Docket No. 50-423
B15193

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
24-Month Fuel Cycle

Marked-Up Pages

May 1995

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
24-Month Fuel Cycle

Section	Title	Page Number and Amendment Number
Index	New Table 4.4-2a	vii, January 1986
3.4.5	Steam Generators	3/4 4-14*, 15*, 17,* 18*, 19*, 20*
		3/4 4-16, Amendment No. 100
		3/4 4-20a, New Page
4.6.1.2	Containment Systems Containment Leakage	3/4 6-3, Amendment No. 100
Bases Section 3/4.4.5	Steam Generators	B3/4 4-3, January 1986
Bases Section 3/4.6.1.2	Containment Leakage	B3/4 6-1, Amendment No. 89

*Page for information only - no change proposed.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

0348

No change
For Info ONLY

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

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:

NO change
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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 and Insert A.

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.

Insert 'A'

a. Inservice inspections shall be performed at the frequencies specified in Table 4.4-~~2~~. 2a

b. The provisions of Specification 4.0.2 do not apply to extending the frequency for performing inservice inspections as specified in Table 4.4-~~2~~. 2a

No change
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September 11, 1989

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 imperfections 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.

No change
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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 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.

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 inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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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

No Change For info only

New Page

TABLE 4.4-2a

STEAM GENERATOR TUBE INSPECTION FREQUENCIES (A)

FINAL SAMPLE SIZE	CURRENT INSPECTION CATEGORY	PREVIOUS INSPECTION CATEGORY	INSPECTION (B) INTERVAL	
			MIN	MAX
<20%	C-1 C-1 C-2	C-1 C-2 OR C-3 N.A.	12 12 12	50 (D) (D)
20-39%	C-1 C-1 C-2	C-1 C-2 OR C-3 N.A.	12 12 12	50 30 (D)
40-99%	C-1 C-1 C-2	C-1 C-2 OR C-3 N.A.	12 12 12	50 30 (C)
100%	C-1 C-1 C-2 C-3	C-1 C-2 OR C-3 N.A. N.A.	12 12 12 12	50 30 30 (C)

(A) - Based on results of individual steam generators

(B) - Time between inspections (Calendar Months)

(C) - An engineering assessment is required to determine the maximum inspection interval, not to exceed 30 months.

(D) - Increase sample size to a higher category

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SURVEILLANCE REQUIREMENTS (Continued)

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the test by verifying that the supplemental test results, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a, 53.27 psia (38.57 psig), at intervals no greater than 24 months except for tests involving:
least once each REFUELING INTERVAL

1) Air locks

each REFUELING INTERVAL

e. The combined bypass leakage rate shall be determined to be less than or equal to 0.042 L_a by applicable Type B and C tests at least once ~~per 24 months~~ except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 53.27 psig (38.57 psig), during each Type A test;

f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

g. Purge supply and exhaust isolation valves shall be demonstrated OPERABLE by the requirements of Specifications 4.6.3.2.c and 4.9.9.

h. The provisions of Specification 4.0.2 are not applicable.
deleted

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.

INSERT X'

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.

Insert 'X':

An engineering assessment of steam generator tube integrity will confirm that no undue risk is associated with plant operation for up to 30 months from the most recent steam generator tube inspection. To provide this confirmation, the assessment would demonstrate that all tubes will retain adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the most recent inspection of each steam generator.

2. An assessment of the maximum flaw size that can be expected before the end of the current fuel cycle or 30 months, whichever comes first, and the corresponding structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the previous steam generator tube integrity assessment with actual inspection results from the most recent inspection.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50, except for the ~~performance of Type B and test~~ frequency * for the performance of Type B and Type C

The enclosure building bypass leakage paths are listed in Operating Procedure 3273, "Technical Requirements - Supplementary Technical Specifications." The addition or deletion of the enclosure building bypass leakage paths shall be made in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operation Review Committee.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does

* leakage testing. This is an exemption from Appendix J to 10CFR50.

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Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
24-Month Fuel Cycle

Retyped Pages

May 1995

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 the frequencies specified in Table 4.4-2a.

b. The provisions of Specification 4.0.2 do not apply to extending the frequency for performing inservice inspections as specified in Table 4.4-2a.

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 Operati[illegible]is 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.

