioned as it depends on that action to make this administrative change.

Serial No. 06-001
Docket No. 50-336

ATTACHMENT 2

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

MARKED-UP TECHNICAL SPECIFICATION PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn. INSERT 1.14

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. 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.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals.

DELETE

INSERT 1.14

LEAKAGE

1.14 LEAKAGE shall be:

1.14.1 CONTROLLED LEAKAGE

CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals, and

1.14.2 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or
b. 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
c. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

1.14.3 PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a RCS component body, pipe wall, or vessel wall, and

1.14.4 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all LEAKAGE which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

 INSERT 3.4.5

DELETE

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following Augmented Inservice Inspection Program. 40

4.4.5.1 Augmented Inservice Inspection Program

4.4.5.1.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-5.

4.4.5.1.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-6. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.1.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.1.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

INSERT 3.4.5

STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

-- NOTE --

Repair of defective tubes shall be limited to sleeving. Tubes with defective sleeves shall be plugged.

3.4.5 Steam Generator (SG) tube integrity shall be maintained.

AND

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

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

ACTIONS

-- NOTE --

Separate ACTION entry is allowed for each SG tube.

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a. With one or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program:

    1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and

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

b. With Required ACTION and associated Completion Time of ACTION a. not met or SG tube integrity not maintained:

    1. Be in HOT STANDBY within 6 hours, and

    2. Be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

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

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

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (>20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.1.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection. 31

c. The tubes selected as the second and third samples (if required by Table 4.4-6) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspection include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

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	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

DELETE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months ater the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-6 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.1.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-6 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

3. Degraded Tube or sleeve means a tube or sleeve containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall or sleeve thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be repaired because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal wall thickness for tubes or sleeves.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.1.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg or an inspection from the point of entry (hot leg or cold leg side) completely around the U-bend to the opposite tube end.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or sleeve all tubes exceeding the plugging limit and plug all defecting sleeves) required by Table 4.4-6.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged and sleeved in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged or sleeved.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported pursuant to 10 CFR 50.72. A Special Report pursuant to Specification 6.9.2 shall be submitted prior to resumption of plant operation and shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

DELETE

TABLE 4.4-5

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Two
First Inservice Inspection	One
Second & Subsequent Inservice Inspections	One[1]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

DELETE

TABLE 4.4-6
STEAM GENERATOR TUBE INSPECTION

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	Repair defective tubes and inspect additional 25 tubes in this S.G.*	C-1	None	N/A	N/A
			C-2	Repair defective tubes and inspect additional 45 tubes in this S.G.*	C-1	None
					C-2	Repair 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	Inspection all tubes in this S.G., repair defective tubes and inspect 25 tubes in each other S.G.* Prompt notification to NRC pursuant to 10 CFR 50.72	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform ACTION for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and repair defective tubes.* Prompt notification to NRC pursuant to 10 CFR 50.72	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

* Repair of defective tubes shall be limited to plugging with the exception of those tubes which may be sleeved. Tubes with defective sleeves shall be plugged.

DELETE

OPERATIONAL

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

Operational

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

75 GPD

c. ~~0.035 GPM~~ primary-to-secondary leakage through any one steam generator, and

INSERT 3.4.6.2

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

DELETE

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE shall be demonstrated to be within limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

INSERT 3.4.6.2

ACTION:

a. With any RCS operational LEAKAGE not within limits for reasons other than PRESSURE BOUNDARY LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within limits within 4 hours.

b. With ACTION and associated Completion Time of ACTION a. not met, or PRESSURE BOUNDARY LEAKAGE exists, or primary to secondary LEAKAGE not within limits, be in HOT STANDBY within 6 hours and be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 ---------------------------------- NOTES ----------------------------------

1. Not required to be performed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance at least once per 72 hours.

4.4.6.2.2 ---------------------------------- NOTE ----------------------------------

Not required to be performed until 12 hours after establishment of steady state operation. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4.

Verify primary to secondary LEAKAGE is $\leq$ 75 gallons per day through any one SG at least once per 72 hours.

ANNUAL REPORTS[1]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted in accordance with 10 CFR 50.4

6.9.1.5a.DELETED

DELETE

6.9.1.5b The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.1.5.b). The report covering the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.5c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit. The report covering the previous calendar year shall be submitted prior to March 1 of each year.

1 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

CORE OPERATING LIMITS REPORT (CONT.)

8. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company.

9. XN-NF-82-06(P)(A), and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company.

10. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

11. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company.

12. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

13. EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.

14. EMF-2130(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.

15. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

d. d.The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

INSERT 6.9.1.9

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Deleted

INSERT 6.9.1.9

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.9 A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.26, 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 or repaired during the inspection outage for *each active degradation mechanism,*
f. Total number and percentage of tubes plugged or repaired to date,
g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and
h. The effective plugging percentage for all plugging and tube repairs in each SG.
i. Repair method utilized and the number of tubes repaired by each repair method.

