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July 25, 2007
5928-07-20168

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

Three Mile Island, Unit 1 (TMI Unit 1)
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Response To Request For Additional Information – Technical Specification Change Request No. 331: Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TAC No. MD1807)

References:

1) USNRC letter to AmerGen Energy Company, LLC dated July 20, 2007, "Request for Additional Information Regarding Proposed Steam Generator Tube Integrity Technical Specification Changes (TAC No. MD1807)."

2) AmerGen Energy Company, LLC letter to NRC dated May 15, 2006 (5928-06-20390), "Technical Specification Change Request No. 331 – Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity."

3) AmerGen Energy Company, LLC letter to NRC dated May 31, 2007 (5928-07-20072), "Response To Request For Additional Information – Technical Specification Change Request No. 331: Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TAC No. MD1807)."

This letter provides additional information in response to the NRC request for additional information (RAI), dated July 20, 2007 (Reference 1), regarding TMI Unit 1 Technical Specification Change Request No. 331, submitted to NRC for review on May 15, 2006 (Reference 2). The additional information is provided in Enclosure 1.

As described in the Enclosure 1 responses, the proposed Technical Specification (TS) page markups have been revised from our submittal of May 31, 2007 (Reference 3) to incorporate additional clarifications, consistent with the NRC approved TSTF-449, Revision 4. Additionally, the proposed TS 4.19 surveillance requirement statement is revised to substitute the phrase

A122
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"have tube integrity" for the term "OPERABLE" and thus states "Each steam generator shall be determined to have tube integrity by performance of the following:". This change provides consistency with the corresponding wording in TSTF-449, Rev. 4, TS Section 3.4.17 (BWOG), which refers to steam generator tube integrity in lieu of steam generator operability. Reference 1 also provided NRC staff observations for consideration regarding the TMI Unit 1 proposed TS

Bases page markups incorporating TSTF-449, Revision 4 Bases changes. These observations have been evaluated and incorporated, as applicable, into the revised proposed TS page markups provided in Enclosure 2.

These changes have no impact on the conclusions of the original safety analysis or no significant hazards consideration evaluation provided in Reference 2. The revised proposed Technical Specification pages are provided in Enclosure 2. Enclosure 2 provides a complete replacement set of the proposed Technical Specification pages previously submitted in Reference 3.

We suggest that a meeting be scheduled to facilitate resolution if any significant open issues remain regarding TMI Unit 1 implementation of TSTF-449, Rev. 4.

Regulatory commitments established by this submittal are identified in Enclosure 3. If any additional information is needed, please contact David J. Distel at (610) 765-5517.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of July, 2007.

Sincerely,

Russell G. West
Vice President TMI, Unit 1

Enclosures:
1) Response to Request for Additional Information
2) Revised TS Page Markups
3) List of Commitments

cc: S. J. Collins, USNRC Administrator, Region I
P. J. Bamford, USNRC Project Manager, TMI Unit 1
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
File No. 06007

ENCLOSURE 1

TMI UNIT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION CHANGE REQUEST No. 331
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
TMI UNIT 1 TECHNICAL SPECIFICATION CHANGE REQUEST No. 331
APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT
REGARDING STEAM GENERATOR TUBE INTEGRITY

1. NRC Question

In your letter dated May 31, 2007, you provided references to support your assumption that you only need to determine the amount of leakage during a main steam line break (MSLB) and compare this to the assumptions in your accident analyses. In reviewing these references, the staff did make some conclusions regarding the leakage from certain types of flaws from certain locations (e.g., the leak rate determined under MSLB conditions for flaws within the kinetic expansion region bounds the actual leakage which would occur under feed line break conditions provided the methodology discussed in the staff's safety evaluation is used); however, the staff did not see any reference to a conclusion that only the leakage under a MSLB condition needs to be assessed and compared to its limit. Assuming the MSLB accident always results in the most leakage (for all flaw types and all flaw locations), it would be conservative to calculate the leakage for only a MSLB accident and to compare the calculated leakage to the amount of primary-to-secondary leakage assumed in each of the design basis accidents (DBAs) (to ensure the facility is operated within its design and licensing basis). Alternatively, other methodologies could be developed for assessing the amount of leakage during each of the other DBAs. In light of the above, either (1) indicate that you will confirm (e.g., as part of the requirement to ensure tube integrity) that the amount of leakage for each DBA will be within the limits assumed in the accident analyses or (2) demonstrate the acceptability of just determining the leak rate under MSLB conditions and comparing this value to the value assumed in the MSLB accident analyses by demonstrating (a) that this leak rate is conservative for all other accidents and all flaw types and (b) by demonstrating that comparing this calculated leakage (rate or volume) to the value of leakage assumed in the MSLB accident analyses is the most limiting scenario (for all DBAs) from a dose assessment standpoint.

Response

This response addresses Option (2) above. While numerous hypothetical plant transients and design parameters have been assessed, determination of hypothetical steam generator tube flaw leakage has been conservatively based solely on the Main Steam Line Break (MSLB) event. This has been the TMI Unit 1 licensing basis. For example, the following is an excerpt from ECR TM 01-00328 (License Amendment No. 204, dated October 2, 1997): (This document is currently referenced in the TMI Unit 1 Technical Specifications, and is currently referenced in the proposed Technical Specifications.)

> "...The limiting primary-to-secondary pressure differential associated with accident conditions is the safety relief valve setpoint of 2,575 psi. This condition is associated with a MSLB condition and includes a 3% allowance for setpoint tolerance..." (Page 12)

> "...To provide a reasonable assurance that the leakrate will not be exceeded, MSLB primary-to-secondary leakrates must be determined......The MSLB conditions assumed in this analysis were a primary-to-secondary pressure differential of 2575 psi, and the tube and primary fluid temperatures were assumed to be 600F..."

In summary, hypothetical leakage rates to disposition indications for acceptability to remain in service in the TMI Unit 1 steam generators under ECR TM 01-00328 were conservatively based on leakrates calculated using this assumed MSLB-induced differential pressure.

The MSLB event has also been utilized to bound tube tensile loads, which open circumferentially-oriented flaw extents and consequently increase hypothetical leakrates (as well as increase potential for a tube rupture). In order to disposition flaws in the kinetic expansions under ECR TM 02-01121, leakrates were calculated using MSLB-induced thermal-hydraulic conditions with MSLB-induced tensile loads applied. This was consistent with the licensing basis for the original design of the kinetic expansions. For example, GPUN Topical Report-007 (1983) stated that, "...The controlling accident is a main steam line break (MSLB)" for tube shrinkage, which creates the tensile loads. The NRC SER for the kinetic expansion tube repairs (NUREG-1019, Page 21) states, "...The licensee performed additional calculations to determine the maximum crack size that would remain stable under loads experienced during a main steam line break (MSLB) accident... Results show that leakage rate increases with tube axial loads and is detectable under conditions that might cause tube failures...."