TABLE 4.4-2a
STEAM GENERATOR TUBE INSPECTION FREQUENCIES (A)

FINAL SAMPLE SIZE	CURRENT INSPECTION CATEGORY	PREVIOUS INSPECTION CATEGORY	INSPECTION (B) INTERVAL	
			MIN	MAX
<20%	C-1	C-1	12	50
	C-1	C-2 OR C-3	12	(D)
	C-2	N.A.	12	(D)
20-39%	C-1	C-1	12	50
	C-1	C-2 OR C-3	12	30
	C-2	N.A.	12	(D)
40-99%	C-1	C-1	12	50
	C-1	C-2 OR C-3	12	30
	C-2	N.A.	12	(C)
100%	C-1	C-1	12	50
	C-1	C-2 OR C-3	12	30
	C-2	N.A.	12	30
	C-3	N.A.	12	(C)

(A) — Based on results of individual steam generators.
(B) — Time between inspections (Calendar Months).
(C) — An engineering assessment is required to determine the maximum inspection interval, not to exceed 30 months.
(D) — Increase sample size to a higher category.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the test by verifying that the supplemental test results, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a, 53.27 psia (38.57 psig), at least once each REFUELING INTERVAL except for tests involving:

   1) Air locks

e. The combined bypass leakage rate shall be determined to be less than or equal to 0.042 L_a by applicable Type B and C tests at least once each REFUELING INTERVAL except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 53.27 psig (38.57 psig), during each Type A test;

f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

g. Purge supply and exhaust isolation valves shall be demonstrated OPERABLE by the requirements of Specifications 4.6.3.2.c and 4.9.9.

h. Deleted.

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An engineering assessment of steam generator tube integrity will confirm that no undue risk is associated with plant operation for up to 30 months from the most recent steam generator tube inspection. To provide this confirmation, the assessment would demonstrate that all tubes will retain adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the most recent inspection of each steam generator.

2. An assessment of the maximum flaw size that can be expected before the end of the current fuel cycle or 30 months, whichever comes first, and the corresponding structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the previous steam generator tube integrity assessment with actual inspection results from the most recent inspection.

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

STEAM GENERATORS (cont'd.)

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50, except for the frequency for the performance of Type B and Type C leakage testing. This is an exemption from Appendix J to 10CFR50.

The enclosure building bypass leakage paths are listed in Operating Procedure 3273, "Technical Requirements - Supplementary Technical Specifications." The addition or deletion of the enclosure building bypass leakage paths shall be made in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operation Review Committee.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Description of the Proposed Technical
Specification Changes

May 1995

Millstone Nuclear Power Station, Unit No. 3
Description of the Proposed Technical
Specification Changes

Introduction

Beginning with Cycle 6, Millstone Unit No. 3 will operate utilizing a 24-month fuel cycle. Currently, the plant is in a scheduled refueling. To take advantage of this longer fuel cycle, NNECO is proposing to modify the frequency of a number of the surveillance requirements existing in the Millstone Unit No. 3 Technical Specifications. The proposed changes are described below:

Description of the Proposed Changes

1. Section 3.4.5, Steam Generators and Bases Section 3/4.4.5, Steam Generators

   NNECO proposes to revise Technical Specification Section 4.4.5.3, "Inspection Frequencies" for steam generators. The proposed technical specification changes provide an alternative to compensate for any delay that could cause the interval for steam generator inspections to occur near the end of a 24-month fuel cycle but before the refueling outage. The alternative includes, (1) an increase in the sample size of tubes examined, and (2) a suitable analysis of the integrity of steam generator tubes, if the inspection results are in a C-2 or a C-3 category, depending on sample size.

   The proposed changes will delete existing Surveillance Requirement 4.4.5.3.a and will replace it with a new Table 4.4-2a, "Steam Generator Tube Inspection Frequencies." Surveillance Section 4.4.5.3.b has been added to clarify that the provisions of Technical Specification Section 4.0.2 does not apply to extend the steam generator inspection interval because Technical Specification Section 4.4.5.3.a (the new Table 4.4-2a) addresses those conditions under which the 24-month surveillance interval for steam generator tube inspections may be extended.

   In addition, Bases Section 3/4.4.5, "Steam Generators," has been modified to reflect the intent of the engineering assessment for steam generator integrity addressed in Technical Specification Section 4.4.5.3.a (the new Table 4.4-2a). This addition addresses those conditions under which the 24-month surveillance interval for steam generator tube inspections may be extended.

It is noted that the above changes are based on the guidance provided by the NRC Staff in GL 91-04.

II. Section 4.6.1.2, Containment Leakage and Bases Section 3/4.6.1.2, Containment Leakage

NNECO proposes to revise Technical Specification Sections 4.6.1.2.d and 4.6.1.2.e by deleting references to 24 months and adding the words, "at least once each REFUELING INTERVAL." In addition, Section 4.6.1.2.h is being deleted. These proposed changes would increase the maximum allowable surveillance interval from 24 months to 30 months. The proposed changes are being made in accordance with the guidance contained in GL 91-04.

The amendment would also require approval of an exemption to 10CFR50, Appendix J, to allow the surveillance intervals to exceed 24 months. Such an exemption is requested in this submittal and discussed in Attachment 5.

Bases Section 3/4.6.1.2, "Containment Leakage," is being revised to indicate that an exemption from Appendix J of 10CFR50 has been granted by the NRC.