SPECIAL REPORTS (CONT.)

b. Deleted

c. Deleted

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Deleted

f. Deleted

g. RCS Overpressure Mitigation, Specification 3.4.9.3.

h. Deleted

i. Tendon Surveillance Report, Specification 6.25

~~j. Steam Generator Tube Inspection, Specification 4.4.5.1.5.~~ DELETE

k. Accident Monitoring Instrumentation, Specification 3.3.3.8.

l. Radiation Monitoring Instrumentation, Specification 3.3.3.1.

m. Deleted

6.10 Deleted.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.24 DIESEL FUEL OIL TEST PROGRAM

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

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

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

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

   3. Water and sediment ≤ 0.05%.

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

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276-78, Method A.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Diesel Fuel Oil Test Program test frequencies.

6.25 PRE-STRESSED CONCRETE CONTAINMENT TENDON SURVEILLANCE PROGRAM

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989.

The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken. This Tendon Surveillance Report is an administrative requirement listed in Technical Specifications 6.9.2, "Special Reports."

INSERT 6.26

INSERT 6.26

6.26 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 or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged or repaired to confirm that the performance criteria are being met.

b. Provision for 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. Design basis leakage is not to exceed 150 gpd for any one SG and 300 gpd total.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System 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 or repaired.

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.

f. Provisions for SG tube repair methods: Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

    1. Sleeving

ATTACHMENT 3

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

PROPOSED AMENDMENT PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

1.14.1 CONTROLLED LEAKAGE

CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals, and

1.14.2 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. 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

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

1.14.3 PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a RCS component body, pipe wall, or vessel wall, and

1.14.4 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all LEAKAGE which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

NOTE

Repair of defective tubes shall be limited to sleeving. Tubes with defective sleeves shall be plugged.

3.4.5 Steam Generator (SG) tube integrity shall be maintained.

AND

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

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

ACTION:

NOTE

Separate ACTION entry is allowed for each SG tube.

---

a. With one or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program:

   1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and

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

b. With Required ACTION and associated Completion Time of ACTION a. not met or SG tube integrity not maintained:

   1. Be in HOT STANDBY within 6 hours, and

   2. Be in COLD SHUTDOWN within 36 hours.

STEAM GENERATOR TUBE INTEGRITY

SURVEILLANCE REQUIREMENTS

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

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

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REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System Operational LEAKAGE shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 75 GPD primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any RCS operational LEAKAGE not within limits for reasons other than PRESSURE BOUNDARY LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within limits within 4 hours.

b. With ACTION and associated Completion Time of ACTION a. not met, or PRESSURE BOUNDARY LEAKAGE exists, or primary to secondary LEAKAGE not within limits, be in HOT STANDBY within 6 hours and be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1

- - - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - - -

1 Not required to be performed until 12 hours after establishment of steady state operation.

2 Not applicable to primary to secondary LEAKAGE.

- -

Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance at least once per 72 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.2

- - - - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - - -

Not required to be performed until 12 hours after establishment of steady state operation. The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4.

- -

Verify primary to secondary LEAKAGE is $\leq$ 75 gallons per day through any one SG at least once per 72 hours.

ANNUAL REPORTS[1]

6.9.1.4

Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted in accordance with 10 CFR 50.4

6.9.1.5a.Deleted

6.9.1.5b Deleted

6.9.1.5c.

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit. The report covering the previous calendar year shall be submitted prior to March 1 of each year.

1 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

CORE OPERATING LIMITS REPORT (CONT.)

8. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company.

9. XN-NF-82-06(P)(A), and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company.

10. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

11. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company.

12. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

13. EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.

14. EMF-2130(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.

15. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.9 A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.26, 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 or repaired during the inspection outage for each active degradation mechanism,

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

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

h. The effective plugging percentage for all plugging and tube repairs in each SG.

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

The following references have been deleted a through c, e, f, h, j, m.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

g. RCS Overpressure Mitigation, Specification 3.4.9.3.

i. Tendon Surveillance Report, Specification 6.25

6.9.2 (Continued)

k. Accident Monitoring Instrumentation, Specification 3.3.3.8.

l. Radiation Monitoring Instrumentation, Specification 3.3.3.1.

6.10 DELETED.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.26 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 or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged or repaired to confirm that the performance criteria are being met.

b. Provisions for 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. Design basis leakage is not to exceed 150 gpd for any one SG and 300 gpd total.

    3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational LEAKAGE."

6.26 STEAM GENERATOR (SG) PROGRAM (Continued)

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 or repaired.

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.

f. Provisions for SG tube repair methods: Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

   1. Sleeving

ATTACHMENT 4

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

MARKED UP BASES PAGES (INFORMATION ONLY)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

stuck open PORV at a time that the block valve is inoperable. This may be accomplished by various methods. These methods include, but are not limited to, placing the NORMAL/ISOLATE switch at the associated Bottle Up Panel in the "ISOLATE" position or pulling the control power fuses for the associated PORV control circuit.

Although the block valve may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used to perform its function. Block valve inoperability may be due to seat leakage, instrumentation problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. This condition is only intended to permit operation of the plant for a limited period of time. The block valve should normally be available to allow PORV operation for automatic mitigation of overpressure events. The block valves must be returned to OPERABLE status prior to entering MODE 3 after a refueling outage.

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the completion time of 1 hour or isolate the flow path by closing and removing the power to the associated block valve and cooldown the RCS to MODE 4.

