NMC
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Nuclear Management Company, LLC

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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Supplement to Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity (TAC Nos. MD0209 and MD0210)

References 1) License Amendment Request (LAR) titled, "Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity", dated February 16, 2006, Accession Number ML060480440.

2) Supplement to Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity, dated July 21, 2006. Accession Number ML062370052.

By letter dated February 16, 2006 (Reference 1), Nuclear Management Company (NMC) submitted an LAR to adopt Technical Specification (TS) improvements regarding steam generator tube integrity provided in Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-449, "Steam Generator Tube Integrity", Revision 4. By letter dated July 21, 2006 (Reference 2), NMC submitted proposed TS and Bases changes which replaced in their entirety the changes proposed in Reference 1. This letter supplements the referenced LAR to address the Nuclear Regulatory Commission (NRC) Staff requests for additional information (RAIs) sent by email on August 29, 2006, October 25, 2006 and November 9, 2006 regarding Enclosures 1 and 2 of Reference 2. NMC is submitting this supplement in accordance with the provisions of 10 CFR 50.90.

Enclosure 1 provides the NRC RAIs and NMC responses. Enclosure 2, which includes the TS and Bases pages marked up in response to the RAIs, replaces Enclosure 2 of Reference 2 in its entirety. Additions to the current TS and Bases are shown with double-underline and deletions are shown with strikethrough. The proposed changes associated with this supplement appear in Enclosure 2 on pages 5.0-13, 5.0-14, 5.0-20, 5.0-21, 5.0-22, 5.0-26, 5.0-27, 5.0-28, 5.0-40, 5.0-41, B 3.4.14-2, B 3.4.19-2 and B 3.4.19-3. Enclosure 3, which includes the TS pages revised in response to the RAIs, replaces Enclosure 3 of Reference 2 in its entirely.

The additional information provided in this supplement does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the referenced February 16, 2006 submittal as supplemented July 21, 2006.

In accordance with 10 CFR 50.91, NMC is providing a copy of this letter and enclosures to the designated State Official.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on **27 December 2006.**

GSalamon

Gabor Salamon
Acting Director, Nuclear Licensing and Regulatory Services
Nuclear Management Company, LLC

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island Nuclear Generating Plant, USNRC
Senior Resident Inspector, Prairie Island Nuclear Generating Plant, USNRC
State Official, Minnesota Department of Commerce

Enclosure 1

Nuclear Regulatory Commission Requests for Additional Information and Nuclear Management Company Responses

The NRC Staff provided comments on revisions of Nuclear Management Company (NMC) proposed Technical Specifications (TS) and Bases in emails dated August 29, 2006, October 25, 2006 and November 9, 2006. The NRC Staff comments and NMC responses are provided below for each TS paragraph or Bases page on which comments were received. Page numbers refer to the page in Enclosure 2.

General Comment

October 25, 2006, Comment 1

Any strikeouts/underlines should be based on their current Tech Specs (not previous versions of their proposal since their previous versions are inconsequential). As currently written, it is almost impossible to read without spending an inordinate amount of time. All that matters is the currently approved TS and the current proposal. Prior versions of the proposal are immaterial. If they keep it as is, there will be a lot of effort involved in reviewing their submittal.

NMC Response:

The current TS and Bases were marked up to show additions and deletions incorporating Technical Specifications Task Force (TSTF) industry traveler 449 (TSTF-449) with consideration for all NRC requests for additional information (RAIs) and email comments. Additions and deletions associated with previous versions have been removed. Enclosures 2 and 3 to this letter show the current proposal.

TS 5.5.8.b.1, Page 5.0-13

August 29, 2006, Comment 2

A safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials was indicated in TS Section 5.5.8.b.1. GL 95-05 indicated that there is a possibility that a tube may have a burst pressure less than 1.4 times the steam line break pressure differential (given the uncertainties associated with the various correlations), therefore, the GL 95-05 alternate repair criteria (ARC) imposed a limit on the POB [probability of burst] of 1x10-2. As currently proposed, the flaws to which the voltage-based ARC is applied must maintain a safety factor of 1.4 against burst during design basis accidents. Since this is inconsistent with the staff's original approval (as evidenced by the probability of burst criteria), please verify that this was your intent. If this was not your intent, please discuss your plans to modify your submittal to address this

issue. Discuss your plans to clarify your proposal, for example: "This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and, except for flaws addressed through application of the alternate repair criteria discussed in Specification 5.5.8.c.2(c), a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials."

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff and now states:

> This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and, except for flaws addressed through application of the alternate repair criteria discussed in Specification 5.5.8.c.2(c), a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials."

<u>TS 5.5.8.b.1, Page 5.0-14</u>

August 29, 2006, Comment 1

In your proposed Structural Integrity Performance Criteria (SIPC) in TS 5.5.8.b.1, you stated the following: "For Unit 2, when tubes are left in service with predominantly axially oriented stress corrosion cracking at the tube support plate (TSP) elevations, the probability of burst (POB) under main steam line break conditions shall be maintained below 1E-02 in accordance with the requirements of NRC Generic Letter (GL) 95-05." As currently proposed, once tubes are left in service with predominantly axially oriented stress corrosion cracking at the tube support plate elevations, the probability of burst for all indications (even those that are not axially oriented stress corrosion cracking at TSP locations) is limited to 1x10-2. In addition, since NRC GL 95-05 does not contain any "requirements," the last portion of this statement is not accurate. If it was not your intent to have the 1x10-2 criteria apply to all forms of degradation, please discuss your plans to modify your submittal.

Please discuss your plans to address the above. The proposed TS may be modified by using something similar to the following:

For Unit 2, when alternate repair criteria discussed in Specification 5.5.8.c.2(c) are applied to axially oriented outside diameter stress corrosion cracking indications at tube support plate locations, the probability that one or more of these indications in a SG [steam generator] will burst under postulated main steam line break conditions shall be less than 1x10-2.

Please note that your Bases may also need to be revised to clarify this issue.

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff and now states:

> For Unit 2, when alternate repair criteria discussed in Specification 5.5.8.c.2(c) are applied to axially oriented outside diameter stress corrosion cracking indications at the tube support plate locations, the probability that one or more of these indications in an SG will burst under postulated main steam line break conditions shall be less than 1E-02.

The Bases changes are discussed below.

<u>5.5.8.b.2, Page 5.0-14</u>

August 29, 2006, Comment 3

Regarding TS 5.5.8.b.2, you reference the "voltage-based repair criteria." Since this reference isn't specific, it could be misinterpreted to apply to any flaws to which a voltage-based sizing method is applied. As a result, discuss your plans to clarify your proposed TS to indicate that the "voltage-based repair criteria" that you are referring to is the one in TS 5.5.8.c.2(c).

NMC Response:

This TS paragraph was revised to specifically reference Specification 5.5.8.c.2(c).

<u>5.5.8.c.2, Page 5.0-14</u>

August 29, 2006, Comment 4

As currently written, it is not clear whether all of the criteria listed under TS 5.5.8.c.2 must be met in order to require plugging or repair. In addition, the criteria under TS 5.5.8.c.2 not only discuss the criteria for plugging and repair, but also criteria for leaving flaws in service. As a result, please discuss your plans to modify your submittal to address this issue. For example: "Unit 2 steam generator tubes found by inservice inspection to contain flaws shall be dispositioned as follows:"

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff and now states, "Unit 2 steam generator tubes found by inservice inspection to contain flaws shall be dispositioned as follows:".