The TMI Unit 1 current licensing basis to date is to determine the postulated leakrate under MSLB conditions, and that the calculated MSLB-induced leakrates are conservative for all flaw types and locations. The TMI Unit 1 calculated hypothetical MSLB accident-induced dose consequences, described in the Updated Final Safety Analysis Report (UFSAR) and approved by the NRC, are the most limiting for steam generator tube accident-induced leakage described in the UFSAR.

The TMI Unit 1 design basis accidents were examined for potential accident induced leakage to ensure compliance with dose criteria in 10 CFR 50.67 and 10CFR 100. Accidents identified that result in a dose consequence and consider primary to secondary leakage include the Maximum Hypothetical Accident (MHA), LOCA, Rod Ejection Accident, Steam Generator Tube Rupture (SGTR), and Main Steam Line Break (MSLB). The MHA and the LOCA are both accidents with an assumed gross release of radioisotopes to the reactor building. Since all reactor coolant is discharged into containment and primary system pressure is lost, no tube primary-to-secondary leakage is assumed. In the case of the SGTR, if a tube were to degrade to the point of rupture, such an accident is analyzed and radiological consequences are acceptable.

The Rod Ejection Accident assumes a primary-to-secondary leakage rate of 1 gpm until the system is depressurized. It was calculated that less than 5 gallons of reactor coolant is released to the secondary system during the time that the system is depressurizing. The MSLB accident analysis assumes a leakage of 3228 gallons for the first 2 hours of the accident and 9960 gallons over the duration of the accident. The MSLB analysis also assumes an additional 1 gpm leak from the unaffected steam generator. Both the rod ejection accident analysis and the MSLB accident bound the operational leakage limit of 0.1 gpm. When examining the total consequences of the more dose limiting accident, the

MSLB, the total dose thyroid and whole body dose is less than 10% of the 10 CFR 100 dose limits (28 Rem thyroid and <1 Rem whole body at the EAB, 5.3 Rem thyroid and <1 Rem whole body at the LPZ, and 29 Rem thyroid and < 1 Rem whole body in the control room) in all analyzed locations. The Rod Ejection Accident results in even less dose, approximately 2% of the 10 CFR 100 dose limits (5.2 Rem thyroid and 0.007 Rem whole body at the EAB, and 9.72 Rem thyroid and 0.009 Rem whole body at the LPZ). Therefore, when examining the effects of additional tube stresses to dose consequences the MSLB accident is bounding to other accidents that consider primary-to-secondary leakage.

2. **NRC Question**

You provided references to support your position that you do not need to propose inspection and repair criteria for the parent tube behind the upper sleeve joints. The staff reviewed these references and they do not appear to support your conclusion that the staff accepted this position when we reviewed the engineering report addressing the criteria for kinetic expansion indications. As a result, discuss your plans for performing inspections of the upper sleeve joint area (including the parent tube) to confirm that you are operating the sleeve in accordance with its original design criteria (in terms of the state of the parent tube). In addition, provide the technical basis for this design criteria.

Response

The proposed TMI Unit 1 Technical Specifications have been revised to include inspections of the parent tube in the upper tubesheet sleeve roll joint area to determine if parent tube degradation is occurring.

When sleeve upper tubesheet roll joints are examined, the associated parent tube will be examined using +Point MRPC probes using EPRI qualified techniques. These techniques are qualified for the detection of OD IGA and OD IGA/SCC in the parent tube. The acceptance criteria for the parent tube will be based on a comparison of the current outage's data and the earliest available historical +Point data for that tube. The qualitative comparison will address the following attributes of the parent tubing behind sleeve upper roll expansions:

1) Determine that the parent tubing is still present.
2) Determine if there is any change in the number of indications present.
3) Verify there is no change in the orientation/morphology of the indications
4) Verify there is no significant change in the circumferential extents of the circumferential and volumetric flaws.
5) Verify there is no significant change in the axial extents of the axial and volumetric flaws.

If the comparison of the data indicates ongoing degradation, that sleeved tube will be removed from service. This inspection scope and criteria has been added to proposed TS Section 6.19.c.3

These examinations, and these criteria, will provide assurance that the parent tubing behind the upper tubesheet sleeve joints is not being subjected to additional degradation,

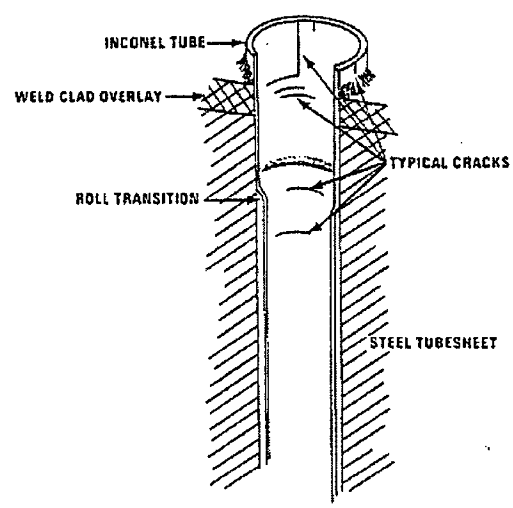
and assures that parent tubing subjected to additional degradation is removed from service.

The technical basis for these examination criteria is that the kinetically-expanded parent tubing behind the upper sleeve roll joints was known to contain numerous flaws when the sleeves were installed in 1991 and 1993. This parent tubing was successfully removed from service by the installation of the kinetic expansions in the early 1980's, and then subsequently "covered" by the installation of sleeves in 1991 and 1993. To illustrate the typical degradation at the parent tubing behind the upper sleeve rolled joints, the following image is reprinted from Topical Report 008, Rev. 3, "Assessment of TMI-1 Plant Safety for Return to Service After Steam Generator Repair", 1983. Topical Report 008 was one of the bases documents for the NRC's "TMI-1 Steam Generator Repair Safety Evaluation Report" (NUREG-1019). (Note that, as part of the kinetic expansion repairs, the tube ends above the tube-to-tubesheet welds were removed.)

TR 008 Rev.
p11

FIGURE 1-5

TMI-1 Steam Generator
Typical Cracks



CRACK CHARACTERISTICS: CIRCUMFERENTIAL BELOW FILET WELD
NOT FULL ARC
GENERALLY VERY TIGHT
PRIMARY SIDE INITIATED

This parent tubing behind the sleeve upper rolls is very flaw tolerant since it has been removed from service, is located at the outboard end of the 17" or 22" long kinetic tube-to-tubesheet expansion repairs, and is completely captured within the confines of the upper tubesheets. This parent tubing location also is isolated from side loads, axial loads, potential loose parts, and flow-induced vibrations, and is covered with the expanded Alloy 690 sleeve material.