3/4.4.4 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the reactor coolant system during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which occurs if the heaters are energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish and maintain natural circulation.

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The requirement for two groups of pressurizer heaters, each having a capacity of 130 kW, is met by verifying the capacity of the pressurizer proportional heater groups 1 and 2. Since the pressurizer proportional heater groups 1 and 2 are supplied from the emergency 480V electrical buses, there is reasonable assurance that these heaters can be energized during a loss of offsite power to maintain natural circulation at HOT STANDBY.

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3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is

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3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LCO

The LCO requires that steam generator (SG) tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be repaired or plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, or repaired, 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 6.26, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

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

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 (Reference 4) and Draft Regulatory Guide 1.121 (Reference 5).

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 GPD per SG. 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.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.6.2, "Reactor Coolant System Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 75 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 during MODES 1, 2, 3, and 4.

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

ACTIONS

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

<u>a.1 and a.2</u>

ACTION a. applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by TS 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair 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 or repaired 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, ACTION b. applies.

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

If the evaluation determines that the affected tube(s) have tube integrity, ACTION a.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tube(s). However, the affected tube(s) must be plugged or repaired prior to entering HOT 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.

b.1 and b.2

If the ACTIONS and associated Completion Times of ACTION a. are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours.

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

SURVEILLANCE REQUIREMENTS

TS 4.4.5.1

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

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

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws 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 4.4.5.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.26 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

<u>TS 4.4.5.2</u>

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The tube repair criteria delineated in Specification 6.26 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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

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

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 LCO 3.4.1.1, "STARTUP and POWER OPERATION," LCO 3.4.1.2, "HOT STANDBY," LCO 3.4.1.3, "HOT SHUTDOWN," and LCO 3.4.1.4, "COLD SHUTDOWN, Loops Filled."

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 6.26, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.26, SG 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 6.26. 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 (Reference 1).

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting offsite dose 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 two times the operational LEAKAGE rate limits in LCO 3.4.6.2, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves or atmospheric dump valves.

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 any one SG of 150 gpd or from all SGs of 300 gpd as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "RCS Specific Activity" limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 50.67 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, " Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

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evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking.

The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.035 GPM, per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.035 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Sleeving repair will be limited to those steam generator tubes with a defect between the tube sheet and the first eggcrate support. Tubes containing sleeves with imperfections exceeding the plugging limit will be plugged. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR 50.72. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." INSERT B 3/4.4.6.2-OL

3/4.4.6.2 REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The steam generator tube leakage limit of 0.035 GPM per steam generator ensures that the dosage contribution from the tube leakage will be less than the limits of General Design Criteria 19 of 10CFR50 Appendix A in the event of either a steam generator tube rupture or steam line break. The 0.035 GPM limit is consistent with the assumptions used in the analysis of these accidents.

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PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

The IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE limits listed in LCO 3.4.6.2 only apply to the reactor coolant system pressure boundary within the containment.

In accordance with 10 CFR 50.2 "Definitions" the RCS Pressure Boundary means all those pressure-containing components such as pressure vessels, piping, pumps and valves which are (1) Part of the Reactor Coolant System, or (2) Connected to the Reactor Coolant System, up to and including any and all of the following: (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment, (ii) The second of two valves normally closed in system piping which does not penetrate primary reactor containment, or (iii) The reactor coolant safety and relief valves.

The definitions for IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE are provided in the Technical Specifications definitions section, definitions 1.14 and 1.15 respectively.

Leakage outside of the second isolation valve for containment which is included in the RCS Leak Rate Calculation is not considered RCS leakage and can be subtracted from RCS UNIDENTIFIED LEAKAGE.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into the containment area is necessary. Quickly separating 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. LCO 3.4.6.2 deals with protection of the reactor coolant pressure boundary from degradation and the core from inadequate cooling, in addition accident analysis radiation release assumptions from being exceeded.

INSERT B 3/4.4.6.2-OL

LCO RCS operational LEAKAGE shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary (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 LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Primary to Secondary LEAKAGE through Any One Steam Generator:

The limit of 75 gallons per day per Steam Generator (SG) is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 4) and the Accident Analyses described in the FSAR (Reference 3). The Steam Generator Program operational LEAKAGE performance criterion 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 result 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. The main steam line break (MSLB) accident analysis assumes a primary to secondary LEAKAGE of 150 gallons per day per SG. To account for the increased primary to secondary delta-P due to blowdown of the SG secondary side during the accident, and the resultant increase in primary to secondary LEAKAGE, operational primary to secondary LEAKAGE limit is set at 75 gallons per day per SG.

d. 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 LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

The IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE limits listed in LCO 3.4.6.2 only apply to the reactor coolant system pressure boundary within the containment. Leakage outside of the second isolation valve for containment, which is included in the RCS Leak Rate Calculation, is not considered RCS LEAKAGE and can be subtracted from RCS UNIDENTIFIED LEAKAGE. The definitions for IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE are provided in the technical specifications definitions section, definition 1.14.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

a. UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE in excess of the LCO 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.

b. If any PRESSURE BOUNDARY LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, or if UNIDENTIFIED or IDENTIFIED 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 STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. 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 reactor coolant pressure boundary are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1

Verifying RCS LEAKAGE to be within the 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. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal leakoff flows). The 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 water 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 leakoff 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. These leakage detection systems are specified in LCO 3.4.6.1, “Leakage Detection Systems.”