<u>5.5.8.c.2(a)(1), Page 5.0-20</u>

August 29, 2006, Comment 5

It appears that TS 5.5.8.c.2(a)(1) and TS 5.5.8.c.2(a)(2) are intended to address the repair criteria for the non-sleeved and sleeved region of the tube, respectively. In your current proposal (and TSTF-449), a "tube" is considered to include the tube wall and any repairs to it. As a result, it would appear that there are two different set of repair limits for the sleeves (since TS 5.5.8.c.2(a)(1) and TS 5.5.8.c.2(a)(2) apply to the sleeve). Please discuss your plans to clarify that TS 5.5.8.c.2(a)(1) addresses the non-sleeved region of the tube and TS 5.5.8.c.2(a)(2) addresses the sleeved region of the tube.

October 25, 2006, Comment 2

In TS 5.5.8.c.2(a)(1), they should broaden the exception to include the F*/EF* criteria (i.e., except if permitted to remain in service through application of the alternate tube repair criteria discussed in Specification 5.5.8.c.2(b) or Specification 5.5.8.c.2(c).

NMC Response:

This TS paragraph was revised to apply only to a flaw in a non-sleeved region of the tube and the exception recommended in the October 25, 2006, Comment 2 was incorporated.

<u>5.5.8.c.2(a)(2), Page 5.0-20</u>

August 29, 2006, Comment 5

It appears that TS 5.5.8.c.2(a)(1) and TS 5.5.8.c.2(a)(2) are intended to address the repair criteria for the non-sleeved and sleeved region of the tube, respectively. In your current proposal (and TSTF-449), a "tube" is considered to include the tube wall and any repairs to it. As a result, it would appear that there are two different set of repair limits for the sleeves (since TS 5.5.8.c.2(a)(1) and TS 5.5.8.c.2(a)(2) apply to the sleeve). Please discuss your plans to clarify that TS 5.5.8.c.2(a)(1) addresses the non-sleeved region of the tube and TS 5.5.8.c.2(a)(2) addresses the sleeved region of the tube.

August 29, 2006, Comment 6

In proposed TS 5.5.8.c.2(a)(2), you indicated that the repair criteria for the original tube wall in the sleeve to tube joint is 25-percent of the nominal sleeve wall thickness. This does not appear to be consistent with your current technical specifications (and it probably is not consistent with the design and licensing basis for the sleeves). The staff believes that you intended to indicate that the repair criteria for the sleeve is 25-percent of the sleeve wall thickness and that the repair criteria for the parent tube at the sleeve-to-tube joint is to plug on detection. Please discuss your plans to modify your proposal to address this issue.

In addition, as currently written, proposed TS 5.5.8.c.2(a)(2) would permit tubes to be either plugged or repaired in the event that flaws exceeded the repair criteria. Please discuss your plans to indicate that flaws that exceed these repair limits must be plugged.

NMC Response:

The TS paragraph was revised to apply only to a flaw in a sleeved region of the tube and require tube plugging when the criterion is exceeded. A new TS paragraph 5.5.8.c.2(a)(3) was added to require plugging of tubes with a flaw in a sleeve to tube joint.

<u>5.5.8.c.2(a)(3), Page 5.0-20</u>

August 29, 2006, Comment 6

In proposed TS 5.5.8.c.2(a)(2), you indicated that the repair criteria for the original tube wall in the sleeve to tube joint is 25-percent of the nominal sleeve wall thickness. This does not appear to be consistent with your current technical specifications (and it probably is not consistent with the design and licensing basis for the sleeves). The staff believes that you intended to indicate that the repair criteria for the sleeve is 25-percent of the sleeve wall thickness and that the repair criteria for the parent tube at the sleeve-to-tube joint is to plug on detection. Please discuss your plans to modify your proposal to address this issue.

In addition, as currently written, proposed TS 5.5.8.c.2(a)(2) would permit tubes to be either plugged or repaired in the event that flaws exceeded the repair criteria. Please discuss your plans to indicate that flaws that exceed these repair limits must be plugged.

October 25, 2006, Comment 3

TS 5.5.8.c.2(a)(3) should be clarified to indicate that "Tubes with a flaw in a sleeve to tube joint that occurs in the original tube wall of the joint shall be plugged." (to avoid potential overlap with the prior requirement).

NMC Response:

This TS paragraph was added to require plugging of tubes with a flaw in a sleeve to tube joint and the phrasing recommended in the October 25, 2006, Comment 3 was incorporated.

5.5.8.c.2(b), Page 5.0-20

August 29, 2006, Comment 7

In proposed TS 5.5.8.c.2(b), it would appear that the following phrase is not needed since it is also contained in proposed TS 5.5.8.c.2(b)(1) and (2): "Flaws may be left in service when they are located below F* or EF* [region] defined below:." Please discuss your plans to remove this phrase.

August 29, 2006, Comment 15

In your July 21, 2006 response to question 3, you stated (see item 2) that the F* and EF* criteria could be applied to the cold-leg side of the tubesheet. At the time the F* and EF* criteria were approved, your technical specification only addressed the hot-leg portion of the tubesheet (i.e., no inspections were required by the technical specifications in the cold-leg). At the time of these F* and EF* proposals, no modifications were made to the technical specifications to require cold-leg inspections. As a result, the staff reviewed your proposal to incorporate technical specification inspection and repair criteria for the hot-leg. As a result of the above, discuss your plans to submit for review and approval, the structural and leakage integrity analysis for application of the F* and EF* criteria to the cold-leg or alternatively discuss your plans to clarify that the F* and EF* criteria apply to the hot-leg.

October 25, 2006, Comment 4

Regarding the F*/EF* criteria, reference should be made to TS 5.5.8.c.2(a)(1). That is, "....alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1) (since it would be inappropriate to apply this to the depth based criteria of sleeves).

October 25, 2006, Comment 5

It would seem that it should be clear that the F* criterion does not apply to tubes that have a sleeve installed below the uppermost hardroll transition.

October 25, 2006, Comment 6

For the voltage based repair criteria, I have similar comments as made above regarding the F* criterion.

November 9, 2006, Comment 9

We still have the issue with respect to the F*/EF* criteria and the cold leg.

NMC Response:

This TS paragraph was revised to remove the sentence, ""Flaws may be left in service when they are located below F* or EF* [region] defined below: . ." and specifically reference Specification 5.5.8.c.2(a)(1).

NMC agrees that the F* criterion and voltage based repair criteria do not apply to tubes that have a sleeve installed below the uppermost hardroll transition. No TS changes were made to address these comments.

NMC has added, "to the hot-leg of the tubesheet" in this paragraph to restrict the use of the F* and EF* criteria to the hot-leg.