Since the parent tubing had known defects, prior to sleeve installation testing was performed to evaluate the potential impact of parent tube defects on sleeve leakage and structural integrity. This work was described in the 10CFR50.59 evaluation for the sleeve installation; it was concluded that parent tube defects were not detrimental to sleeve integrity.

During sleeve installations, sleeve expansion parameters were monitored to verify the proper diametral expansion of the sleeves against the parent tubing. Process controls were present to verify that the parent tubing was present during the sleeve installations.

The sleeve installations, at TMI Unit 1 and the other OTSG PWR plants, have been successful. Thousands of sleeves were installed and they were not sources of operational leakage. Leaker outages, which plagued the OTSG plants as a result of tube fatigue flaws at the secondary faces of the upper tubesheets, were mitigated. The TMI Unit 1 sleeves have provided approximately 16 years of excellent service to date, and are scheduled for removal from service with the planned replacement of the TMI Unit 1 OTSGs in Fall 2009. The proposed inspection requirements provide reasonable assurance that the parent tube behind the sleeve upper tubesheet roll joint is not subject to ongoing degradation, and is being retained consistent with the original sleeve design and installation basis.

3. **NRC Question**

In your May 31, 2007, letter you indicated that "No degradation meeting the proposed sleeve inspection repair criteria was identified in the sleeve or the adjacent parent tube above the lower sleeve end." *Given that indications existed in some of the parent tubes* prior to sleeve installation, please discuss why this degradation has not been found. In addition, confirm that inspection methods and probes capable of detecting all flaw types are being utilized to inspect the parent tube (consistent with your proposed request).

Response

A small number of the sleeved tubes had small IDIGA flaws behind the sleeves, prior to sleeve installation. These flaws have been removed from service by the sleeves. These areas of IDIGA are often difficult to detect by bobbin coil exams in the non-sleeved tubes and are more difficult to detect in the sleeved sections of the tubes. (Refer, for example, to ECR TM 01-00328, where TMI Unit 1 "first-detected by MRPC" Volumetric IDIGA flaws are described.)

Most of the lengths of the sleeves are reexamined using bobbin probes to ensure that the sleeves retain their integrity. There is a gap between the parent tube and the majority of the installed sleeve axial length. This gap prevents reliable probability of detection (POD) of flaws in the original parent tubing. A reliable POD for such flaws is not necessary since these lengths of parent tubing have been removed from service.

Refer also to the response to Question 2, above, regarding inspections of parent tubing behind the upper tubesheet sleeve roll joints.

The primary purpose of the sleeve examinations is to assess whether sleeve integrity is being retained. The proposed TS change request intends that reliable inspection methods be utilized at locations where the sleeves and parent tubing form the pressure boundary. It

is not the intent of the proposed specification that optimal inspection methods be utilized at locations where the sleeve or parent tubing does not serve as pressure boundary. The design of the sleeves considered the potential of a complete severance of the parent tube.

4. **NRC Question**

In letter dated May 31, 2007, you deleted TS Section 6.9.6.i. The staff is aware that TMI-1 does not have approved repair methods, however, the number and percentage of inservice tubes repaired by each method existing in the SGs should be reported. Please discuss your plans to include this as a reporting requirement in TS Section 6.9.6.

Response

Proposed TS Section 6.9.6 is revised to add the following additional reporting requirement: "The number and percentage of inservice tubes repaired by each method existing in the SGs."

5. **NRC Question**

In Table 4.1-2, the test frequency for primary-to-secondary leakage was relaxed from every 24 hours to every 72 hours. Please provide justification for this test frequency relaxation, or remove this relaxation to the test frequency from your proposal.

Response

The TSTF-449, Rev. 4 Bases INSERT B.3.4.13D (BWOG) provides the justification for the surveillance frequency and states that 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 proposed TMI Unit 1 TS changes adopting the STS surveillance frequency of 72 hours for primary-to-secondary leakage in TMI Unit 1 TS Table 4.1-2, and the associated BWOG STS Bases, are consistent with TSTF-449, Rev. 4, TS Surveillance Requirement 3.4.13.2 (BWOG STS).

6. **NRC Question**

Regarding Table 4.1-2, a note is needed (consistent with TSTF-449) to Item 7 (Reactor Coolant System Leakage) that this requirement is not applicable to primary-to-secondary leakage.

Response

Proposed TS Table 4.1-2 is revised to add the following note to Item 7, "Reactor Coolant System Leakage": "(Not applicable to primary-to-secondary leakage.)"

7. **NRC Question**

The wording of proposed TS Sections 6.19.b.2 and 6.19.c.1 as it pertains to the exceptions of the 1 gpm accident-induced leakage performance criteria does not appear to reflect your assumptions in your accident analysis. The 1 gpm limit is a risk-informed limit. In the case of TMI-1, the NRC staff has approved an exception to the 1 gpm limit for the leakage associated with indications attributed to flaws (or postulated flaws) within the kinetic expansion region (excluding sleeved areas); however, the leakage from inside diameter intergranular attack (ID IGA) indications must be compared to the 1 gpm limit. As currently proposed, the leakage from ID IGA indications could be just below the 1 gpm limit with no consideration given to other leakage sources (plugs, sleeves, other flaws). As a result, please discuss your plans to modify the proposed accident induced leakage performance criteria contained within TS Section 6.19.b.2 and 6.19.c to accurately reflect the exception to the risk informed 1 gpm limit. For example the second sentence of TS Section 6.19.b.2 could be modified to state, "Leakage from all sources excluding the leakage attributed to the degradation described in TS Section 6.19.c.2 is also not to exceed 1 gpm per SG." The corresponding change in TS Section 6.19.c.1.a would be to delete the sentence containing reference to 1 gpm and the corresponding change in TS Section 6.19.c.1.b would be to delete the sentence regarding MSLB accident induced leakage. Additionally, to be more representative of your design and licensing basis, discuss your plans to modify the first sentence of TS Section 6.19.b.2 to read as follows,

"... The primary to secondary accident induced leakage volume or rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate or volume in the accident analysis in terms of total leakage rate or volume of leakage for all SGs and leakage rate or volume for an individual SG."

Response

Proposed TS Sections 6.19.b.2, 6.19.c.1.a, and 6.19.c.1.b are revised as described in the above question, except that the applicable TS Section reference in the change to TS Section 6.19.b.2 is to TS Section 6.19.c.1.b, in lieu of 6.19.c.2 as stated in the above question.