Attachment 4

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
24 Month Fuel Cycle
Safety Assessment and Significant Hazards Consideration for:

I. Safety Assessment and Significant Hazards Consideration for Changes to Section 3.4.5, Steam Generators

II. Safety Assessment and Significant Hazards Consideration for Changes to Section 4.6.1.2, Containment Leakage, Surveillance Requirements

May 1995

I. SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR CHANGES TO SECTION 3.4.5, STEAM GENERATORS

Safety Assessment

A. Background

The nuclear steam supply system uses four Westinghouse (W) Model F steam generators (SGs) which are designed to Section III of the ASME Code. Each SG contains 5626 thermally treated, Inconel 600 U-tubes hydraulically expanded into the tube sheet at each end. The SG tubing is nominally 0.688 inch outside diameter (OD) with 0.040 inch wall thickness. The tube bundle is supported with a series of 'V' shaped anti-vibration bars (AVBs) in the U-bend region and eight stainless steel tube support plates (TSP) (including the flow distribution baffle as TSP #1) in the straight section.

The surveillance requirements for the SG tubes, included in the Millstone Unit No. 3 Technical Specifications, ensure that the structural integrity of this portion of the reactor coolant system (RCS) will be maintained. The program for inservice inspection of SG tubes is based on a modification of Regulatory Guide (RG) 1.83, Revision 1. Inservice inspection of SG 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 SG tubes also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The current frequency for inservice inspection of SG tubes is not compatible with a 24-month fuel cycle. The proposed technical specification changes provide an alternative to compensate for any delay that could cause the interval for SG tube inspections to occur near the end of a 24-month fuel cycle but before the refueling outage. The alternatives include (1) an increase in the sample size of tubes examined, and (2) a suitable analysis of the integrity of SG tubes if the inspection results are in a C-2 or C-3 category, depending on sample size.

The Millstone Unit No. 3 SG tubes have experienced primarily AVB wear. To date, 18 tubes have been plugged due to the AVB wear and another 10 tubes have been plugged due to miscellaneous fabrication and preventive measures.

B. Secondary-Side Initiated Degradation

To determine the effect of cycle extension on safe SG operation with respect to secondary-side initiated degradation, it is necessary to determine how deep the flaws would become in 24 months. RG 1.121 provides the requirements for determining the degraded depth at which tubes must be repaired. The purpose of RG 1.121 is to ensure that the minimum required wall thickness to maintain acceptable margins of safety is maintained during plant operation.

The only active secondary-side SG tube degradation mechanism identified to date at Millstone Unit No. 3 is AVB wear. Since AVB wear has been shown to continue, even after SG tubes have been plugged, AVB wear rates for tubes in service, as well as plugged tubes, were evaluated against the following criteria:

1. Current 40% through-wall plugging criteria

2. 75% through-wall structural limit (RG 1.121)

An independent wear rate analysis performed by W conservatively addressed the wear issue by assuming a linear AVB wear rate based upon the highest rate identified within the Millstone Unit No. 3 SGs. The average wear rate for Millstone Unit No. 3 AVB flaws has, however, demonstrated a substantial decrease with time. This observation is also consistent with overall industry experiences. Even with conservative linear AVB wear rate assumptions, no AVB flaws are expected to exceed RG 1.121 criteria before the proposed maximum inspection interval (i.e., 50 months) allowed by the proposed technical specification change. The W AVB wear rate assumptions are based on actual operating time, not by calendar months between inspections.

Millstone Unit No. 3's future inspection program will be based upon NRC Generic Letter 91-04 whereby 20 percent of the tubes will be inspected each refueling outage. The inspection sample size may also be adjusted based on the number of steam generators inspected. (Refer to the proposed Table 4.4-2a).

Millstone Unit No. 3 steam generator inspections have not identified any other secondary-side flaws or degradation mechanisms other than AVB wear. Due to secondary-side chemistry improvements and chemistry programs in place, secondary-side corrosion is not anticipated to be a major factor during future cycles. The SG inspection programs will continue to use appropriate nondestructive examinations (NDE) techniques to effectively monitor for the development of

steam generator secondary-side tube flaws or degradation mechanisms.

In summary, no secondary-side initiated flaws are expected to exceed RG 1.121 during an extended fuel cycle.

C. **Primary-Side Degradation Mechanisms**

The Millstone Unit No. 3 SG tubing is made of thermally treated Alloy 600. This alloy is considered to be highly resistant, but not immune to primary water stress corrosion cracking (PWSCC) when compared to low temperature mill annealed Alloy 600 tubing. In addition, the U-bends of the lower 10 rows of the SG tubes have also been stress relieved. Based on current data, it is concluded that thermally treated Alloy 600, in combination with lower row stress relief, is expected to provide resistance to PWSCC. Even so, the Millstone Unit No. 3 SG inspection program will continue to include the appropriate NDE techniques for PWSCC detection. In summary, there are no structural issues that would cause PWSCC in SG tubes to adversely affect SG safety as a result of fuel cycle extension at Millstone Unit No. 3.

The current technical specification limits of one gallon per minute total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one SG will continue to limit the risk associated with any through-wall flaws. The extension to the 24-month refueling cycle will not affect these limits. Thus, the radiological assumption of the initial primary-to-secondary leakage at the technical specification limit is unaffected by the proposed changes.