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 75 gallons per day cannot be measured accurately by an RCS water inventory balance.

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

4.4.6.2.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 75 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 LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 75 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 leakoff 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 sampling in accordance with the EPRI guidelines (Reference 5).

<u>BACKGROUND</u>

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

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

10 CFR 50, Appendix A, GDC 30 (Reference 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Reference 2) describes acceptable methods for selecting leakage detection systems.

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

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% 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 LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

<u>APPLICABLE SAFETY ANALYSES – OPERATIONAL LEAKAGE</u>

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 any one steam generator (SG) of 150 gpd or from all SGs of 300 gpd as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 75 gallons per day is significantly less than the conditions assumed in the safety analysis. The limit of 75 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 4) with conservatism added to account for the MPS2 radiological analysis. The Steam Generator Program operational LEAKAGE performance criterion 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 result in tube leakage. The operational leakage rate criterion, as modified to account for the MPS2 radiological analysis, in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

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

The FSAR (Reference 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves or atmospheric dump valves.

The MSLB is the more limiting accident for MPS2 control room dose. The safety analysis for the MSLB accident assumes 150 gpd primary to secondary LEAKAGE is through the affected generator and 150 gpd from the intact SG as an initial condition. The dose consequences resulting from the MSLB 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).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 14
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

ATTACHMENT 5

<u>APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY</u>

<u>EVALUATION OF PROPOSED LICENSE AMENDMENT</u>

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY EVALUATION OF PROPOSED LICENSE AMENDMENT

1. INTRODUCTION

The proposed license amendment revises the requirements in the Millstone Power Station Unit 3 (MPS3) 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). In addition, this proposed amendment groups the TS definitions of leakages under one definition of LEAKAGE, similar to that found in the standard technical specifications (NUREG 1431, Rev. 3.0).

2. DESCRIPTION OF PROPOSED AMENDMENT

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

- Revise MPS3 TS INDEX
- Replace MPS3 TS 1.16, "IDENTIFIED LEAKAGE" with new MPS3 TS 1.16, "LEAKAGE" (includes ""CONTROLLED LEAKAGE," "IDENTIFIED LEAKAGE," "PRESSURE BOUNDARY LEAKAGE," "UNIDENTIFIED LEAKAGE")
- Delete MPS3 TS 1.8 "CONTROLLED LEAKAGE"
- Delete MPS3 TS 1.22 "PRESSURE BOUNDARY LEAKAGE"
- Delete MPS3 TS 1.37 "UNIDENTIFIED LEAKAGE"
- Replace MPS3 TS 3.4.5 in its entirety with a new TS 3.4.5 and rename it, "Steam Generator Tube Integrity"
- Replace existing MPS3 TS SR 4.4.5.0, 4.4.5.1, 4.4.5.2, 4.4.5.3, 4.4.5.4, 4.4.5.5, TABLE 4.4-1, and TABLE 4.4-2 with new TS SR 4.4.5.1 and 4.4.5.2
- Revise wording of MPS3 TS 3.4.6.2
- Revise MPS3 TS LCO 3.4.6.2 c., Leakage through any one steam generator
- Remove redundant wording from MPS3 TS LCO 3.4.6.2.d
- Revise MPS3 TS 3.4.6.2, ACTIONs a., b., and c.
- Revise wording of MPS3 TS SR 4.4.6.2.1
- Replace existing MPS3 TS SR 4.4.6.2.1.d with new MPS3 TS SR 4.4.6.2.1.d

- Add new MPS3 TS SR 4.4.6.2.1.e
- Renumber existing MPS3 TS SR 4.4.6.2.1.e to MPS3 TS SR 4.4.6.2.1.f
- Add new MPS3 TS 6.8.4.g., "Steam Generator (SG) Program"
- Add new MPS3 TS 6.9.1.7, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS bases are also included for information only in Attachment 8 of 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 MPS3 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 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.

5.0 TECHNICAL ANALYSIS

Dominion Nuclear Connecticut, Inc. (DNC) 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 staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. DNC has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to MPS3 and justify this amendment for the incorporation of the changes to the MPS3 TS.

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.

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.	Millstone Power Station Unit 3 (MPS3)
Steam Generator Model	Westinghouse Model F, 4-Loop
Effective Full Power Years (EFPY) of service for currently installed SGs	13.9 EFPY at last inspection in October 2005
Tubing Material	600TT
Number of tubes per SG	5626 tubes/SG
Number and percentage of tubes plugged in each SG	SG A – 36 (0.64%) SG B – 4 (0.07%) SG C – 11 (0.20%) SG D – 78 (1.39%)
Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	•Outer diameter (OD) wear at anti-vibration bar intersections •OD wear due to transient foreign objects Neither of these mechanisms meet the definition of an active mechanism
Current primary-to-secondary leakage limits	Limit / TS / Admin. Control Per SG: 500 gpd* ** / 150 gpd** Increase 15 gpd** in 30 min >75 gpd** for 1 hr Total: 1 gpm* *for SG not isolated from RCS **at room temperature
Approved Alternate Tube Repair Criteria (ARC)	None
Approved SG Tube Repair Methods	None
Current performance criteria for accident leakage	1 gpm total SG leakage at room temperature

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

7.1 Incorporation of TSTF-449, Revision 4

Dominion Nuclear Connecticut, Inc. (DNC) has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. DNC has concluded that the proposed determination presented in the notice is applicable to Millstone Power Station Unit 3 (MPS3) and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91 (a).