<u>5.5.8.c.2(b)(1), Page 5.0-21 and 5.5.8.c.2(b)(2), Page 5.0-21</u>

August 29, 2006, Comment 8

In several instances, the term "defect" is used in your proposed TS (e.g., 5.5.8.c.2(b)(1), proposed TS 5.5.8.c.2(b)(2), and proposed TS 5.6.7.a.10). Since a "defect" is not defined in your proposed TS, please discuss your plans to replace this term with "flaw" which is the term used in TSTF-449. In addition, the term "degradation" is used in your proposed TS (e.g., 5.5.8.c.2(c)(1) and 5.5.8.c.2(c)(2)). Since "degradation" is not defined in your proposed TS, please discuss your plans to replace this term with "flaw" which is the term used in TSTF-449.

November 9, 2006, Comment 1

In 5.5.8.c.2.b.1 and 5.5.8.c.2.b.2, it appears that the second from the last sentence should be modified (This 1.07-inch span (not including eddy current uncertainty) is referred to as the F* region.) I believe this sentence should be modified to

indicate that "This 1.07-inch span (when increased for eddy current uncertainty) is referred to as the F* region." The corresponding change should also be made to the EF* section. The reason for the change is that the F* region definition is used to indicate that all tubes with flaws in this region should be plugged or repaired. As currently written, one could interpret the sentence as the F* region does not include eddy current uncertainty (which is not the correct interpretation).

NMC Response:

These TS paragraphs were revised to replace "defects" with "flaws" as proposed by the NRC Staff. The parenthetical statements following the last mention of the 1.07-inch span in TS 5.5.8.c.2(b)(1) and the last mention of the 1.67-inch span in TS 5.5.8.c.2(b)(2) were revised to state, "increased for measurement uncertainty" as agreed to in a phone call with the NRC Staff on November 21, 2006.

Revisions to TS 5.5.8.c.2(c)(1), TS 5.5.8.c.2(c)(2) and TS 5.6.7.a.10 are discussed below.

<u>5.5.8.c.2(c), Page 5.0-21</u>

November 9, 2006, Comment 2

In 5.5.8.c.2.c, they should add "....as an alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1)." This will make it similar to the F* criterion writeup (but more appropriately it clarifies that these alternate repair criteria can only be applied to the non-sleeved portion of the tube).

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff and now states, ". . .as an alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1)".

<u>5.5.8.c.2(c)(1), Page 5.0-21 and 5.5.8.c.2(c)(2), Pages 5.0-21 and 5.0-22</u>

August 29, 2006, Comment 8

In several instances, the term "defect" is used in your proposed TS (e.g., 5.5.8.c.2(b)(1), proposed TS 5.5.8.c.2(b)(2), and proposed TS 5.6.7.a.10). Since a "defect" is not defined in your proposed TS, please discuss your plans to replace this term with "flaw" which is the term used in TSTF-449. In addition, the term "degradation" is used in your proposed TS (e.g., 5.5.8.c.2(c)(1) and 5.5.8.c.2(c)(2)).

Since "degradation" is not defined in your proposed TS, please discuss your plans to replace this term with "flaw" which is the term used in TSTF-449.

NMC Response:

These TS paragraphs were revised to replace “degradation” with “indication” since bobbin voltages identify “indications” rather than “flaws”.

<u>5.5.8.d, Page 5.0-26</u>

August 29, 2006, Comment 9

Please discuss your plans to indicate in TS 5.5.8.d that: "In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection."

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff.

<u>5.5.8.d.3(a), Page 5.0-27</u>

August 29, 2006, Comment 10

In proposed TS 5.5.8.d.3(a), you indicate that the region of the tube below the F* and EF* regions may be excluded from the inspection requirements. In addition, in your response to question 4c in your July 21, 2006 letter (ML062370052), you indicate that full depth tubesheet sleeves are installed at the lower end of the parent tube (presumably this is near the tube-to-tubesheet weld). Since this latter region is below the F* and EF* region, it would appear that a tube in which a full depth tubesheet sleeve is installed may not require an inspection near the lower end of the sleeve (depending on exactly where the sleeve is installed with respect to the F* and EF* region). As a result, please discuss your plans to modify your proposal to ensure that full depth tubesheet sleeves require an inspection.

August 29, 2006, Comment 11

In proposed TS 5.5.8.d.3(a), you reference a "refueling outage inspection." Under the proposed TS, inspections need not be performed during a refueling outage. They only need to be performed at intervals not to exceed 24 effective full power months or one operating interval between refueling outages (whichever is less). As a result, if you were to elect to perform inspections at times other than

refueling outages, the F* and EF* region may not be inspected for multiple cycles. Since this is inconsistent with your current requirements (and the design/licensing basis), discuss your plans to modify your submittal to indicate that the "F* and EF* tubes" will be inspected in the F* and EF* regions every 24 effective full power months or one refueling outage (whichever is less). A similar comment applies to proposed TS 5.5.8.d.3(c) which references inspections during refueling outages.

October 25, 2006, Comment 7

In 5.5.8.d.3.a, the term "periodic" is introduced. Since this is confusing it should be rewritten (e.g., one may interpret this as the 60 month periodic inspection and this would be inappropriate). I would suggest terminology such as "every 24 EFPMs or one refueling outage (whichever is less)." A similar comment applies to other uses of "periodic"

November 9, 2006, Comment 3

In 5.5.8.d.3.a, the last sentence is awkward. I would suggest the following: The region of these tubes below the F* and EF* regions do not need to be inspected unless there is a sleeve (or portion of a sleeve) that extends below the F* or EF* region.

NMC Response:

The NRC comments have been resolved through adoption of the parenthetical phrase, "every 24 effective full power months (EFPM) or one refueling outage (whichever is less)", in the first sentence as suggested by the NRC Staff in the October 25, 2006 Comment 7 and revision of the last sentence as suggested in the November 9, 2006 Comment 3. Resolution of these comments as applicable to other TS paragraphs is discussed below.

<u>5.5.8.d.3(b), Page 5.0-27</u>

August 29, 2006, Comment 12

In proposed TS 5.5.8.d.3(b) and (c), you refer to the repair criteria discussed in proposed TS 5.5.8.c.2(c) using different terminology. This can cause confusion on what is being referred to (since neither of these sections match the "title" in 5.5.8.c.2(c). As a result, please discuss your plans to modify these two sections to simply reference the "alternate repair criteria discussed in TS 5.5.8.c.2(c)." A similar comment applies to proposed TS 5.6.7.b.

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff.

5.5.8.d.3(c), Page 5.0-28

August 29, 2006, Comment 11

In proposed TS 5.5.8.d.3(a), you reference a "refueling outage inspection." Under the proposed TS, inspections need not be performed during a refueling outage. They only need to be performed at intervals not to exceed 24 effective full power months or one operating interval between refueling outages (whichever is less). As a result, if you were to elect to perform inspections at times other than refueling outages, the F* and EF* region may not be inspected for multiple cycles. Since this is inconsistent with your current requirements (and the design/licensing basis), discuss your plans to modify your submittal to indicate that the "F* and EF* tubes" will be inspected in the F* and EF* regions every 24 effective full power months or one refueling outage (whichever is less). A similar comment applies to proposed TS 5.5.8.d.3(c) which references inspections during refueling outages.