D. **Probabilistic Risk Assessment**

In terms of probabilistic risk, the only impact this proposed change could potentially have is on the modelled steam generator tube rupture (SGTR) frequency. Based on the age of the plant, the previous inspection results, and the enhanced testing program, the Millstone Unit No. 3 SGTR vulnerability is expected to be lower than the industry average.

Significant Hazards Consideration

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and has concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

   The proposed changes to Surveillance Requirements 4.4.5.3.a and 4.4.5.3.b involve an increase in a surveillance interval. However, the proposed changes do not alter the intent or method by which the surveillances are conducted. In addition, the acceptance criteria for each SG tube inspection is unchanged. However, the proposed Table 4.4-2a would require an increase in the sample size of tubes examined if the inspection results fall into a C-2 or C-3 category. As such, the proposed change to the surveillance interval will not degrade the ability of the SG to perform its intended function.

   Secondary-side initiated flaws experienced at Millstone Unit No. 3 and expected in the future will not pose a threat to the safe operation of the SG. In addition, there are no structural issues that would cause PWSCC in SG tubes to adversely affect the safe operation of the SG. The current limits on primary-to-secondary leakage will continue to limit the risk associated with through-wall cracking. Thus, the radiological assumption of the initial primary-to-secondary leakage at the technical specification limit is unaffected by the proposed changes. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

   The proposed changes involve the extension of the surveillance interval for steam generator tube inspections. The proposed changes do not alter the intent or method by which the surveillances are conducted. There are no hardware changes associated with the proposed changes and no change to the functioning of any equipment which could introduce new or unique operational modes or accident precursors. Therefore, there is no possibility of a new or different kind of accident of a different type than previously evaluated.

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

   The proposed changes propose to increase the surveillance interval for SG tube inspection. As a result of a longer cycle and a longer time until inspection, flaws initiated by primary- or secondary-side degradation could occur and propagate. However, NNECO has determined that these types of tube degradation will not reduce the margins of safety as specified by RG 1.121., and therefore, there is no significant reduction in a margin of safety.

II. SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR CHANGES TO SECTION 4.6.1.2, CONTAINMENT LEAKAGE, SURVEILLANCE REQUIREMENTS

Description of the Proposed Change

10CFR50, Appendix J, Paragraphs III.D.2 and III.D.3 require that Type B and Type C containment penetration leakage rate testing be performed at intervals no greater than two years. Each of these tests are also intended to be performed during reactor shutdowns for refueling. However, Millstone Unit No. 3 will be utilizing core designs for refueling to exceed the maximum allowable two-year interval. NNECO proposes to revise Technical Specification Sections 4.6.1.2.d and 4.6.1.2.e by deleting the reference to 24 months and adding the words "at least once each REFUELING INTERVAL." In addition, Section 4.6.1.2.h is being deleted. These proposed changes would increase the maximum allowable surveillance interval from 24 months to 30 months. These proposed changes are consistent with the guidance contained in Generic Letter (GL) 91-04. Bases Section 3/4.6.1.2, "Containment Leakage," is being revised to indicate that there exists an exemption from Appendix J of 10CFR50. In conjunction with this proposed license amendment, NNECO is requesting an exemption (schedular) from the requirements of 10CFR50, Appendix J, Paragraphs III.D.2 and III.D.3.

Safety Assessment

A. Type B and Type C Surveillances

10CFR50, Appendix J, identifies a maximum allowable leakage rate, L_a at the calculated peak containment internal pressure related to the design basis accident (DBA). This maximum allowable leakage rate is used as the input assumption in dose consequence calculations for DBAs. An overall, or integrated, containment leakage rate limit has been specified as 0.75 L_a to provide a margin for increase of the leakage rate during the operating cycle. This integrated leakage is periodically verified through a Type A test. Similarly, a combined containment penetrations leakage rate limit has been specified as 0.6 L_a to provide a margin for increase of the leakage rates of the individual penetrations during the operating cycle. This combined penetrations leakage limit is periodically verified through performance of Type B and Type C leakage tests and a direct summation of their results.

Surveillance Requirement 4.6.1.2.d verifies the operability of containment electrical and gasketed penetrations and containment isolation valves to ensure that the combined leakage rate is within the technical specification limit

(i.e., 0.6 L_a). The schedule requirements provide for testing on intervals of "no greater than 24 months." The proposed revision would increase the maximum allowable surveillance interval from 24 months to 30 months by removing the "intervals no greater than" restriction on the current 24 month surveillance interval. In addition, the deletion of Section 4.6.1.2.h would allow application of Specification 4.0.2 which provides for a maximum allowable extension of 25 percent of the specified surveillance interval.