8. **NRC Question**

In your May 31, 2007, letter you indicated that [Engineering Report] ECR NO. TM 02-01121 is required by the TSTF since it provides examination and flaw dispositioning criteria for kinetically expanded tubing in the upper tubesheets. Given the information provided by ECR NO. TM 02-01121 does not clarify the repair criteria for sleeves as stated in TS Section 6.19.c.2, discuss your plans to remove reference to this document in TS Section 6.19.c.2.

Response

Proposed TS Section 6.19.c.2 is revised to remove the reference to ECR No. TM 02-01121, Rev. 2.

9. **NRC Question**

In proposed TS Section 6.19.d.4, you indicated that "Implementation of the repair criteria for ID IGA requires 100% bobbin coil inspection of all non-plugged tubes in accordance with AmerGen Engineering Report, ECR NO. TM 01-00328 during all subsequent SG inspections." While the staff agrees that the eddy current testing methods and probes utilized should be in accordance with the referenced engineering report, there is a concern regarding the inspection sample. The engineering report requires that effort should be taken to schedule 100% bobbin coil inspection for ID IGA repair criteria implementation. Since this wording does not specify what inspection will be performed, please discuss your plans to modify this TS Section to further clarify reference to the engineering report. For example, "Implementation of the repair criteria for ID IGA requires 100 percent bobbin coil inspection of all non-plugged tubes using inspection methods and probes in accordance with ECR No. TM 01-00328." A similar comment applies for proposed TS Section 6.19.d.5.

Response

Proposed TS Sections 6.19.d.4 and 6.19.d.5 are revised as described in the above question.

10. **NRC Question**

In proposed TS Section 6.19.d, the purpose for the clarification that the portion of the original tube wall above the sleeve's lower sleeve-to-tube rolled joints is not an area requiring re-inspection is not clear in light of the proposal in TS Section 6.19.c.2 to repair flaws in the parent tube between the lower sleeve end and the parent tube kinetic expansion. Please clarify the requirements. If the exception to the inspection requirements is retained, please discuss your plans to clarify that the inspection is exempted from the top of the middle sleeve roll to the bottom of the upper most sleeve roll (or as appropriate per Item 2 above).

Response

Based on the response to Item 2 above, proposed TS Section 6.19.d is revised to state the following: "In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's sleeve-to-tube rolled joints is not an area requiring re-inspection since the inspection is exempted from the top of the middle sleeve roll to the bottom of the uppermost sleeve roll (upper tubesheet roll)."

ENCLOSURE 2

TMI Unit 1 Technical Specification Change Request No. 331

Revised Markup of Proposed License, Technical Specifications, and Bases Page Changes

Revised License Pages

6
7

Revised Technical Specifications & Bases Pages

Table of Contents Page iv
Table of Contents Page v
Table of Contents Page vi
3-1a
3-2
3-12
3-15a
3-26c
4-2b
4-8
4-77
4-78
4-79
4-80
4-81
4-82
4-83
4-83a
4-84
4-85
6-19
6-26

(8) Repaired Steam Generators — DELETED

In order to confirm the leak-tight integrity of the Reactor Coolant System, including the steam generators, operation of the facility shall be in accordance with the following:

1. Prior to initial criticality, the licensee shall submit to NRC the results of the steam generator hot test program and a summary of its management review.

2. The licensee shall confirm baseline primary-to-secondary leakage rate established during the steam generator hot test program. If leakage exceeds the baseline leakage rate by more than 0.1 gpm*, the facility shall be shut down and leak tested. If any increased leakage above baseline is due to defects in the tube free span, the leaking tube(s) shall be removed from service. The baseline leakage shall be re-established, provided that the leakage limit of Technical Specification 3.1.6.3 is not exceeded.

3. The licensee shall complete its post-critical test program at each power range (0-5%, 5%-50%, 50%-100%) in conformance with the program described in Topical Report 008, Rev. 3, and shall have available the results of that test program and a summary of its management review, prior to ascension from each power range and prior to normal power operation.

4. The licensee shall conduct eddy-current examinations, consistent with the extended inservice inspection plan defined in Table 3.3-1 of NUREG-1019, either 90 calendar days after reaching full power, or 120 calendar days after exceeding 50% power operation, whichever comes first. In the event of plant operation for an extended period at less than 50% power, the licensee shall provide an assessment at the end of 180 days of operation at power levels between 5% and 50%, such assessment to contain recommendations and supporting information as to the necessity of a special eddy-current testing (ECT) shutdown before the end of the refueling cycle. (The NRC staff will evaluate that assessment and determine the time of the next eddy-current examination, consistent with the other provisions of the license conditions.) In the absence of such an assessment, a special ECT shutdown shall take place before an additional 30 days of operation at power above 5%.

*If leakage exceeds the baseline leakage rate by more than 0.1 gpm during the remainder of the Cycle 8 operation, the facility shall be shutdown and leak tested. Operation at leakage rates of up to 0.2 gpm above the baseline leakage rate shall be acceptable during the remainder of Cycle 8 operation. After the 9R refueling outage, the leakage limit and accompanying shutdown requirements revert to 0.1 gpm above the baseline leakage rate.

5. ~~The licensee~~ shall provide routine reporting of the ~~long-term~~ corrosion "lead tests" test results ~~on a quarterly~~ basis as well as more timely ~~notification~~ if adverse corrosion test results are ~~discovered.~~

(9) Long Range Planning Program - Deleted

Sale and License Transfer Conditions

(10) Deleted

(11) Deleted

(12) Deleted

(13) Deleted

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TABLE OF CONTENTS

Amendment No. ~~11, 22, 30, 41, 47, 55, 72, 78, 95, 97, 119, 122, 129, 137, 146, 197, 242, 245, 246, 248~~,

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TABLE OF CONTENTS

Amendment No. ~~11, 47, 72, 77, 128, 150, 173, 212, 252, 263, 258~~,

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LIST OF TABLES

Amendment No. ~~59, 72, 100, 106, 118, 137, 142, 147, 150, 173, 197, 216~~,

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3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator (SG) Tube Integrity

INSERT

a. ~~Both steam generators shall be operable whenever the reactor coolant average temperature is above 250°F.~~

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ± 1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Amendment No. 12, 17, 28, 47, 98,

INSERT TO TS PAGE 3-1a (REVISED TS 3.1.1.2)

a. Whenever the reactor coolant average temperature is above 200°F, the following conditions are required:

(1.) SG tube integrity shall be maintained.

AND

(2.) All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program. (The Steam Generator Program is described in Section 6.19.)

ACTIONS:

---NOTE---

Entry into Sections 3.1.1.2.a.(3.) and (4.), below, is allowed for each SG tube.

(3.) If the requirements of Section 3.1.1.2.a.(2.) are not met for one or more tubes then perform the following:

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

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

b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to exceeding a reactor coolant average temperature of 200°F following the next refueling outage or SG tube inspection.