August 29, 2006, Comment 12

In proposed TS 5.5.8.d.3(b) and (c), you refer to the repair criteria discussed in proposed TS 5.5.8.c.2(c) using different terminology. This can cause confusion on what is being referred to (since neither of these sections match the "title" in 5.5.8.c.2(c). As a result, please discuss your plans to modify these two sections to simply reference the "alternate repair criteria discussed in TS 5.5.8.c.2(c)." A similar comment applies to proposed TS 5.6.7.b.

October 25, 2006, Comment 7

In 5.5.8.d.3.a, the term "periodic" is introduced. Since this is confusing it should be rewritten (e.g., one may interpret this as the 60 month periodic inspection and this would be inappropriate). I would suggest terminology such as "every 24 EFPMs or one refueling outage (whichever is less)." A similar comment applies to other uses of "periodic"

November 9, 2006, Comment 4

In 5.5.8.d.3.c, they should confirm that the Spec referenced is 5.5.8.c.2(c) since I could not read portions of the spec in the hard copy that I have.

NMC Response:

The NRC comments have been resolved through adoption of the phrase, "every 24 effective full power months (EFPM) or one refueling outage (whichever is less)", as suggested by the NRC Staff in the October 25, 2006 Comment 7 and referencing the alternate repair criteria as suggested by the NRC Staff in the August 29, 2006 Comment 12. The reference to Specification 5.5.8.c.2(c) is correct.

5.5.8.d.3(d) (Not included in the current proposed TS)

August 29, 2006, Comment 10

In proposed TS 5.5.8.d.3(a), you indicate that the region of the tube below the F* and EF* regions may be excluded from the inspection requirements. In addition, in your response to question 4c in your July 21, 2006 letter (ML062370052), you indicate that full depth tubesheet sleeves are installed at the lower end of the parent tube (presumably this is near the tube-to-tubesheet weld). Since this latter region is below the F* and EF* region, it would appear that a tube in which a full depth tubesheet sleeve is installed may not require an inspection near the lower end of the sleeve (depending on exactly where the sleeve is installed with respect to the F* and EF* region). As a result, please discuss your plans to modify your proposal to ensure that full depth tubesheet sleeves require an inspection.

October 25, 2006, Comment 8

The frequency should be added to 5.5.8.d.3.d (i.e., every 24 EFPM or 1 RFO). The reference to sleeving is awkward (i.e., inspect 100% of the inservice tubes in the non-sleeved tubesheet region) since the tubesheet isn't sleeved. The easiest fix would be to delete non-sleeved. Alternatively wording such as the following should be considered, "For tubes with no portion of the sleeve within the [hot leg] tubesheet region, inspect 100% of the inservice tubes in the [hot-leg] tubesheet region, . . . when the F* or EF* methodology has been implemented."

November 9, 2006, Comment 5

It is not clear why they deleted 5.5.8.d.3.d. Our preference is that they retain it (as modified based on our previous comments).

NMC Response:

In response to the August 29, 2006 Comment 10, NMC included additional requirements in a draft proposed TS based on another plant's submittal. After further review, NMC realized that the additional requirements were beyond the Prairie Island Nuclear Generating Plant (PINGP) licensing basis. Current TS 5.5.8.b.3 states:

In addition to the sample required in Specification 5.5.8.b.2(a) through (c), all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of Specification 5.5.8.b.2(a).

These current TS requirements are embodied in the requirements of TS 5.5.8.d.3(a) proposed in this supplement and thus this paragraph is not included in this supplement.

<u>5.5.8.f.2(b), Page 5.0-28</u>

August 29, 2006, Comment 13

In proposed TS 5.5.8.f.2, you indicate that hardroll expanding portions of tubes in the tubesheet is an acceptable tube repair method. Since a tube may includes a sleeve, please discuss your plans to clarify that this repair criteria is only applicable to tubes that do not have sleeves installed in the tubesheet region. For example, "Hardroll expanding non-sleeved portions of tubes in the tubesheet in order to apply the F* and EF* criteria."

NMC Response:

This TS paragraph was revised to incorporate the wording as proposed by the NRC Staff.

<u>5.6.7.a.10, Page 5.0-40</u>

August 29, 2006, Comment 8

In several instances, the term "defect" is used in your proposed TS (e.g., 5.5.8.c.2(b)(1), proposed TS 5.5.8.c.2(b)(2), and proposed TS 5.6.7.a.10). Since a "defect" is not defined in your proposed TS, please discuss your plans to replace this term with "flaw" which is the term used in TSTF-449. In addition, the term "degradation" is used in your proposed TS (e.g., 5.5.8.c.2(c)(1) and 5.5.8.c.2(c)(2)). Since "degradation" is not defined in your proposed TS, please discuss your plans to replace this term with "flaw" which is the term used in TSTF-449.

November 9, Comment 6

In 5.6.7.a.10, the specification referenced should be 5.5.8.d (not 5.5.8.d.3(a)). Our suggestion would make it consistent with their current spec. Their proposal limits their current reporting requirement.

NMC Response:

This paragraph was revised to refer to "flaws" as proposed by the NRC Staff. Since F* and EF* only apply to Unit 2, the specification referenced is TS 5.5.8.d.3.

<u>**5.6.7.b, Page 5.0-40**</u>

August 29, 2006, Comment 12

In proposed TS 5.5.8.d.3(b) and (c), you refer to the repair criteria discussed in proposed TS 5.5.8.c.2(c) using different terminology. This can cause confusion on what is being referred to (since neither of these sections match the "title" in 5.5.8.c.2(c). As a result, please discuss your plans to modify these two sections to simply reference the "alternate repair criteria discussed in TS 5.5.8.c.2(c)." A similar comment applies to proposed TS 5.6.7.b.

October 25, 2006, Comment 9

On page 5.0-40, requirement "b...", it does not appear that "to tube support plate intersections" is needed. In fact, maybe more appropriate wording should be, "When the alternate repair criteria discussed in...... are implemented, notify....."

NMC Response:

This TS paragraph has been revised by specifically referencing "alternate repair criteria discussed in TS 5.5.8.c.2(c)" and deleting "to tube support plate intersections" as suggested by the NRC Staff.

<u>**5.6.7.b.4, Page 5.0-41**</u>

August 29, 2006, Comment 14

Regarding proposed TS 5.6.7.b.4, you indicated that removing this reporting requirement would constitute a change in your licensing basis (refer to your response to question 2 in the July 21, 2006 letter). The staff notes that by incorporating the 1x10-2 probability of burst criteria into TS 5.5.8.b.1, you will not be able to operate under the condition where the burst probability exceeds 10-2. As a result, providing a safety assessment is not needed. As a result, the reporting requirement is not needed. The staff also notes that you are required per 10 CFR 50.73 to report if the performance criteria are not maintained. As a result of the above, discuss your plans to remove the subject reporting requirement.

NMC Response:

Current TS requirement 5.6.7.5.e has been deleted as recommended by the NRC Staff.

Bases B 3.4.14, Page 3.4.14-2

August 29, 2006, Comment 3

Regarding TS 5.5.8.b.2, you reference the "voltage-based repair criteria." Since this reference isn't specific, it could be misinterpreted to apply to any flaws to which a voltage-based sizing method is applied. As a result, discuss your plans to clarify your proposed TS to indicate that the "voltage-based repair criteria" that you are referring to is the one in TS 5.5.8.c.2(c).