Many of the containment penetration leakage rate tests must be conducted during the shutdown, since the test requires removing from service the system which utilizes the penetration and also requires significant manpower within the containment. By allowing the standard extension of the surveillance interval, considerable flexibility is provided for scheduling refueling outages. Without the requested flexibility, refueling outages could be scheduled at no more than 24-month intervals. A short, unplanned outage during an expected 24-month fuel cycle would then result in an additional outage to perform the penetrations leakage testing on the required schedule, or would result in an early entry into the refueling outage without complete fuel utilization. Additional outages result in additional shutdown and startup transients and thermal cycles for plant equipment. Therefore, removing the need for these additional outages would improve the overall safety of the plant.

Surveillances for components governed by Surveillance Requirement 4.6.1.2.d are performed under Surveillance Procedure SP3612B.3 for Type B components and SP3612B.4 for Type C components. The Type C components covered by the above Surveillance Requirement 4.6.1.2.d are listed in the Millstone Unit No. 3 Final Safety Analysis Report (FSAR) Table 6.2-65. There are eighty-four (84) Type B components. Eighty are electrical penetrations and the remaining four (4) are the personnel air lock, equipment hatch, fuel transfer tube, and equipment hatch manway. Type C components consist of 170 containment isolation valves.

The leakage requirement specified in Technical Specification Limiting Condition for Operations (LCO) requires that the combined leakage from all penetrations and valves subject to Type B and Type C tests be less than 0.6 L_a. Type B and C surveillances were performed in 1987, 1989, 1991, Mid-cycle 4 (1991/1992) and 1993 to meet regular surveillance interval requirements. The evaluation of these surveillance results indicates that the overall performance ("As-Found" leakage) of the Type B and Type C components was good with only a few

failures. The "As-Found" maximum pathway results from the 1989 test and from the 1991 test were greater than 0.6 L_a and the "As-Left" results were within the technical specification limit.

These test failures were because of a few isolated leakage paths, all of which were corrected and were mitigated by other in series valves or closed systems. Based on a review of past surveillance test results, the "As-Left" leakage values have been below 0.6 L_a, usually only a small fraction of this specified limit. A comparison of the "As-Found" leakage rates, based on the length of the surveillance interval found no evidence that a component leakage is a function time. In addition, the satisfactory "As-Found" and "As-Left" results from the last two surveillances provide further assurance that the containment integrity and the leakage margin will be maintained with the increase in surveillance interval from 24 months to 30 months. Considering the relatively low "As-Left" Type 'B' and 'C' leakage results from the last three surveillance intervals and the corrective measures performed on the failures which occurred, there is reasonable assurance that the safety margin will be preserved with the current technical specification limit of 0.6 L_a.

The proposed change has been evaluated from a PRA perspective, and it is concluded that the change does not represent a significant risk to public safety.

B. Type B and Type C Bypass Surveillances

Surveillance Requirement 4.6.1.2.e verifies the operability of secondary containment electrical and gasketed penetrations and containment isolation valves to ensure that the combined leakage rate is within the technical specification limit. The present leakage requirement, specified in Technical Specification 3.6.1.2, requires that the combined leakage from all components subject to Type B and C Bypass tests be less than 0.042 L_a. The previous leakage limit, 0.01 L_a was increased during the implementation of a change to containment operating pressure.

Surveillances for Bypass components governed by Surveillance Requirement 4.6.1.2.e are performed under Surveillance Procedure SP3612B.3 for Type B components and SP3612B.4 for Type C components. The components covered by Surveillance Requirement 4.6.1.2.e are listed in Attachment 1.

Type B and C Bypass surveillances were performed in 1987, 1989, 1991, Mid-cycle 4 (1991/1992), and 1993. The evaluation of the surveillance results indicates that the

overall performance (i.e., "As-Found" leakage) was good with only a few failures. The few failures were attributed to containment isolation valve failures, where penetration leakage is based on the maximum pathway leakage (i.e., highest leakage of the two valves). However, a review of the valve failures and related penetrations/systems reveals that the minimum pathway leakage (i.e., smallest leakage of two valves) maintained containment integrity. The configuration of the penetrations, based on the minimum pathway leakage condition of containment, provide reasonable assurance that the containment will be maintained within its required limit.

Considering the relatively low "As-Left" Type B and C bypass leakage results from the last three surveillances and the corrective actions taken on the failures which occurred, there is a reasonable assurance that the safety margin will be preserved. In addition, the satisfactory "As-Found" and "As-Left" results from the last two surveillances provide further assurance that the containment integrity and the leakage margin will be maintained with the increase in surveillance interval from 24 to 30 months. Therefore, the proposed changes to Surveillance Requirements 4.6.1.2.d, 4.6.1.2.e, and 4.6.1.2.h have minimal impact on safety.

The proposed change has been evaluated from a PRA perspective, and it is concluded that the change does not represent a significant risk to public safety.