(4.) If Action 3., above, is not completed within the specified completion times, or SG tube integrity is not maintained, be in HOT SHUTDOWN within 6 hours and be in COLD SHUTDOWN within 36 hours.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature (References 1, 2, and 3).

Both steam generators must ~~be operable~~ have tube integrity before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes. Refer to Section 3.1.6.3 for allowable primary-to-secondary leakage. Refer to Section 4.19 for Bases for Steam Generator tube integrity.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents (Reference 4). The pressurizer code safety valve lift set point shall be set at 2500 psig ±1% allowance for error. **Surveillance requirements are specified in the Inservice Testing Program. Pressurizer code safety valve setpoint drift of up to 3% is acceptable in accordance with ASME Section XI** (Reference 5) **and the assumptions of TMI-1 safety analysis.**

References

(1) UFSAR, Tables 9.5-1 and 9.5-2

(2) UFSAR, Sections 4.2.5.1 and 9.5 – "Decay Heat Removal"

(3) UFSAR, Section 4.2.5.4 – "Secondary System"

(4) UFSAR, Section 4.3.10.4 – "System Minimum Operational Components"

(5) UFSAR, Section 4.3.7 – "Overpressure Protection"

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3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system.

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If the sum of the primary-to-secondary leakage from both steam generators ~~through the steam generator tubes~~ exceeds ~~1 gpm total for both steam generators,~~ 0.1 gpm (144 GPD), the reactor shall be placed in hot shutdown within 6 hours, and the reactor shall be placed in cold shutdown within 36 hours. ~~of detection.~~

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the dose rate limits of the ODCM.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmosphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

Bases (Continued)

The unidentified reactor coolant leakage limit of 1 gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a matter of hours, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Total reactor coolant leakage is limited by this specification to 10 gpm. This limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage.

~~The sum of the primary-to-secondary leakage through the from both steam generators tubes is limited to 1 gpm total. 0.1 gpm (144 GPD). This limit ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of a steam line break. Steam generator primary-to-secondary leakage is quantified by analysis of secondary plant activity.~~

If reactor coolant leakage is to the auxiliary building, it may be identified by one or more of the following methods:

a. The auxiliary and fuel handling building vent radioactive gas monitor is sensitive to very low activity levels and would show an increase in activity level shortly after a reactor coolant leak developed within the auxiliary building.

b. *Water inventories around the auxiliary building sump.*

c. Periodic equipment inspections.

d. In the event of gross leakage, in excess of 13 gpm, the individual cubicle leak detectors in the makeup and decay heat pump cubicles, will alarm in the control room to backup "a", "b", and "c" above.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by TMI-1 Plant Operations.

INSERT

REFERENCES

(1) NEI 97-06, "Steam Generator Program Guidelines."

INSERT TO TS PAGE 3-15a (BASES FOR SECTION 3.1.6)

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 a leakage volume or rate of primary to secondary leakage from all steam generators (SGs) depending on the specific accident analyses. The leakage rate may increase (over that observed during normal operation) as a result of accident-induced conditions. The TS requirement to limit the sum of the primary to secondary leakage from both SGs to less than or equal to 144 gallons per day is significantly less than the conditions assumed in the safety analysis.

The limit on the sum of the primary to secondary leakage from both SGs of 144 gallons per day is less than the TSTF-449, Rev. 4 limit of 150 gallons per day per SG, which is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 1). 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 (Continued)

If EFW were required during surveillance testing, minor operator action (e.g., opening a local isolation valve or manipulating a control switch from the control room) may be needed to restore operability of the required pumps or flowpaths. An exception to permit more than one EFW Pump or both EFW flowpaths to a single OTSG to be inoperable for up to 8 hours during surveillance testing requires 1) at least one motor-driven EFW Pump operable, and 2) an individual involved in the task of testing the EFW System must be in communication with the control room and stationed in the immediate vicinity of the affected EFW flowpath valves. Thus the individual is permitted to be involved in the test activities by taking test data and his movement is restricted to the area of the EFW Pump and valve rooms where the testing is being conducted.

The allowed action times are reasonable, based on operating experience, to reach the required plant operating conditions from full power in an orderly manner and without challenging plant systems. Without at least two EFW Pumps and one EFW flowpath to each OTSG operable, the required action is to immediately restore EFW components to operable status, and all actions requiring shutdown or changes in Reactor Operating Condition are suspended. With less than two EFW pumps or no flowpath to either OTSG operable, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, which might result in a trip. The seriousness of this condition requires that action be started immediately to restore EFW components to operable status. TS 3.0.1 is not applicable, as it could force the unit into a less safe condition.

The EFW system actuates on: 1) loss of all four Reactor Coolant Pumps, 2) loss of both Main Feedwater Pumps, 3) low OTSG water level, or 4) high Reactor Building pressure. A single active failure in the HSPS will neither inadvertently initiate the EFW system nor isolate the Main Feedwater system. OTSG water level is controlled automatically by the HSPS system or can be controlled manually, if necessary.

The MSSVs will be able to relieve to atmosphere the total steam flow if necessary. Below 5% power, only a minimum number of MSSVs need to be operable as stated in Specifications 3.4.1.2.1 and 3.4.1.2.2. This is to provide OTSG overpressure protection during hot functional testing and low power physics testing. Additionally, when the Reactor is between hot shutdown and 5% full power operation, the overpower trip setpoint in the RPS shall be set to less than 5% as is specified in Specification 3.4.1.2.2. The minimum number of MSSVs required to be operable allows margin for testing without jeopardizing plant safety. Plant specific analysis shows that one MSSV is sufficient to relieve reactor coolant pump heat and stored energy when the reactor has been subcritical by 1% delta K/K for at least one hour. Other plant analyses show that two (2) MSSVs on either OTSG are more than sufficient to relieve reactor coolant pump heat and stored energy when the reactor is below 5% full power operation but had been subcritical by 1% delta K/K for at least one hour subsequent to power operation above 5% full power. According to Specification 3.1.1.2a, both OTSGs shall ~~be operable~~ have tube integrity whenever the reactor coolant average temperature is above ~~250~~ 200 degrees F. This assures that all four (4) MSSVs are available for redundancy. During power operations at 5% full power or above, if MSSVs are inoperable, the power level must be reduced, as stated in Specification 3.4.1.2.3 such that the remaining MSSVs can prevent overpressure on a turbine trip.

Bases (Cont'd)

The equipment testing and system sampling frequencies specified in Tables 4.1-2, 4.1-3, and 4.1-5 are considered adequate to maintain the equipment and systems in a safe operational status.