November 9, 2006, Comment 7

In the first paragraph on B 3.4.14-2, they do not include the last sentence that the TSTF indicated should be included. Namely, "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." Is there a reason for this?

NMC Response:

The first and last paragraphs on this Bases page were revised to reference TS 5.5.8.c.2(c) as suggested in the August 29, 2006 Comment 3. The TSTF-449 sentence was restored to this page in accordance with the November 9, 2006 Comment 7.

Bases B 3.4.19, Page B 3.4.19-2

August 29, 2006, Comment 1

In your proposed Structural Integrity Performance Criteria (SIPC) in Technical Specification (TS) 5.5.8.b.1, you stated the following: "For Unit 2, when tubes are left in service with predominantly axially oriented stress corrosion cracking at the tube support plate (TSP) elevations, the probability of burst (POB) under main steam line break conditions shall be maintained below 1E-02 in accordance with the requirements of NRC Generic Letter (GL) 95-05." As currently proposed, once tubes are left in service with predominantly axially oriented stress corrosion cracking at the tube support plate elevations, the probability of burst for all indications (even those that are not axially oriented stress corrosion cracking at TSP locations) is limited to 1x10-2. In addition, since NRC GL 95-05 does not contain any "requirements," the last portion of this statement is not accurate. If it

was not your intent to have the 1x10-2 criteria apply to all forms of degradation, please discuss your plans to modify your submittal.

Please discuss your plans to address the above. The proposed TS may be modified by using something similar to the following:

For Unit 2, when alternate repair criteria discussed in Specification 5.5.8.c.2(c) are applied to axially oriented outside diameter stress corrosion cracking indications at tube support plate locations, the probability that one or more of these indications in a SG will burst under postulated main steam line break conditions shall be less than 1x10-2.

Please note that your Bases may also need to be revised to clarify this issue.

August 29, 2006, Comment 3

Regarding TS 5.5.8.b.2, you reference the "voltage-based repair criteria." Since this reference isn't specific, it could be misinterpreted to apply to any flaws to which a voltage-based sizing method is applied. As a result, discuss your plans to clarify your proposed TS to indicate that the "voltage-based repair criteria" that you are referring to is the one in TS 5.5.8.c.2(c).

October 25, 2006, Comment 10

On page B 3.4.19-2, there appears to be a typo "thes" should be "these".

NMC Response:

The second paragraph of the Applicable Safety Analyses discussion was revised to specifically reference Specification 5.5.8.c.2(c) as suggested by the NRC Staff (August 29, 2006, Comment 3) and the typographical error was corrected. Discussion about axially oriented outside diameter stress corrosion cracking indications similar to that proposed by the NRC Staff (August 29, 2006, Comment 1) was also included in this paragraph.

Bases B 3.4.19, Page B 3.4.19-3

August 16, 2006, Comment 16

In the Limiting Condition for Operation section of B 3.4.19, you indicate that the F* and EF* distances are not considered part of the tube. Since these distances are no longer defined in your proposed TS, please discuss your plans to modify this phrase to indicate that the region of tube below the F* and EF* regions is not considered part of the tube. In addition, discuss your plans to indicate that the

parent tube (original tube wall) between sleeve joints is also not considered part of the tube.

October 25, 2006, Comment 11

On page B 3.4.19-3, the wording will need to be clarified since a sleeve installed below the F* and EF* region is still part of the tube (i.e., when a sleeve is installed, there is still an F*/EF* region - it's just no longer part of the pressure boundary).

November 9, 2006, Comment 8

In the 3rd paragraph in the LCO section on page B 3.4.19-3, they should remove the "(sleeves)" qualifier on repairs since it is not needed. In addition, it is not clear that the last sentence is complete. We would recommend something like the following: The tube-to-tubesheet weld is not considered part of the tube, nor is the region of the tube below the F* and EF* region (provided no sleeve extends below the F* and EF* region in which case the sleeve is part of the tube), nor the portion of the original tube wall between the sleeve joints.

NMC Response:

The third paragraph in the Limiting Condition for Operation discussion on this page was revised by removing "(sleeves)" which was included in a draft version. The other issues in the comments applicable to this paragraph were resolved by the addition of a parenthetical clause "(except as noted below)" and an additional sentence to which the NRC Staff agreed in a phone call on November 21, 2006.

ENCLOSURE 2

Proposed Technical Specification and Bases Pages (markup)

Prairie Island Nuclear Generating Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.0-22
3.4.14-2	5.0-23
3.4.14-3	5.0-24
3.4.19-1	5.0-25
3.4.19-2	5.0-26
5.0-13	5.0-27
5.0-14	5.0-28
5.0-15	5.0-36
5.0-16	5.0-37
5.0-17	5.0-38
5.0-18	5.0-39
5.0-19	5.0-40
5.0-20	5.0-41
5.0-21	

Bases pages

B 3.4.4-2	B 3.4.19-2
B 3.4.14-2	B 3.4.19-3
B 3.4.14-3	B 3.4.19-4
B 3.4.14-4	B 3.4.19-5
B 3.4.14-5	B 3.4.19-6
B 3.4.14-6	B 3.4.19-7
B 3.4.14-7	B 3.4.19-8
B 3.4.14-8	B 3.4.19-9
B 3.4.14-9	B 3.4.19-10
B 3.4.19-1	

46 pages follow

1.1 Definitions (continued)

$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
LEAKAGE	LEAKAGE from the Reactor Coolant System (RCS) shall be:

a. Identified LEAKAGE

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

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

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

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS identified LEAKAGE not within limit for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2.1 Reduce LEAKAGE to within limits.	14 hours
	OR	
	C.2.2 Be in MODE 5.	44 hours
D. Pressure boundary LEAKAGE exists. OR Primary to secondary ~~SG~~ LEAKAGE not within limit.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	------------------------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~~leakage~~ within limits by performance of RCS water inventory balance.	24 hours
SR 3.4.14.2	------------------------NOTE------------------------ Not required to be performed until 12 hours after establishment of steady state operation. --- Verify ~~steam generator tube integrity is in accordance with the Steam Generator Program~~primary to secondary LEAKAGE is < 150 gallons per day through any one SG..	~~In accordance with the Steam Generator Program~~ 72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 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 Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	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 MODE 4 following a SG tube inspection

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ 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. 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, except for flaws addressed through application of the alternate repair criteria discussed in Specification 5.5.8.c.2(c), 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. For Unit 2, when alternate repair criteria discussed in Specification 5.5.8.c.2(c) are applied to axially oriented outside diameter stress corrosion cracking indications at the tube support plate locations, the probability that one or more of these indications in an SG will burst under postulated main steam line break conditions shall be less than 1E-02.