Significant Hazards Consideration

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

   The proposed changes to Surveillance Requirements 4.6.1.2.d and 4.6.1.2.e and deletion of Surveillance Requirement 4.6.1.2.h will increase in surveillance interval from 24 to 30 months. The proposed changes do not alter the intent or method by which the surveillances are conducted. In addition, the acceptance criterion (i.e., leakage limit) for each surveillance is unchanged. As such, the proposed changes to the surveillance interval will not degrade the ability of the containment to perform its intended function. Surveillance Requirement 4.6.1.2.a (i.e., Type A test) which is an integrated leakage rate test for containment performed at regular intervals during a 10-year service period,

provides an additional assurance that the required containment integrity will be maintained.

An evaluation of past surveillance results and corrective maintenance concluded that an increase in the interval will have a little or no impact on safety. Since the containment leakage is a direct assumption of dose calculations for all DBAs, the proposed changes could potentially affect the consequences of previously evaluated accidents. However, the assumed maximum allowable leakage rate is not changed and must continue to be met. A comparison of the "As-Found leakage rates, based on the length of the surveillance interval, found no evidence that the component leakage is a function of time. Therefore, the consequences of previously analyzed accidents are not significantly affected by these proposed changes.

The proposed changes cannot affect the probability of any previously analyzed accident, since the proposed changes only affect the surveillance interval.

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

   The proposed changes to Surveillance Requirements 4.6.1.2.d and 4.6.1.2.e do not change the design or operation of any plant system. The proposed changes do not alter the intent or method by which the surveillances are conducted other than increasing the interval from 24 months to 30 months. The proposed changes do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions, and do not modify the manner in which the plant is operated. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The evaluation of past surveillances and corrective maintenances concluded that an increase in the surveillance interval will have little impact on safety. Additionally, the method of performing the surveillance is unchanged. Since the containment leakage is a direct assumption of dose calculation for all DBAs and the assumed maximum allowable leakage rate is not changed and based on the satisfactory "As-Found" and "As-Left" results from the last two surveillances, especially the low "As-Left" Type B and Type C results, further assurance is provided that containment leakage will be maintained within its required limit, therefore, the proposed changes do not involve a significant reduction in the margin of safety.

TABLE 1

Components Tested by Technical Specification
4.6.1.2.e

PENETRATION NO.	COMPONENT	DESCRIPTION	TYPE OF TEST
3CS*HATCH2	EQUIP. HATCH	DBL SEAL	B
	EQUIP. MANWAY	DBL SEAL	B
3FWT*TFT1	FUEL TFR. TUBE	DBL SEAL	B
14	3SIL*CV8969 3SIL*CV8880	N2 TO SAFETY INJ	C C
15(i)	3PGS*CV8046	PRI. GRADE WATER TO PRT	C
(o)	3PGS*CV8028 3PGS*RV77		C C
35(o) (o)	3CVS*CTV20A 3CVS*CTV21A	CTMT VACUUM PUMP SUCTION	C C
36(o) (o)	3CVS*CTV20B 3CVS*CTV21B	CTMT VACUUM PUMP SUCTION	C C
37(i)	3CVS*AOV23 3CVS*V20	CTMT VACUUM PUMP SUCTION	C C
38(i)	3CDS*CTV91A	CHILL WATER SUPPLY	C
(o)	3CDS*CTV38A 3CDS*RV105A		C C
45(i)	3CDS*CTV40B	CHILL WATER RETURN	C
(o)	3CDS*CTV39B 3CDS*RV106B		C C
52(i) (o)	3SAS*V875 3SAS*V50	SERVICE AIR	C C
54	3IAS*MOV72 3IAS*PV15	INSTRUMENT AIR	C C
56(i)	3FPW*CTV49 3FPW*V661	FIRE PROTECTION	C C
(o)	3FPW*CTV48 3FPW*V666 3FPW*RV87		C C C
59(i) (o)	3SFC*V991 3SFC*V992	CTMT FPC IN	C C
60(i) (o)	3SFC*V990 3SFC*V989	REFUELING CAVITY OUT	C C

TABLE 1

Components Tested by Technical Specification
4.6.1.2.e (cont'd)

PENETRATION NO.	COMPONENT	DESCRIPTION	TYPE OF TEST
70(i) (o)	3CCP*V886 3CCP*V887	DIMIN. WATER SUPPY	C C
72(i)	3CDS*CTV91B	CHILL WATER SUPPLY	C
(o)	3CDS*CTV38B 3CDS*RV105B		C C
85	3HVU*CTV33B 3HVU*CTV32B	CTMT PURGE AIR EXHAUST	C C
86	3HVU*CTV33A 3HVU*CTV32A 3HVU*V5	CTMT PURG AIR SUPPLY	C C C
116(i)	3CDS*CTV40A	CHILL WATER RETURN	C
(o)	3CDS*CTV39A 3CDS*RV106A		C C
124	3GSN*CTV105 3GSN*CTV8033	N2 SUPPLY HEADER	C C

Attachment 5

Millstone Nuclear Power Station, Unit No. 3

Request for Exemption From 10CFR50, Appendix J

May 1995

Millstone Nuclear Power Station, Unit No. 3
Request for Exemption From 10CFR50, Appendix J

Purpose

Northeast Nuclear Energy Company (NNECO) is requesting, on behalf of Millstone Unit No. 3, a schedular exemption from the requirements of III.D.2 and III.D.3 of Appendix J to 10CFR50. This schedular exemption would permit Millstone Unit No. 3 to perform Type B and Type C tests and bypass leakage tests beyond 24 months, but not to exceed 30 months between each test. The exemption would be valid for one cycle only.