7.2 Changes to Address Improved Technical Specifications Format

DNC is proposing minor variations and/or deviations from the technical specification (TS) changes described in TSTF-449, Revision 4, to provide consistent terminology and format within MPS3 TS. For example, the improved Standard TS (STS) wording for the definition for LEAKAGE, as modified by the CLIIP, is being incorporated into MPS3 TS in this license amendment request. In addition, contrary to STS, MPS3 TS separate Limiting Conditions for Operation (LCOs) and Action Statements from Surveillance Requirements (SRs) by placing them in different TS sections (3 and 4, respectively). MPS3 TS use specific definitions for each operating condition, e.g., POWER OPERATION, STARTUP, HOT STANDBY, HOT SHUTDOWN, COLD SHUTDOWN and REFUELING that, while not identical, are generally consistent with the reactor operating MODEs specified in the CLIIP. The minor variations and/or deviations from the specific wording/format provided in the proposed MPS3 TS do not change the meaning, intent or applicability of the CLIIP.

A significant hazards consideration determination has been performed for the TS changes associated with terminology and format differences between the MPS3 TS and the STS to facilitate incorporation of the changes described in TSTF-449, Revision 4. DNC has concluded that 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 changes involve rewording the existing technical specifications (TS) to be consistent with NUREG-1431, Revision 3. These changes do not involve any physical plant modifications or changes in plant operation; consequently, no technical changes are being made to the existing TS. 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 changes involve rewording the existing TS to be consistent with NUREG-1431, Revision 3. These administrative changes do not involve 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. 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 changes involve rewording the existing TS 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. Also, 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

DNC has reviewed the environmental evaluation included in the model Safety Evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIIP. DNC has concluded that the staff's findings presented in that evaluation are applicable to MPS3, and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. DNC is not proposing significant variations and/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, the TS changes proposed by the CLIIP would be implemented such that they are consistent with the existing MPS3 TS format requirements. Consequently, these minor variations and/or deviations do not conflict with the applicability of the NRC's model safety evaluation to the proposed changes.

ATTACHMENT 6

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

MARKED-UP TECHNICAL SPECIFICATION PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

DEFINITIONS

I3

DEFINITIONS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation valve system*, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

~~CONTROLLED LEAKAGE~~

~~1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~

DELETE

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

* In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\overline{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

1.12 DELETED

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

1.14 DELETED

FREQUENCY NOTATION

INSERT 1.16

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. 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

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

DELETE

MASTER RELAY TEST

1.17 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include continuity check of each associated slave relay.

INSERT 1.16

LEAKAGE

1.16 LEAKAGE shall be:

1.16.1 CONTROLLED LEAKAGE

CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals, and

1.16.2 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or
b. 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
c. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE), or
d. LEAKAGE through a RCS pressure isolation valve;

1.16.3 PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a RCS component body, pipe wall, or vessel wall, and

1.16.4 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all LEAKAGE which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

MEMBER(S) OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

The term "REAL MEMBER OF THE PUBLIC" means an individual who is exposed to existing dose pathways at one particular location.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

DELETE

PURGE - PURGING

1.23 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.32 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.37 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

DELETE

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

DELETE

INSERT 3.4.5

3.4.5 Each steam generator associated with an operating RCS loop shall be OPERABLE.

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

ACTION:

With one or more steam generators associated with an operating RCS loop inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

INSERT 3.4.5

LIMITING CONDITION FOR OPERATION

3.4.5 Steam Generator (SG) tube integrity shall be maintained.

AND

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

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

ACTIONS

-- NOTE --
Separate ACTION entry is allowed for each SG tube.
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a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program:

   1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and

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

b. With Required ACTION and associated Completion Time of ACTION a. not met or SG tube integrity not maintained:

   1. Be in HOT STANDBY within 6 hours, and

   2. Be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

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

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

DELETE

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

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	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

DELETE

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months* after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1) Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

2) A seismic occurrence greater than the Operating Basis Earthquake, or

3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4) A main steam line or feedwater line break.

* Except the surveillance related to Steam Generator Inspection, due no later than September 24, 1998, may be deferred until the next refueling outage or no later than July 1, 1999, whichever is earlier.

DELETE

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; or an inspection from the point of entry (Hot Leg or Cold Leg Side) completely around the U-bend to the opposite tube end.