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. For Unit 1, leakage is not to exceed 1 gpm per SG. For Unit 2, leakage from all sources, excluding the leakage attributed to the degradation associated with implementation of the voltage-based repair criteria discussed in Specification 5.5.8.c.2(c), is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria:

1. Unit 1 steam generator 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.

2 Unit 2 steam generator tubes found by inservice inspection to contain flaws shall be dispositioned as follows:

(a) Depth Based Criteria:

~~Steam generator tubes in each unit shall be determined OPERABLE by the following:~~

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

~~Each steam generator shall be determined OPERABLE in accordance with the in-service inspection schedule in Specification 5.5.8.c. The in-service inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single steam generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.~~

~~b. 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 Tables 5.5.8-1 and 5.5.8-2. The in-service inspection of steam generator tubes shall be performed at the Frequencies specified in Specification 5.5.8.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.d. The tubes selected for each in-service inspection shall include at least 3% of the total number of tubes in all steam generators and at least 20% of the total number of sleeves in service in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

1. ~~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.~~

2. ~~The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(a) all tubes that previously had detectable wall penetrations (> 20%) that have not been plugged or sleeve repaired in the affected area.~~

~~(b) tubes in those areas where experience has indicated potential problems.~~

~~(c) a tube inspection (pursuant to Specification 5.5.8.d.1.(h)) 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.~~

~~3. In addition to the sample required in Specification 5.5.8.b.2(a) through (c), all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of Specification 5.5.8.b.2(a).~~

~~4. The tubes selected as the second and third samples (if required by Tables 5.5.8-1 or 5.5.8-2) during each in-service inspection may be subjected to a partial tube or sleeve inspection provided:~~

~~(a) the tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.~~

~~(b) the inspections include those portions of the tubes or sleeves where imperfections were previously found.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~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.~~

~~5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.~~

~~6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~c. Inspection Frequencies~~

~~The above required in-service inspections of steam generator tubes shall be performed at the following Frequencies:~~

~~1. In-service 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 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.~~

~~2. If the results of the in-service inspection of a steam generator conducted in accordance with Table 5.5.8-1 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 5.5.8.c.1; the interval may then be extended to a maximum of once per 40 months.~~

~~3. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-1 during the shutdown subsequent to any of the following conditions:~~

~~(a) primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.~~

~~(b) a seismic occurrence greater than the Operating Basis Earthquake.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(c) a loss-of-coolant accident requiring actuation of the engineered safeguards.~~

~~(d) a main steam line or feedwater line break.~~

~~d. Acceptance Criteria~~

~~1. As used in this Specification:~~

~~(a) 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.~~

~~(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~(c) Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.~~

~~(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is~~

(1) Tubes found by inservice inspection containing a flaw in a non-sleeved region with a depth equal to or exceeding 50% of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate tube repair criteria discussed in Specification 5.5.8.c.2(b) or in Specification 5.5.8.c.2(c). If significant general tube thinning occurs, this ~~criteria will be~~ criterion is reduced to 40% wall penetration. ~~This definition does not apply to the portion of the tube in the tubesheet below the F* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance for F* or EF* tubes.~~

-(2) Tubes found by inservice inspection containing a flaw in~~The repair limit for~~ the pressure boundary region of any sleeve exceeding~~is~~ 25% of the nominal sleeve wall thickness shall be plugged. ~~This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.5.8.d.4 for the repair limit applicable to these intersections.~~

(3) Tubes with a flaw in a sleeve to tube joint that occurs in the original tube wall of the joint shall be plugged.

(b) The following F* or EF* Alternate Repair Criteria may be applied to the hot-leg of the tubesheet as an alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1):

(1) F* Criterion: If the bottom of the uppermost hardroll transition in the tubesheet is below the midplane of the tubesheet, then all flaws located below 1.07 inches from the bottom of this uppermost hardroll transition (not including eddy current uncertainty) may be allowed to remain in service provided the tube does not contain any flaws within this 1.07-inch span (not including eddy current uncertainty). This 1.07-inch span (increased for measurement uncertainty) is referred to as the F* region. If flaws are contained within the F* region, the tube shall be plugged or repaired.

(2) EF* Criterion: If the bottom of the uppermost hardroll transition in the tubesheet is above the midplane of the tubesheet but at least 2.0 inches below the top of the secondary face of the tubesheet, then all flaws located below 1.67 inches from the bottom of the uppermost hardroll transition (not including eddy current uncertainty) may be allowed to remain in service provided the tube does not contain any flaws within this 1.67-inch span (not including eddy current uncertainty). This 1.67-inch span (increased for measurement uncertainty) is referred to as the EF* region. If flaws are contained within the EF* region, the tube shall be plugged or repaired.

(c) The following Alternate Tube Support Plate Voltage-Based Repair Criteria may be applied as an alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1): For regions of the tube affected by predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of tube support plates the plugging or repair limit is as follows:

(1) If the bobbin voltage associated with the indication is less than or equal to 2.0 Volts, the indication is allowed to remain in service.

(2) If the bobbin voltage associated with the indication is greater than 2.0 Volts, the tube shall be plugged or repaired unless the voltage is less than or equal to the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) and a rotating pancake coil (or comparable examination technique) does

not detect a flaw. In this latter case, the indication may remain in service.

~~(g) 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.~~

~~(h) 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.~~

~~(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(j) F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.~~

~~(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.~~

~~(l) EF* Distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has~~

~~been conservatively determined to be 1.67 inches (not including eddy current uncertainty). EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.~~

~~(m) EF* Tube is a tube with degradation, below the EF* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the EF* distance.~~

~~2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* or EF* tubes) required by Tables 5.5.8-1 and 5.5.8-2.~~

~~3. Tube repair, after April 1, 1999, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:~~

~~CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:~~

~~(a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 Volts will be allowed to remain in service.~~

~~(b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 Volts, will be repaired or plugged, except as noted in Specification 5.5.8.d.4(c) below.~~

~~(c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 Volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes, with indications ofoutside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit, will be plugged or repaired.~~

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program (continued)

(3~~d)~~ If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.c.2(c)(1).~~d.4(a), (b)~~ and 5.5.8.c.2(c)(2) above~~(c)~~. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program (continued)

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.c.2(c)(1) ~~d.4(a), (b)~~ and 5.5.8.c.2(c)(2) above~~(c)~~.

Note: The upper voltage repair limit is calculated according to the methodology in ~~Generic Letter~~ GL 95-05 as supplemented.

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. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3 and d.4 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. For Unit 1 SGs, 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. For Unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

   (a). During each Unit 2 SG inspection (every 24 effective full power months (EFPM) or one refueling outage (whichever is less)), all tubes within that SG which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions do not need to be inspected, unless there is a sleeve (or portion of a sleeve) that extends below the F* or EF* region.

   (b). Implementation of the SG tube alternate repair criteria discussed in Specification 5.5.8.c.2(c) requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

(c). SG tube indications left in service as a result of application of the alternate repair criteria discussed in Specification 5.5.8.c.2(c) shall be inspected by bobbin coil probe every 24 EFPM or one refueling outage (whichever is less).

4. 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. There are no approved SG tube repair methods for the Unit 1 SGs.

2. For Unit 2, the following are approved repair methods:

(a). Alloy 690 tungsten inert gas welded sleeves in accordance with CEN-629-P, Revision 03-P,"Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

(b). Hardroll expanding non-sleeved portions of tubes in the tubesheet in order to apply the F* and EF* criteria.