In addition to this exemption request, NNECO is preparing a change to Surveillance Requirements 4.6.1.2.d and 4.6.1.2.e and to delete 4.6.1.2.h which would increase the maximum allowable surveillance interval from 24 months to 30 months.

Background

Millstone Unit No. 3 has implemented a testing program to measure containment leakage throughout the life of the plant. The testing program conforms to the requirements of Appendix J to 10CFR50. It includes the performance of Type A tests to measure the overall integrated leakage rate, Type B tests to detect and measure local leakage across electrical and gasketed penetrations, Type C to measure containment isolation valves leakage rates, and Type B and C bypass tests to measure secondary containment penetrations and containment isolation valves leakage rates.

10CFR50, Appendix J, Paragraphs III.D.2 and III.D.3 require that Type B and Type C containment penetration leakage rate testing be performed at intervals no greater than two years. Each of these tests are also intended to be performed during reactor shutdowns for refueling. However, Millstone Unit No. 3 will be utilizing core designs which, when considering unplanned outages, if applicable will allow the intervals between reactor shutdowns for refueling to exceed the maximum allowable two year interval. The use of new extended-cycle core design has been recognized as a growing trend in the industry by the NRC as discussed in Generic Letter (GL) 91-04.

GL 91-04 also indicates that the NRC Staff is developing changes to Appendix J to 10CFR50 to accommodate a 24-month fuel cycle and to resolve other problems with the regulation. The requested technical specification change and exemption are, therefore, expected to be temporary, in that they would provide an acceptable basis for testing intervals beyond 24 months until the

regulations are revised and other acceptable bases are established.

Discussion

NNECO is requesting, on behalf of Millstone Unit No. 3, a schedular exemption from the requirements of III.D.2 and III.D.3 of Appendix J to 10CFR50. The schedular exemption would permit Millstone Unit No. 3 to perform Type B and Type C tests and bypass leakage tests beyond 24 months, but not to exceed 30 months between each test. The exemption would be valid for one cycle only.

Justification

10CFR50.12 states that the Commission may grant exemptions from the regulations in 10CFR50 provided that they are "authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security."

1. The Requested Exemption is Authorized by Law.

   10CFR50, Appendix J, Paragraphs III.D.2 and III.D.3 require that Type B and Type C containment penetration leakage rate testing be performed at intervals no greater than two years. Each of these tests are also intended to be performed during reactor shutdown for refueling. However, Millstone Unit No. 3 will be utilizing core designs which will allow the intervals between reactor shutdowns for refueling to exceed the maximum allowable two-year interval. The use of new extended-cycle core design has been recognized as a growing trend to the industry by the NRC, as discussed in GL 91-04. Similar schedular exemptions have been granted, on an individual penetration basis, for other facilities and for Millstone Unit No. 3. Therefore, the Commission is authorized by law to grant this exemption.

2. The Requested Exemption Does Not Present an Undue Risk to the Public Health and Safety.

   10CFR50, Appendix J, identifies a maximum allowable leakage rate, L_a at the calculated peak containment internal pressure related to the design basis accident (DBA). This maximum allowable leakage rate is used as the input assumption in dose consequence calculations for DBAs. An overall, or integrated, containment leakage rate limit has been specified as 0.75 L_a to provide a margin for increase of the leakage rate during the operating cycle. This integrated leakage is periodically verified through a Type A test. Similarly, a combined containment penetrations leakage rate limit has been specified as 0.6 L_a to provide a margin for increase of the

leakage rates of the individual penetration during the operating cycle. This combined penetrations leakage limit is periodically verified through performance of Type B and Type C leakage tests and a direct summation of their results.

The leakage requirement specified in Technical Specification Limiting Condition for Operation (LCO) requires that the combined leakage from all penetrations and valves subject to Type B and Type C tests be less than 0.6 L_a. Type B and C surveillances were performed in 1987, 1989, 1991, Mid-cycle 4 (1991/1992), and 1993 to meet regular surveillance interval requirements. The evaluation of these surveillance results indicates that the overall performance ("As-Found leakage) of the Type B and Type C components was good with only a few failures. The "As-Found" maximum pathway results from the 1989 test and from the 1991 test were greater than 0.6 L_a and the "As-Left" results were within the technical specification limit.