DELETE

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspectors indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspection shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

DELETE

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

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.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	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., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N.A	N.A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform ACTION for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A

$S = 3\frac{N}{n}$ %Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

DELETE

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

operational LEAKAGE

3.4.6.2 Reactor Coolant System ~~leakage~~ shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

DELETE

b. 1 gpm UNIDENTIFIED LEAKAGE,

150 primary to secondary LEAKAGE

c. ~~1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500~~ gallons per day through any one steam generator ~~not isolated from the Reactor Coolant System,~~

DELETE

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2250 ± 20 psia, and

LEAKAGE

f.* 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

primary to secondary LEAKAGE not within limits or

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

RCS operational LEAKAGE not within

DELETE

b. With any ~~Reactor Coolant System leakage greater than any one of the above~~ limits, ~~excluding~~ PRESSURE BOUNDARY LEAKAGE, ~~and~~ leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

other than

LEAKAGE

or primary to secondary LEAKAGE

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LEAKAGE

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* This requirement does not apply to Pressure Isolation Valves in the Residual Heat Removal flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

operational LEAKAGE

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Deleted

b. Deleted

Insert 4.4.6.2.1.d

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 ± 20 psia at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

DELETE

~~d. Performance of a Reactor Coolant System water inventory balance within 12 hours of achieving steady state operation, and at least once per 72 hours thereafter during steady state operation. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4; and~~

Insert 4.4.6.2.1.e

f. ~~e.~~ Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2[(1)(2)] Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

LEAKAGE

a. At least once per 24 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

---

(1) The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

INSERT 4.4.6.2.1.d

d. -- NOTES --

1. Not required to be performed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

Performance of a Reactor Coolant System water inventory balance at least once per 72 hours. The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4;

INSERT 4.4.6.2.1.e

e. -- NOTE --

1. Not required to be performed until 12 hours after establishment of steady state operation.

Verification that primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one Steam Generator at least once per 72 hours, and;

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

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

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38.57 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq\ 0.042\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is L 0.05 La when tested at ≥ Pa.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

INSERT 6.8.4.g

* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

INSERT 6.8.4.g

6.8.4.g. 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. Provisions for 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 a 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 1 gpm from all SGs not isolated from the RCS and 500 gpd from any one SG not isolated from the RCS.

3. The operational LEAKAGE performance criterion is specified in RCS LCO 3.4.6.2, "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 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 120, 90, 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 48 effective full power months or two 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.

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

INSERT 6.9.1.7

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator Region I, and one copy to the NRC Resident Inspector, within the time period specified for each report.

6.10 Deleted.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent; that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

INSERT 6.9.1.7

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.8.4.g, 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, and
h. The effective plugging percentage for all plugging in each SG.

Serial No. 06-001
Docket No. 50-423

ATTACHMENT 7

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

PROPOSED AMENDMENT PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

DEFINITIONS

INDEX

DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system*, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

1.8 DELETED

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

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* In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\overline{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

1.12 DELETED

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

1.14 DELETED

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

LEAKAGE

1.16 LEAKAGE shall be:

1.16.1 CONTROLLED LEAKAGE

CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals, and

1.16.2 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. 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

c. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE), or

d. LEAKAGE through a RCS pressure isolation valve;

1.16.3 PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in a RCS component body, pipe wall, or vessel wall, and

1.16.4 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all LEAKAGE which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

MASTER RELAY TEST

1.17 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

The term "REAL MEMBER OF THE PUBLIC" means an individual who is exposed to existing dose pathways at one particular location.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

1.22 DELETED

PURGE - PURGING

1.23 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.32 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.35 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

1.37 DELETED

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam Generator (SG) tube integrity shall be maintained.

AND

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

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

ACTION:

------------------------------ NOTE ------------------------------

1. Separate ACTION entry is allowed for each SG tube.

----

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program:

   1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and

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

b. With Required ACTION and associated Completion Time of ACTION a. not met or SG tube integrity not maintained:

   1. Be in HOT STANDBY within 6 hours, and

   2. Be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

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

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

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational LEAKAGE shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary to secondary LEAKAGE through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2250 ± 20 psia, and

f.* 0.5 gpm LEAKAGE per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

a. With primary to secondary LEAKAGE not within limits or any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any RCS operational LEAKAGE not within limits, other than PRESSURE BOUNDARY LEAKAGE, LEAKAGE from Reactor Coolant System Pressure Isolation Valves or primary to secondary LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve LEAKAGE greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

* This requirement does not apply to Pressure Isolation Valves in the Residual Heat Removal flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational LEAKAGE shall be demonstrated to be within each of the above limits by:

a. Deleted

b. Deleted

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 ± 20 psia at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d.

------------------- NOTE -------------------

1. Not required to be performed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

Performance of a Reactor Coolant System water inventory balance at least once per 72 hours. The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4;

e.

------------------- NOTE -------------------

1. Not required to be performed until 12 hours after establishment of steady state operation.

Verification that primary to secondary LEAKAGE is ≤ 150 gallons per day through any one Steam Generator at least once per 72 hours, and;

f. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2[(1)(2)] Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying LEAKAGE to be within its limit:

---

(1) The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 24 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

g. 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 a 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. Provisions for 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 a 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.

PROCEDURES AND PROGRAMS (Continued)

Leakage is not to exceed 1 gpm from all SGs not isolated from the RCS and 500 gpd from any one SG not isolated from the RCS.

3. The operational LEAKAGE performance criterion is specified in RCS LCO 3.4.6.2, "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 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 120, 90, 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 48 effective full power months or two 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

PROCEDURES AND PROGRAMS (Continued)

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.