~~Table 5.5.8-1~~
~~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 2S tubes in this S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Repair defective tubes and inspect additional 4S 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~~	~~Inspect all tubes in this S.G., Repair defective tubes and inspect 2S tubes in each other S.G.~~	~~All other S.G.s are C-1~~	~~None~~	~~N/A~~	~~N/A~~
		~~Prompt notification to NRC~~	~~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.~~	~~N/A~~	~~N/A~~

~~S=3%; When two steam generators are inspected during that outage.~~
~~S=6%; When one steam generator is inspected during that outage.~~

~~Table 5.5.8-2~~
~~STEAM GENERATOR TUBE SLEEVE INSPECTION~~

~~1st Sample Inspection~~			~~2nd Sample Inspection~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of 20% of Tube Sleeves (1)~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Inspect all remaining tube sleeves in this S.G. and plug or repair defective sleeved tubes.~~	~~C-1~~	~~None~~
			~~C-2~~	~~Plug or repair defective sleeved tubes~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~
	~~C-3~~	~~Inspect all tube sleeves in this S.G., inspect 20% of the tube sleeves in the other S.G., and plug or repair defective sleeved tubes~~	~~The other S.G. is C-1~~	~~None~~
			~~The other S.G. is C-2~~	~~Perform action for C-2 result of first sample~~
			~~The other S.G. is C-3~~	~~Inspect all tube sleeves in each S.G. and plug or repair defective sleeved tubes~~

~~(1) Each type of sleeve is considered a separate population for determination of scope expansion~~

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

~~1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.~~

~~2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports 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 or sleeved.~~

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

~~3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:~~

~~a. Identification of F* and EF* tubes, and~~

~~b. Location and extent of degradation.~~

a. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

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

5. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

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

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

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

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

10. The results of inspections performed under Specification 5.5.8.d.3 for all tubes that have flaws below the F* or EF* distance, and were not plugged. The report shall include: a) identification of F* and EF* tubes; and b) location and extent of degradation.

b..~~5~~. For implementation of the alternate repair criteria discussed in Specification 5.5.8.c.2(c) ~~voltage-based repair criteria to tube support plate intersections~~, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

~~a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.~~

1~~b~~. If circumferential crack-like indications are detected at the tube support plate intersections,~~.~~

2~~c~~. If indications are identified that extend beyond the confines of the tube support plate,~~.~~ or

3~~d~~. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.~~.~~

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

~~e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.~~

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

BASES

APPLICABLE SAFETY ANALYSES (continued)	forced flow rate, which is represented by the number of RCS loops in service. Both transient and steady state analyses include the effect of flow on the departure from nucleate boiling ratio (DNBR). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, and rod withdrawal events (Ref. 1). The plant is designed to operate with both RCS loops in operation to maintain DNBR within limits during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. RCS Loops - MODES 1 and 2 satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at power. An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~.
APPLICABILITY	In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes the total primary to secondary LEAKAGE is 1 gallon per minute from the faulted SG or is assumed to increase to 1 gallon per minute as a result of accident induced conditions plus 150 gallons per day from the intact SG. 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. When the alternate repair criteria discussed in Specification 5.5.8.c.2(c) are implemented for Unit 2 (only), the safety analysis assumes the leakage from the faulted SG is limited to 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).~~a 1 gpm primary to secondary LEAKAGE as the initial condition~~.

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

The USAR (Ref. 2) analysis for SGTR assumes the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop. Following the tube rupture, the initial primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential when compared to the mass transfer through the ruptured tube.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the total primary to secondary LEAKAGE is 1 gallon per minute from the faulted SG or is assumed to increase to 1 gallon per minute as a result of accident induced conditions plus 150 gallons per day from the intact SG. When the alternate repair criteria discussed in Specification 5.5.8.c.2(c) are 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. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere

LCO

c. Identified LEAKAGE (continued)

with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified leakage must be evaluated to assure that continued operation is safe. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

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

The limit of 150 gallons per day per ~~The 150 gallons per day (gpd) limit on one~~ SG is based on ~~implementation of the Steam Generator Voltage Based Alternate Repair Criteria and is more restrictive than standard operating leakage limits to provide additional margin to accommodate a crack which might grow at greater than the expected rate or unexpectedly extend outside the thickness of the tube support plate.~~the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 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.

BASES

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.

LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE in excess of the LCO limits must be identified or 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.1, B.2.1, and B.2.2

implemented for Unit 2 (only), the safety analysis assumes the leakage from the faulted SG for this repair method will be limited to 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F) ~~1 gpm (at 70°F) primary to secondary LEAKAGE in one generator as an initial condition~~. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 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).

BASES

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.

Seal welds are provided at the threaded joints of all reactor vessel head penetrations (spare penetrations, full-length Control Rod Drive Mechanisms, and thermocouple columns). Although these seals are part of the RCPB as defined in 10CFR50 Section 50.2, minor leakage past the seal weld is not a fault in the RCPB or a structural integrity concern. Pressure retaining components are differentiated from leakage barriers in the ASME Boiler and Pressure Vessel Code. In all cases, the joint strength is provided by the threads of the closure joint.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is

If unidentified LEAKAGE cannot be identified or 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, gaskets, and pressurizer safety valves seats is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours. If the LEAKAGE source cannot be identified within 54 hours, then the reactor must be placed in MODE 5 within 84 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

BASES

ACTIONS (continued)

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 MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

C.1, C.2.1, and C.2.2

If RCS identified LEAKAGE, other than pressure boundary LEAKAGE~~leakage~~ or primary to secondary LEAKAGE, is not within limits, then the reactor must be placed in MODE 3 within 6 hours. In this condition, 14 hours are allowed to reduce the identified leakage to within limits. If the identified LEAKAGE is not within limits within this time, the reactor must be placed in MODE 5 within 44 hours.

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

D.1 and D.2

If RCS pressure boundary LEAKAGE exists or if primary to secondary~~SG~~ LEAKAGE (150 gpd limit) is not within limits, the reactor must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

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. 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. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, equilibrium xenon, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. ~~Therefore, a~~ Note 1 states ~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For

RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by monitoring containment atmosphere radioactivity. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

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

SR 3.4.14.2

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions~~ This SR verifies that primary to secondary LEAKAGE is less 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 3.4.19, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. 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 (Ref. 4).

BASES

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR, Section 1.2.

2. USAR, Section 14.5.

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

4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Steam Generator (SG) Tube Integrity

BASES

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 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, 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 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met.

BASES

BACKGROUND (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. 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 (Ref. 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 LCO 3.4.14, "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 atmospheric steam dumps.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE of 1 gallon per minute from the faulted SG or is assumed to increase to 1 gallon per minute as a result of accident induced conditions plus 150 gallons per day from the intact SG. When the alternate repair criteria discussed in Specification 5.5.8.c.2(c) are implemented for Unit 2 (only), the safety analyses assume the leakage from the faulted SG is limited to 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F). When alternate repair criteria discussed in Specification 5.5.8.c.2(c) are applied to axially oriented outside diameter stress corrosion cracking indications, the probability that one or more of these indications in an SG will burst under postulated main steam line break conditions shall be less than 1E-02.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to or greater than the LCO 3.4.17, "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 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

LCO

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

During an 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, an SG tube is defined as the entire length of the tube, including the tube wall and any repairs made to it, 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, nor is the region of tube below the F* and EF* region (except as noted below), nor the portion of the tube between sleeve joints. When an F* or EF* region is repaired by sleeving, the entire sleeve is considered part of the tube.