REFERENCE

(1) UFSAR, Section 7.1.2.3(d) - "Periodic Testing and Reliability"
(2) NRC SER for BAW-10167A, Supplement 1, December 5, 1988.
(3) BAW-10167, May 1986.
(4) BAW-10167A, Supplement 3, February 1998
(5) EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

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Amendment No. ~~181, 225, 255~~,

INSERT TO TS PAGE 4-2b (BASES FOR SECTION 4.1)

The primary to secondary leakage surveillance in TS Table 4.1-2, Item 12, verifies that the sum of the primary to secondary leakage from both SGs is less than or equal to 144 gallons per day. Satisfying the primary to secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this surveillance is not met, compliance with TS 3.1.1.2, "Steam Generator (SG) Tube Integrity," and TS 3.1.6.3, should be evaluated. The 144 gallons per day limit is measured at room temperature. The operational leakage rate limit applies to the sum of the leakage through both SGs.

The TS Table 4.1-2 primary to secondary leakage surveillance is modified by a Note, which states that the initial surveillance is not required to be performed until 12 hours after establishment of steady state operation.

The TS Table 4.1-2 primary to secondary leakage 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. 5).

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2. Control Rod Movement	Movement of each rod	Every 92 days, when reactor is critical
3. Pressurizer Safety Valves	Setpoint	In accordance with the Inservice Testing Program
4. Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5. Refueling System Interlocks	Functional	Start of each refueling period
6. (Deleted)	--	--
7. Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525 degrees F (Not applicable to primary-to-secondary leakage)
8. (Deleted)	--	--
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation - Visual inspection of Intake Pump House Floor	Not to exceed 24 months
	(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11. Pressurizer Block Valve (RC-V2)	Functional*	Quarterly
12. Primary to Secondary Leakage	Evaluate	Every 72 hours (Note: Not required to be performed until 12 hours after establishment of steady state operation.)

* Function shall be demonstrated by operating the valve through one complete cycle of full travel.

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STEAM GENERATOR (SG) TUBE INTEGRITY

4.19 ~~OTSG TUBE INSERVICE INSPECTION~~

Applicability

This Technical Specification applies to the inservice inspection of the OTSG tube portion of the reactor coolant pressure boundary.

Objective

The objective of this inservice inspection program is to provide assurance of continued integrity of the tube portion of the Once-Through Steam Generators, while at the same time minimizing radiation exposure to personnel in the performance of the inspection.

Specification

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.1.6.3.

4.19.1 Steam Generator Sample Selection and Inspection Methods

a. Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.19.1 at the frequency specified in 4.19.3.

b. Inservice inspection of steam generator tubing shall include nondestructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall be calibrated to provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.19.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.19.2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.19.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.19.4. The tubes selected for

Amendment No. ~~47~~,
(12-22-78)

each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.19.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

4. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

   (1) Group A-1: Tubes in rows 73 through 79 adjacent to the open inspection lane, and tubes between and on lines drawn from tube 66-1 to tube 75-15 and from 86-1 to 77-15.

   (2) Group A-2: Tubes having a drilled opening in the 15th support plate.

b. The tubes selected as the second and third samples (if required by Table 4.19.2) during each inservice inspection may be subjected to a partial tube inspection provided:

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

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

c. Implementation of the repair criteria for Inside Diameter (ID) Inter-Granular Attack (IGA) requires 100% bobbin coil inspection of all non-plugged tubes in accordance with AmerGen Engineering Report, ECR No. TM 01-00328, during all subsequent steam generator inspection intervals pursuant to Section 4.19.3. ID IGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes, as defined in that report.

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 in a steam generator are degraded tubes and none of the inspected tubes are defective.

4.19.2 Specification (Continued)

C-2 One or more tubes, but not more than 1% of the total tubes inspected in a steam generator 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 in a steam generator are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES: (1) In all inspections, previously degraded tubes whose degradation has not been spanned by a sleeve must exhibit significant increase in the applicable degradation size measurement (> **0.24** volt bobbin coil amplitude increase for inside diameter IGA indications or > 10% further wall penetration for all other degradation) to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.19.2.a.4, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.19.3 Inspection Frequencies

The required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first (baseline) inspection was performed after 6 effective full power months but within 24 calendar months of initial criticality. The subsequent inservice inspections shall be performed not more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group of tubes* encompassing not less than 18 calendar months all fall into the C-1 category or demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.19-2 at 40 month intervals for a given group of tubes* fall into Category C-3 the inspection frequency for that group shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.19.3.a; the interval may then be extended to a maximum of once per 40 months.

* A group of tubes means:
(a) All tubes inspected pursuant to 4.19.2.a.4, or
(b) All tubes in a steam generator less those inspected pursuant to 4.19.2.a.4

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.19-2 during the shutdown subsequent to any of the following conditions:

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

2. A loss of coolant accident requiring actuation of engineering safeguards, or

3. A major main steam line or feedwater line break.

d. After primary-to-secondary tube leakage (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.6.3, an inspection of the affected steam generator will be performed in accordance with the following criteria:

1. If the leak is above the 14th tube support plate in a Group as defined in Section 4.19.2.a.4(1) all of the tubes in this Group in the affected steam generator will be inspected above the 14th tube support plate. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

2. If the leaking tube is not as defined in Section 4.19.3.d.1, then an inspection will be performed on the affected steam generator(s) in accordance with Table 4.19-2.

4.19.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawing or specifications. Eddy current testing indications less than degraded tube criteria specified in a.3 below may be considered imperfections.

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

3. Degraded Tube means a tube containing:

(a) an inside diameter (I.D.) IGA indication with a bobbin coil indication ≥ 0.2 volt or ≥ 0.13 inches axial extent or ≥ 0.26 inches circumferential extent, or

(b) imperfections $\geq 20\%$ of the nominal wall thickness caused by degradation.

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

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

6. Repair Limit means the extent of degradation at or beyond which the tube shall be repaired or removed from service because it may become unserviceable prior to the next inspection.

   This limit is equal to 40% of the nominal tube wall thickness. Inside diameter IGA indications shall be repaired or removed from service if they exceed an axial extent of 0.25 inches, or a circumferential extent of 0.52 inches, or a through wall degradation dimensions of ≥ 40% if assigned.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.19.3.c., above.

8. Tube Inspection means an inspection of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet, except as permitted by 4.19.2.b.2, above.

9. Inside Diameter Inter-Granular Attack (IGA) Indication means an indication initiating on the inside diameter surface and confirmed by diagnostic ECT to have a volumetric morphology characteristic of IGA.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, of all tubes exceeding the repair limit and all tubes containing throughwall cracks) required by Table 4.19-2.