These test failures were because of a few isolated leakage paths, all of which were corrected and mitigated by other in-series valves or closed systems. Based on a review of past surveillance test results, the "As-Left" leakage values have been below 0.6 L_a, usually only a small fraction of this specified limit. A comparison of the "As-Found" leakage rates, based on the length of the surveillance interval found no evidence that a component leakage is a function time. In addition, the satisfactory "As-Found" and "As-Left" results from the last two surveillances provide further assurance that the containment integrity and the leakage margin will be maintained with the increase in surveillance interval from 24 months to 30 months. Considering the relatively low "As-Left" Type 'B' and 'C' leakage results from the last three surveillance intervals and the corrective measures performed on the failures which occurred, there is reasonable assurance that the safety margin will be preserved with the current technical specification limit of 0.6 L_a.

Surveillance Requirement 4.6.1.2.e verifies the operability of secondary containment electrical and gasketed penetrations and containment isolation valves to ensure that the combined leakage rate is within the technical specification limit. The present leakage requirement, specified in Technical Specification 3.6.1.2, requires that the combined leakage from all components subject to Type B and C Bypass tests be less than 0.042 L_a. The previous leakage limit, 0.01 L_a was increased during the implementation of a change to containment operating pressure.

Type B and C bypass surveillances were performed in 1987, 1989, 1991, Mid-cycle 4 (1991/1992), and 1993. The

evaluation of the surveillance results indicates that the overall performance (i.e., "As-Found" leakage) was good with only a few failures. The few failures were attributed to containment isolation valve failures, where penetration leakage is based on the maximum pathway leakage (i.e., highest leakage of the two valves). However, a review of the valve failures and related penetrations/systems reveals that the minimum pathway leakage (i.e., smallest leakage of two valves) maintained containment integrity. The configuration of the penetrations, based on the minimum pathway leakage condition of containment, provide reasonable assurance that the containment will be maintained within its required limit.

Considering the relatively low "As-Left" Type B and C Bypass leakage results from the last three surveillances and the corrective actions taken on the failures which occurred, there is reasonable assurance that the safety margin will be preserved. In addition, the satisfactory "As-Found" and "As-Left" results from the last two surveillances provide further assurance that the containment integrity and the leakage margin will be maintained with the increase in surveillance interval from 24 to 30 months.

An evaluation of past surveillance results and corrective maintenance concluded that an increase in the interval will have a little or no impact on safety. Since the containment leakage is a direct assumption of dose calculations for all DBAs, the proposed changes could potentially affect the consequences of previously evaluated accidents. However, the assumed maximum allowable leakage rate is not changed and must continue to be met. A comparison of the "As-Found leakage rates, based on the length of the surveillance interval, found no evidence that the component leakage is a function of time. Therefore, the consequences of previously analyzed accidents are not significantly affected by these proposed changes. Therefore, the requested exemption does not present an undue risk to the public health and safety.

3. <u>The Requested Exemption Will Not Endanger the Common Defense and Security.</u>

   This activity, the containment penetration leak rate testing (i.e., Type B and C), is not considered in the common defense and security of the nation. Therefore, this exemption will not impact the common defense and security.

<u>Special Circumstances</u>

Additionally, 10CFR50.12(a)(2) states that "the Commission will not consider granting an exemption unless special circumstances are present," then it provides a list of special circumstances.

In this instance, special circumstances are applicable. They are 10CFR50.12(a)(2)(iii) and 10CFR50.12(a)(2)(v).

10CFR50.12(a)(2)(iii) states that the Commission may grant exemptions from requirements of 10CFR50 where "compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by other similarly situated." 10CFR50, Appendix J, Paragraphs III.D.2 and III.D.3 require that Type B and Type C containment penetration leakage rate testing be performed at intervals no greater than two years.

Many of the containment penetration leakage rate tests must be conducted during the shutdown since the test requires removing for service the system which utilizes the penetration and also requires significant manpower within the containment. By allowing the standard extension of the surveillance interval, considerable flexibility is provided for scheduling refueling outages. Without the requested flexibility, refueling outages could be scheduled at no more than 24-month intervals. A short, unplanned outage during an expected 24-month fuel cycle would then result in an additional outage to perform the penetrations leak testing on the required schedule, or would result in an early entry into the refueling outage without complete fuel utilization which, in turn, would increase significant cost beyond that intended by the regulation. Additional outages result in additional shutdown and startup evolutions and thermal cycles for plant equipment. Therefore, removing the need for these additional outages would improve the overall safety of the plant.

10CFR50.12(a)(2)(v) states that the Commission may grant exemptions from requirements of 10CFR50 where "the exemption would provide only temporary relief from the applicable regulation and licensee or applicant has made good faith efforts to comply with the regulations." The request for a schedular exemption is only for one cycle only. Therefore, the schedular exemption is warranted because the exemption would provide only temporary relief from the applicable regulation and licensee has made good efforts to comply with the regulation.

Conclusion

NNECO concludes that the request for a schedular exemption from the requirements of Paragraphs III.D.2 and III.D.3 of Appendix J to 10CFR50 are justified pursuant to 10CFR50.12(a)(1) and 10CFR50.12(a)(2)(iii) and 10CFR50.12(a)(2)(v).