6.8.5 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.6 All procedures and procedure changes required for the Radiological Environmental Monitoring Program (REMP) of Specification 6.8.5 above shall be reviewed by an individual (other than the author) from the organization responsible for the REMP and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the organization responsible for the REMP, within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector, unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.8.4.g, 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, and

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator Region I, and one copy to the NRC Resident Inspector, within the time period specified for each report.

6.10 Deleted.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent; that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

6.12 HIGH RADIATION AREA (cont.)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designees, and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of

Serial No. 06-001
Docket No. 50-423

ATTACHMENT 8

APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

MARKED UP BASES PAGES (INFORMATION ONLY)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The purpose of Specification 3.4.1.1 is to require adequate forced flow rate for core heat removal in MODES 1 and 2 during all normal operations and anticipated transients. Flow is represented by the number of reactor coolant pumps in operation for removal of heat by the steam generators. To meet safety analysis acceptance criteria for DNB, four reactor coolant pumps are required at rated power. An OPERABLE reactor coolant loop consists of an OPERABLE reactor coolant pump in operation providing forced flow for heat transport and an OPERABLE steam generator in accordance with Specification 3.4.5. With less than the required reactor coolant loops in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, three reactor coolant loops, and in MODE 4, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, in MODE 3 a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., the Control Rod Drive System is not capable of rod withdrawal.

In MODE 4, if a bank withdrawal accident can be prevented, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (any combination of RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two RHR loops or at least one RHR loop and two steam generators be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

In MODE 5, during a planned heatup to MODE 4 with all RHR loops removed from operation, an RCS loop, OPERABLE and in operation, meets the requirements of an OPERABLE and operating RHR loop to circulate reactor coolant. During the heatup there is no requirement for heat removal capability so the OPERABLE and operating RCS loop meets all of the required functions for the heatup condition. Since failure of the RCS loop, which is OPERABLE and operating, could also cause the associated steam generator to be inoperable, the associated steam generator cannot be used as one of the steam generators used to meet the requirement of LCO 3.4.1.4.1.b.

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3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LCO

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

During a 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 6.8.4.g, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

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

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 (Ref. 4) and Draft Regulatory Guide 1.121 (Reference 5).

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 1 gallon per minute or is assumed to increase to 1 gallon per minute for all steam generators. 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.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in RCS LCO 3.4.6.2, "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 during MODES 1, 2, 3, and 4.

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

ACTIONS

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

a.1 and a.2

ACTION a. applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by TS 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair 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, ACTION b. applies.

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

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

b.1 and b.2

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

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

SURVEILLANCE REQUIREMENTS

TS 4.4.5.1

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

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

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair 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 4.4.5.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.g contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

TS 4.4.5.2

During a 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 6.8.4.g are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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

BACKGROUND

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 LCO 3.4.1.1, "STARTUP and POWER OPERATION," LCO 3.4.1.2, "HOT STANDBY," LCO 3.4.1.3, "HOT SHUTDOWN," and LCO 3.4.1.4.1, "COLD SHUTDOWN-Loops Filled."

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.

SG 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 6.8.4.g., "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.g., 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 6.8.4.g. 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 (Reference 1).

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate greater than the operational LEAKAGE rate limits in RCS LCO 3.4.6.2, "Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves or atmospheric dump valves.

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 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the RCS LCO 3.4.8, "Specific Activity" limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 50.67 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, " Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

3. This monitoring system is not seismic Category I, but is expected to remain OPERABLE during an OBE. If the monitoring system is not OPERABLE following a seismic event, the appropriate ACTION according to Technical Specifications will be taken.

4. Two priority computer alarms (CVLKR2 and CVLKR3I) are generated if the calculated leakage rate is greater than a value specified on the Priority Alarm Point Log. This alarm value should be set to alert the Operators to a possible RCS leak rate in excess of the Technical Specification maximum allowed UNIDENTIFIED LEAKAGE. The alarm value may be set at one gallon per minute or less above the rate of IDENTIFIED LEAKAGE, from the reactor coolant or auxiliary systems, into the containment drains sump. The rate of IDENTIFIED LEAKAGE may be determined by either measurement or by analysis. If the Priority Alarm Point Log is adjusted, the high leakage rate alarm will be bounded by the IDENTIFIED LEAKAGE rate and the low leakage rate alarm will be set to notify the operator that a decrease in leakage may require the high leakage rate alarm to be reset. The priority alarm setpoint shall be no greater than 2 gallons per minute. This ensures that the IDENTIFIED LEAKAGE will not mask a small increase in UNIDENTIFIED LEAKAGE that is of concern. The 2 gallons per minute limit is also within the containment drains sump level monitoring system alarm operating range which has a maximum setpoint of 2.5 gallons per minute.

5. To convert containment drains sump run times to a leakage rate, refer to procedure SP3670.1 for guidance on the conversion method.