4.19.5 Reports

a. DELETED

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 90 days following completion of the inspection and repairs (main generator breaker closure). The report shall include:

1. Number and extent of tubes inspected.

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

3. Location, bobbin coil depth estimate (if determined), bobbin coil amplitude (if determined), and axial and circumferential extent for each inside diameter IGA indication, and

4. Identification of tubes repaired or removed from service.

5. The number of tubes repaired or removed from service in each steam generator,

6. An assessment of growth of inside diameter IGA degradation in accordance with the volumetric ID IGA management program contained in AmerGen Engineering Report, ECR No. TM 01-00328, and

7. Results of in-situ pressure testing, if performed.

c. Results of steam generator tube inspections which fall into Category C-3 require notification in accordance with 10 CFR 50.72 prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence in accordance with 10 CFR 50.73.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained.

The program for inservice inspection of steam generator tubes is based on modification of Regulatory Guide 1.83, Revision 1. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

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

The extent of steam generator tube leakage due to cracking would be limited by the secondary coolant activity, Specification 3.1.6.3.

The extent of cracking during plant operation would be limited by the limitation of total steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gpm). Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired or removed from service.

Wastage-type defects are unlikely with proper chemistry treatment of the primary or the secondary coolant. However, even if a defect would develop in service, it will be found during scheduled inservice steam generator tube examinations. For tubes with ID IGA indications, additional conservatism is being applied to evaluate circumferential and axial dimensions for determining final disposition of the tube. For ID IGA indications through wall dimension will continue to be assigned to those indications where amplitude response permits measuring through wall dimension. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness. Tubes with I.D. initiated intergranular degradation may remain in service without % T.W. sizing if the degradation morphology has been characterized as not crack-like by diagnostic eddy current inspection and the degradation is of limited circumferential and axial length to ensure tube structural integrity. Additionally, serviceability for accident leakage under the limiting postulated Main Steam Line Break (MSLB) accident will be evaluated by determining that this I.D. initiated degradation mechanism is inactive (e.g. comparison of the outage examination

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results with the results from past outages meets the requirements of AmerGen Engineering Report, ECR No. TM 01-00328) and by successful in-situ pressure testing of a sample of these degraded tubes to evaluate their accident leakage potential when in-situ pressure tests are performed.

Where experience in similar plants with similar water chemistry, as documented by USNRC Bulletins/Notices, indicate critical areas to be inspected, at least 50% of the tubes inspected should be from these critical areas. First sample inspections sample size may be modified subject to NRC review and approval.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 on the first sample inspection (See Table 4.19.2), these results will be reported to NRC pursuant to the requirements of Specification 4.19.5.c. Such cases will be considered by the NRC on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.

NOTE: The eddy current examination voltages referred to in this section (section 4.19) are based on a normalization procedure that sets the bobbin coil prime frequency peak-to-peak response from the four 20% through-wall holes of an ASME calibration standard to 4 volts.

TABLE 4.19-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	None
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATION:

1. The Inservice Inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in that steam generator if the results of the first and subsequent inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

Amendment No. ~~47~~
(12-22-78)

TABLE 4.19-2
STEAM GENERATION TUBE INSPECTION[2]

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.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or 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.
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in other S.G. Provide notification to NRC pursuant to 10CFR50.72.b.2.i and submit a report pursuant to 10CFR50.73.-a.2.ii.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Provide notification to NRC pursuant to 10CFR50.72 b.2.i and submit a report pursuant to 10CFR50.73.a.2.ii.	N/A	N/A

Notes: (1) $S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) For tubes inspected pursuant to 4.19.2.a.4: No action is required for C-1 results. For C-2 results in one or both steam generators plug or repair defective tubes. For C-3 results in one or both steam generators, plug or repair defective tubes and provide notification to NRC pursuant to 10 CFR 50.72.b.2.i followed by a written report pursuant to 10 CFR 50.73.a.2.ii.

INSERT TO TS PAGE 4-77 (REVISED TS 4.19)

4.19 STEAM GENERATOR (SG) TUBE INTEGRITY

Applicability: Whenever the reactor coolant average temperature is above 200°F

Surveillance Requirements (SR):

Each steam generator shall be determined to have tube integrity by performance of the following:

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

4.19.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to exceeding an average reactor coolant temperature of 200°F following an SG tube inspection.

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 TS Section 3.4.

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

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

Specification 6.19, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.19, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and

BACKGROUND (continued)

operational leakage. The SG performance criteria are described in Specification 6.19. 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 associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary leakage from all SGs of 1 gallon per minute or is assumed to increase to the leakage rates described in TS 6.19.c.1 as a result of accident-induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is conservatively assumed to be equal to, or greater than, the TS 3.1.4, "Reactor Coolant System 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 TS 3.1.1.2.a

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

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall 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. A portion of the parent tube length has been

LCO (continued)

removed from service in the sleeved tubes, so examination requirements for sleeved and unsleeved tubing lengths are described in the Specification.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.19, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

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

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG, except for specific types of degradation at specific locations

LCO (continued)	where the NRC has approved greater accident induced leakage. (Refer to TS 6.19.c for specific types of degradation and approved repair criteria.) 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 TS 3.1.6.3, "LEAKAGE," and limits the sum of the primary to secondary leakage from both SGs to 144 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.
APPLICABILITY	Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced when the reactor coolant system average temperature is above 200°F. RCS conditions are far less challenging when average temperature is at or below 200°F; primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.
ACTIONS	The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions. <u>3.1.1.2.a.(3.)a. and 3.1.1.2.a.(3.)b.</u> 3.1.1.2.a.(3.) applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 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 a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, 3.1.1.2.a.(4.) applies.

ACTIONS (continued)

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

If the evaluation determines that the affected tube(s) have tube integrity, Required Action 3.1.1.2.a.(3.)b. allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to exceeding a reactor coolant average temperature of 200°F 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.

<u>3.1.1.2.a.(4.)</u>

If the Required Actions and associated Completion Times of Condition 3.1.1.2.a.(3.) are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within 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 REQUIREMENT SR 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.

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

SURVEILLANCE REQUIREMENTS (continued)

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

SURVEILLANCE REQUIREMENT SR 4.19.2:

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.19 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.

Tubes with inside diameter (ID) initiated intergranular degradation may remain in service without percent throughwall sizing if the degradation has been characterized as not crack-like by diagnostic eddy current inspection and if the degradation is of limited circumferential and axial length to ensure tube structural integrity. Additionally, accident leakage under the limiting postulated Main Steam Line Break (MSLB) accident will be evaluated by determining that this ID initiated degradation mechanism is inactive (e.g., comparison of the outage examination results with the results from past outages meets the requirements of AmerGen Engineering Report ECR No. TM 01-00328) and by successful in-situ pressure testing of a sample of these degraded tubes to evaluate their accident leakage potential when in-situ pressure tests are performed.