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the

LCO (continued)

evaluation process for determining conformance with the SG performance criteria.

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation."

Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

BASES

LCO (continued)

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 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed those discussed in the APPLICABLE SAFETY ANALYSES section above. 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 3.4.14, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to an SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

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

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to

APPLICABILITY (continued)

secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

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

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.19.2. 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 an 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, Condition B applies.

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

BASES

ACTIONS A.1 and A.2 (continued)

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

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS SR 3.4.19.1

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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1 (continued)

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 SR 3.4.19.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.19.2

(continued)

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The tube repair criteria delineated in Specification 5.5.8 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 an 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.

BASES (continued)

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
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."

ENCLOSURE 3

Proposed Technical Specification Pages (revised)

Prairie Island Nuclear Generating Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.0-18
3.4.14-2	5.0-19
3.4.14-3	5.0-20
3.4.19-1	5.0-21
3.4.19-2	5.0-22
5.0-13	5.0-30
5.0-14	5.0-31
5.0-15	5.0-38
5.0-16	5.0-39
5.0-17	5.0-40

20 pages follow

1.1 Definitions (continued)

$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
LEAKAGE	LEAKAGE from the Reactor Coolant System (RCS) shall be:

a. Identified LEAKAGE

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

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

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

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS identified LEAKAGE not within limit for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2.1 Reduce LEAKAGE to within limits.	14 hours
	OR	
	C.2.2 Be in MODE 5.	44 hours
D. Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-------------------------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 within limits by performance of RCS water inventory balance.	24 hours
SR 3.4.14.2	-------------------------NOTE------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 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 Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.19.1 Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2 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 MODE 4 following an SG tube inspection

5.5 Programs and Manuals (continued)

5.5.8 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. 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, except for flaws addressed through application of the alternate repair criteria discussed in Specification 5.5.8.c.2(c), 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.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

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. For Unit 2, when alternate repair criteria discussed in Specification 5.5.8.c.2(c) are applied to axially oriented outside diameter stress corrosion cracking indications at the tube support plate locations, the probability that one or more of these indications in an SG will burst under postulated main steam line break conditions shall be less than 1E-02.

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. For Unit 1, leakage is not to exceed 1 gpm per SG. For Unit 2, leakage from all sources, excluding the leakage attributed to the degradation associated with implementation of the voltage-based repair criteria discussed in Specification 5.5.8.c.2(c), is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria:

1. Unit 1 steam generator 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.

2 Unit 2 steam generator tubes found by inservice inspection to contain flaws shall be dispositioned as follows:

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

(a) Depth Based Criteria:

(1) Tubes found by inservice inspection containing a flaw in a non-sleeved region with a depth equal to or exceeding 50% of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate tube repair criteria discussed in Specification 5.5.8.c.2(b) or in Specification 5.5.8.c.2(c). If significant general tube thinning occurs, this criterion is reduced to 40% wall penetration.

(2) Tubes found by inservice inspection containing a flaw in the pressure boundary region of any sleeve exceeding 25% of the nominal sleeve wall thickness shall be plugged.

(3) Tubes with a flaw in a sleeve to tube joint that occurs in the original tube wall of the joint shall be plugged.

(b) The following F* or EF* Alternate Repair Criteria may be applied to the hot-leg of the tubesheet as an alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1):

(1) F* Criterion: If the bottom of the uppermost hardroll transition in the tubesheet is below the midplane of the tubesheet, then all flaws located below 1.07 inches from the bottom of this uppermost hardroll transition (not including eddy current uncertainty) may be allowed to remain in service provided the tube does not contain any flaws within this 1.07-inch span (not including eddy current uncertainty). This 1.07-inch span (increased for measurement uncertainty) is referred to as the F* region. If flaws are contained within the F* region, the tube shall be plugged or repaired.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

(2) EF* Criterion: If the bottom of the uppermost hardroll transition in the tubesheet is above the midplane of the tubesheet but at least 2.0 inches below the top of the secondary face of the tubesheet, then all flaws located below 1.67 inches from the bottom of the uppermost hardroll transition (not including eddy current uncertainty) may be allowed to remain in service provided the tube does not contain any flaws within this 1.67-inch span (not including eddy current uncertainty). This 1.67-inch span (increased for measurement uncertainty) is referred to as the EF* region. If flaws are contained within the EF* region, the tube shall be plugged or repaired.

(c) The following Alternate Tube Support Plate Voltage-Based Repair Criteria may be applied as an alternative to the depth based criteria in Specification 5.5.8.c.2(a)(1): For regions of the tube affected by predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of tube support plates the plugging or repair limit is as follows:

(1) If the bobbin voltage associated with the indication is less than or equal to 2.0 Volts, the indication is allowed to remain in service.

(2) If the bobbin voltage associated with the indication is greater than 2.0 Volts, the tube shall be plugged or repaired unless the voltage is less than or equal to the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) and a rotating pancake coil (or comparable examination technique) does not detect a flaw. In this latter case, the indication may remain in service.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

3 If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.c.2(c)(1) and 5.5.8.c.2(c)(2) above. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.c.2(c)(1) and 5.5.8.c.2(c)(2) above.

Note: The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented.

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. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3 and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

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. For Unit 1 SGs, 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. For Unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

   (a) During each Unit 2 SG inspection (every 24 effective full power months (EFPM) or one refueling outage (whichever is less)), all tubes within that SG which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions do not need to be inspected, unless there is a sleeve (or portion of a sleeve) that extends below the F* or EF* region.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

(b) Implementation of the SG tube alternate repair criteria discussed in Specification 5.5.8.c.2(c) requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

(c) SG tube indications left in service as a result of application of the alternate repair criteria discussed in Specification 5.5.8.c.2(c) shall be inspected by bobbin coil probe every 24 EFPM or one refueling outage (whichever is less).

4. 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. There are no approved SG tube repair methods for the Unit 1 SGs.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. For Unit 2, the following are approved repair methods:

   (a) Alloy 690 tungsten inert gas welded sleeves in accordance with CEN-629-P, Revision 03-P,"Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

   (b) Hardroll expanding non-sleeved portions of tubes in the tubesheet in order to apply the F* and EF* criteria.

5.5 Programs and Manuals (continued)

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5.5 Programs and Manuals (continued)

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5.5 Programs and Manuals (continued)

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5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

a. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

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

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

5. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

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

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

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

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

10. The results of inspections performed under Specification 5.5.8.d.3 for all tubes that have flaws below the F* or EF* distance, and were not plugged. The report shall include: a) identification of F* and EF* tubes; and b) location and extent of degradation.

b. For implementation of the alternate repair criteria discussed in Specification 5.5.8.c.2(c), notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1. If circumferential crack-like indications are detected at the tube support plate intersections,

2. If indications are identified that extend beyond the confines of the tube support plate, or

3. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5.6 Reporting Requirements (continued)

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.