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3/4.4.6.2 OPERATIONAL LEAKAGE

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~~PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.~~

~~Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.~~

~~The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

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LCO RCS operational LEAKAGE shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and 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 LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Primary to Secondary LEAKAGE through Any One Steam Generator (SG)

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 4). The Steam Generator Program operational LEAKAGE performance criterion 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 result 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.

d. 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 (CONTROLLED LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

e. CONTROLLED LEAKAGE

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2250 psia. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

A limit of 40 gpm is placed on CONTROLLED LEAKAGE.

f. RCS Pressure Isolation Valve LEAKAGE

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.
In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

b., c. UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE or RCS pressure isolation valve LEAKAGE in excess of the LCO 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.

a., b. c. If any PRESSURE BOUNDARY LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, or if UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or RCS pressure isolation valve 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 STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. 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 reactor coolant pressure boundary are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1.c

CONTROLLED LEAKAGE is determined under a set of reference conditions, listed below:

a. One Charging Pump in operation.

b. RCS pressure at 2250 +/- 20 psia.

By limiting CONTROLLED LEAKAGE to 40 gpm during normal operation, it can be assured that during an SI with only one charging pump injecting, RCP seal injection flow will continue to remain less than 80 gpm as assumed in the accident analysis. When the seal injection throttle valves are set with a normal charging lineup, the throttle valve position bounds conditions where higher charging header pressures could exist. Therefore, conditions which create higher charging header pressures such as an isolated charging line, or two pumps in service are bounded by the single pump-normal system lineup surveillance configuration. Basic accident analysis assumptions are that 80 gpm flow is provided to the seals by a single pump in a runout condition.

4.4.6.2.1.d

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the reactor coolant pressure boundary 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 performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 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 water 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 RCS LCO 3.4.6.1, "Leakage Detection Systems."

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 balance.

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

4.4.6.2.1.e

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 LCO 3.4.5, "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 sampling in accordance with the EPRI guidelines (Reference 5).

4.4.6.2.2

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions for performance of Surveillance Requirement 4.4.6.2.2 (including Surveillance Requirement 4.4.6.2.2.d) for RCS pressure isolation valves which can only be leak-tested at elevated RCS pressures. The requirements of Surveillance Requirement 4.4.6.2.2.d to verify that a pressure isolation valve is OPERABLE shall be performed within 24 hours after the required RCS pressures has been met.

In MODES 1 and 2, the plant is at normal operating pressure and Surveillance Requirement 4.4.6.2.2.d shall be performed within 24 hours of valve actuation due to automatic or manual action or flow through the valve. In MODES 3 and 4, Surveillance Requirement 4.4.6.2.2.d shall be performed within 24 hours of valve actuation due to automatic or manual actuation of flow through the valve if and when RCS pressure is sufficiently high for performance of this surveillance.

<u>BACKGROUND</u>

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

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

10 CFR 50, Appendix A, GDC 30 (Reference 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Reference 2) describes acceptable methods for selecting leakage detection systems.

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

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% 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 LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES – OPERATIONAL LEAKAGE

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 all steam generators (SGs) is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

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

The FSAR (Reference 3) analysis for SGTR assumes the contaminated secondary fluid is released via atmospheric dump valves. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The safety analysis for the MSLB accident assumes 500 gpd primary to secondary LEAKAGE is through the affected steam generator and the remainder of the 1 gpm is through the intact SGs as an initial condition. The dose consequences resulting from the MSLB accident are within the guidelines based on 10 CFR 50.67 or other staff approved licensing basis.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
6. Letter FSD/SS-NEU-3713, dated March 25, 1985.
7. Letter NEU-89-639, dated December 4, 1989.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2250 psia. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

DELETE

A Limit of 40 gpm is placed on CONTROLLED LEAKAGE. CONTROLLED LEAKAGE is determined under a set of reference conditions, listed below:

a. One Charging Pump in operation.

b. RCS pressure at 2250 +/- 20 psia.

By limiting CONTROLLED LEAKAGE to 40 gpm during normal operation, we can be assured that during an SI with only one charging pump injecting, RCP seal injection flow will continue to remain less than 80 gpm as assumed in accident analysis. When the seal injection throttle valves are set with a normal charging line up, the throttle valve position bounds conditions where higher charging header pressures could exist. Therefore, conditions which create higher charging header pressures such as an isolated charging line, or two pumps in service are bounded by the single pump-normal system lineup surveillance configuration. Basic accident analysis assumptions are that 80 gpm flow is provided to the seals by a single pump in a runout condition.

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

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 reactor coolant pump seal injection and return flows.

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions for performance of Surveillance Requirement 4.4.6.2.2 (including Surveillance Requirement 4.4.6.2.2.d) for RCS pressure isolation valves which can only be leak - tested at elevated RCS pressures. The requirements of Surveillance Requirement 4.4.6.2.2.d to verify that a pressure isolation valve is OPERABLE shall be performed within 24 hours after the required RCS pressure has been met.

In MODES 1 and 2, the plant is at normal operating pressure and Surveillance Requirement 4.4.6.2.2.d shall be performed within 24 hours of valve actuation due to automatic or manual action or flow through the valve. In MODES 3 and 4, Surveillance Requirement 4.4.6.2.2.d shall be performed within 24 hours of valve actuation due to automatic or manual action or flow through the valve if and when RCS pressure is sufficiently high for performance of this surveillance.

References:

1. Letter FSD/SS-NEU-3713, dated March 25, 1985.

2. Letter NEU-89-639, dated December 4, 1989.

DELETE