Steam generator tube repairs are described in TS Section 6.19.f. All in-service tubes were repaired by kinetic expansion in the early 1980's, and approximately 250 tubes in each SG were sleeved in the early 1990's. Installation of additional kinetic expansions, sleeves, or other type of tube repair requires prior NRC approval. ECR 02-01121 prescribes examination requirements and flaw dispositioning criteria for the kinetic expansions and sleeves. NRC approval of ECR 02-01121 was provided under Reference 7.

The frequency of "prior to exceeding an average reactor coolant temperature of 200°F following an SG tube inspection" ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

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".
7. U.S.N.R.C. Letter, "Three Mile Island Nuclear Station, Unit 1 – Steam Generator Tube Kinetic Expansion Inspection and Repair Criteria (TAC No.MC7001)", November 8, 2005.

(Pages 4-84 through 4-85 deleted)

6.9.5 CORE OPERATING LIMITS REPORT

6.9.5.1 The core operating limits addressed by the individual Technical Specifications shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle.

6.9.5.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at TMI-1, specifically:

(1) BAW-10179 P-A, "Safety and Methodology for Acceptable Cycle Reload Analyses." The current revision level shall be specified in the COLR.

(2) TR-078-A, "TMI-1 Transient Analyses Using the RETRAN Computer Code", Revision 0. NRC SER dated 2/10/97.

(3) TR-087-A, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code", Revision 0. NRC SER dated 12/19/96.

(4) TR-091-A, "Steady State Reactor Physics Methodology for TMI-1", Revision 0. NRC SER dated 2/21/96.

(5) TR-092P-A, "TMI-1 Reload Design and Setpoint Methodology", Revision 0. NRC SER dated 4/22/97.

(6) BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", NRC SER dated February 4, 2000.

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

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

6.9.6 STEAM GENERATOR TUBE INSPECTION REPORT

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6.9.6 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 90 days after the average reactor coolant temperature exceeds 200°F following completion of an inspection performed in accordance with Section 6.19, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

i. Location, bobbin coil depth estimate (if determined), bobbin coil amplitude (if determined), and axial and circumferential extent for each inside diameter (ID) IGA indication.

j. An assessment of growth of inside diameter IGA degradation in accordance with the volumetric ID IGA management program contained in AmerGen Engineering Report, ECR No. TM 01-00328.

k. The information specified for reporting in ECR No. 02-01121, Rev.2.

l. The number and percentage of inservice tubes repaired by each method existing in the SGs.

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

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

2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of Specification 6.18.b.1 or 6.18.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71 (e).

6.19 STEAM GENERATOR (SG) PROGRAM

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6.19 STEAM GENERATOR (SG) PROGRAM

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage volume or rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage volume or rate in the accident analysis in terms of total leakage volume or rate for all SGs and leakage volume or rate for an individual SG. Leakage from all sources excluding the leakage attributed to the degradation described in TS Section 6.19.c.1.b is also not to exceed 1 gpm per SG.

   3. The operational leakage performance criterion is specified in TS 3.1.6, "LEAKAGE."

c. Provisions for SG tube repair criteria.

1. The non-sleeved regions of 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.

   The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

   a. Volumetric Inside Diameter (ID) Inter-Granular Attack (IGA) indications may be dispositioned in accordance with ECR No. TM 01-00328. (ECR No. TM 01-00328 is not applicable to tube sleeves nor the parent tubing spanned by the sleeves.) ID IGA indication means an indication initiating on the inside diameter surface and confirmed by diagnostic ECT to have a volumetric morphology characteristic of IGA. ID IGA indications shall be removed from service if they exceed an axial extent of 0.25 inches, or a circumferential extent of 0.52 inches, or a through wall degradation dimension of ≥ 40% if assigned.

   b. Upper tubesheet kinetic expansion indications may be dispositioned in accordance with ECR No. TM 02-01121, Rev. 2.

2. Tubes found by inservice inspection to contain a flaw in a sleeve, or in a sleeve's parent tube adjacent to the sleeve between the lower sleeve end and the parent tube kinetic expansion transition, shall be "plugged-on-detection."

3. Sleeved tubes found by inservice inspection to contain any of the following attributes in the parent tubing adjacent to the sleeve upper tubesheet roll expansion shall be removed from service:

   a) The parent tubing is not present.
   b) There is a change in the number of indications present.
   c) There is a change in the orientation/morphology of the indications.
   d) There is a significant change in the circumferential extents of the circumferential and volumetric flaws.
   e) There is a significant change in the axial extents of the axial and volumetric flaws.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's sleeve-to-tube rolled joints is not an area requiring re-inspection since the inspection is exempted from the top of the middle sleeve roll to the bottom of the upper most sleeve roll (upper tubesheet roll). In addition to meeting the requirements of d.1, d.2, d.3, d.4, and d.5 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and

location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. Inspect 100% of the tubes at sequential periods of 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 for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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

4. Implementation of the repair criteria for ID IGA requires 100% bobbin coil inspection of all non-plugged tubes using inspection methods and probes in accordance with ECR No. TM 01-00328. ID IGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes, as defined in that report.

5. Implementation of the repair criteria for kinetic expansion indications requires 100% rotating probe inspection of the required lengths of the kinetic expansions in all non-plugged, non-sleeved, tubes using inspection methods and probes in accordance with ECR No. TM 02-01121, Rev.2.

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.

TMI-1's kinetic expansion repairs installed in the 1980's, and without flaws exceeding the criteria of 6.19.c.1.b, may remain in service subject to the requirements of TS Sections 3.1.1.2, 4.19, and 6.19.

TMI-1's 80" Inconel-690 rolled sleeves installed in 1991 and 1993, and without flaws exceeding the repair criteria of 6.19.c.2 or 6.19.c.3, may remain in service subject to the requirements of TS Sections 3.1.1.2, 4.19, and 6.19.

Installation of new repair methods, additional kinetic expansions, or additional sleeves, requires prior NRC approval.

NOTE: Refer to Section 6.9.6 for reporting requirements for periodic SG tube inspections.

ENCLOSURE 3

TMI Unit 1 Technical Specification Change Request No. 331

List of Commitments

SUMMARY OF AMERGEN COMMITMENTS

The following table identifies regulatory commitments made in this document by AmerGen. (Any other actions discussed in the submittal represent intended or planned actions by AmerGen. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
When sleeve upper tubesheet roll joints are examined, the associated parent tube will be examined using +Point MRPC probes using EPRI qualified techniques. These techniques are qualified for the detection of OD IGA and OD IGA/SCC in the parent tube. The acceptance criteria for the parent tube will be based on a comparison of the current outage's data and the earliest available historical +Point data for that tube.	1R17 Refueling Outage (Fall 2007)	No	Yes