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10 CFR 50.90

October 9, 2008
5928-08-20010

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

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

Subject: Technical Specification Change Request No. 343: Application for Technical Specification Change to Reflect Steam Generator Replacement

References:

1. TSTF-449, Revision 4, "Steam Generator Tube Integrity."
2. U.S.N.R.C. Letter, "Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Generic Letter 2006-01 (TAC Nos. MD1807 and MD0115)", P. Bamford to C. Crane, September 27, 2007.

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," AmerGen Energy Company LLC (AmerGen) requests an Operating License amendment to revise the existing Three Mile Island, Unit 1 (TMI, Unit 1) Steam Generator (SG) tube surveillance program. The TMI, Unit 1 Technical Specifications (TSs) were previously revised to be consistent with TSTF-449, Revision 4 for its current SGs (References 1 and 2). The proposed changes reflect the new thermally treated Alloy 690 tubing design of the replacement SGs and remove sections of the TSs that are not applicable to the replacement SGs. AmerGen has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1).

The proposed amendment has been reviewed by the TMI, Unit 1 Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the AmerGen Quality Assurance Program.

AmerGen requests approval of the proposed amendment by October 9, 2009. Approval by October 9, 2009 will allow the orderly implementation of the proposed changes at the plant site. Once approved, the amendment shall be implemented prior to resumption of plant operation following the T1R18 (SG replacement) refueling outage.

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No new regulatory commitments are established by this submittal.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), AmerGen is notifying the State of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official. In addition, copies are being distributed to the Bureau of Radiation Protection and the chief executives of the township and county in which the facility is located.

Should you have any questions concerning this letter, please contact Wendy E. Rapisarda at (610) 765-5726.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 9th day of October, 2008.

Respectfully,

Pamela B. Cowan

Pamela B. Cowan
Director - Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specifications and Bases Page Changes

cc: USNRC Administrator, Region I
USNRC Project Manager, TMI, Unit 1
USNRC Senior Resident Inspector, TMI, Unit 1
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection
Chairman, Board of County Commissioners of Dauphin County, PA
Chairman, Board of Supervisors of Londonderry Township, Dauphin County, PA
TMI File No. 08006

ATTACHMENT 1

Evaluation of Proposed Changes

Subject: Application for Technical Specification Change to Reflect Steam Generator Replacement.

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License No. DPR-50 for Three Mile Island, Unit 1 (TMI, Unit 1).

The proposed changes would revise the Operating License to reflect the plant's replacement Steam Generators (SGs) planned to be installed during the T1R18 refueling outage, which is scheduled to begin in the fall of 2009. The proposed amendment modifies the Technical Specifications (TSs) to eliminate the existing requirements associated with tube sleeve repairs and alternate repair criteria, incorporates a revised primary-to-secondary leakage criterion, changes the required reporting period for SG inspection results, and incorporates revised tube integrity surveillance frequency requirements for Alloy 690 tubing.

2.0 DETAILED DESCRIPTION

These proposed changes, although developed in accordance with TSTF-449, Revision 4, are not being submitted via the Consolidated Line Item Improvement Process because the TMI, Unit 1 TSs were previously revised to be consistent with TSTF-449, Revision 4 for its current SGs (References 6.1, 6.2).

The proposed changes revise TS 3.1.6, "LEAKAGE," TS 4.19, "OTSG TUBE INSERVICE INSPECTION," TS 6.19 "STEAM GENERATOR (SG) PROGRAM," and TS 6.9.6 "STEAM GENERATOR TUBE INSPECTION REPORT." The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," with respect to the new TMI, Unit 1 SGs. The proposed TMI, Unit 1 TS changes include:

> **NOTE:** Proposed revisions to the TS Bases are also included in this application for information only. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

- On Page 3-12 of the TS, TS 3.1.6.3 is revised to change the allowable primary-to-secondary leakrate limit from 144 gallons per day (GPD) for the sum of leakage from both SGs to 150 GPD for each SG. This limit is based on operating experience with SG tube degradation mechanisms that result in leakage and provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. Additionally, the 150 GPD limit for each SG is standard for the U. S. PWR industry (Reference 6.1). In the 1980s the TMI, Unit 1 primary-to-secondary leakage limit was revised to the lower limit (0.1 GPM / 144 GPD for the sum of leakage from both TMI, UNIT 1 SGs) as a result of SG tube degradation and operating license conditions associated with the SG tube kinetic expansion repairs (Reference 6.3). The lower limit implemented for the 1980s degradation and kinetic expansion will no longer be applicable with the installation of replacement SGs at the plant.

- Pages 3-15a, 4-2b, and 4-78 of the TS, TS 3.1.6 Bases, TS 4.1 Bases, and TS 4.19 Bases are revised to be consistent with the proposed changes to TS 3.1.6.3, described above.

- Pages 4-78, 4-79, and 4-80 of the TS, and Page 4.19 of the TS Bases are revised to eliminate discussion of alternate repair criteria, including reference to alternate repair criteria of Section 6.19.c.1 of the current TS. TS Section 6.19.c.1 is revised under these proposed changes, as described below, to eliminate two alternate repair criteria that are presently applicable to the plant's current SGs, but will not be applicable to the plant's replacement SGs. There are no alternate repair criteria applicable to, or approved for, the plant's replacement SGs. TS 4.19 Bases are also revised to eliminate discussion pertaining to tube sleeves. While the plant's current SGs have tubes that have been repaired by sleeves, no sleeve repairs have been installed, or are authorized for installation, in the plant's replacement SGs.

- On Page 4-80 of the TS, TS 4.19 Bases are revised to change the primary-to-secondary leakrate limit from 144 gallons per day for the sum of the leakage from both SGs to 150 gallons per day from each SG, consistent with proposed TS 3.1.6.3, as described above.

- On Page 4-82 of the TS, TS 4.19 Bases are revised to eliminate discussion that pertains to the inside diameter initiated intergranular degradation (Volumetric ID IGA) repair criteria. This alternate repair criteria information is applicable to the plant's current SGs and is not applicable to the plant's replacement SGs. As described above, no alternate repair criteria are applicable to the plant's replacement SGs.

- On Page 4-83 of the TS, TS 4.19 Bases are revised to eliminate discussion of kinetic expansion repairs. These repairs are applicable to the plant's current SGs, but are not applicable to the plant's replacement SGs. In addition, Bases reference No. 7, related to the kinetic expansion criteria, is deleted.

- On Page 6-19 of the TS, TS 6.9.6 is revised to change the required reporting period for SG inspection results from 90 days to 180 days. The revised reporting time period is the standard for the U. S. PWR industry (Reference 6.1).

- On Page 6-20 of the TS, TS 6.9.6 is revised to delete reference to tube repairs, since no repair methods have been approved for, or are applicable to, the replacement SGs. In addition, four items are deleted from the list of items to be included in the inspection reports. These four items are applicable to the alternate repair criteria for the plant's current SGs, but are not applicable to the replacement SGs.

- On Page 6-27 of the TS, TS 6.19.b.2 and 6.19.c.1, 2, and 3 are revised to delete all of the sentences pertaining to the current SGs' alternate repair criteria for kinetic expansions and Volumetric ID IGA, since the replacement SGs will have no alternate repair criteria. Since the replacement SGs will have no installed sleeves, all of the sentences pertaining to sleeves are deleted. Text that differentiates sleeved sections of tubing from non-sleeved sections of tubing is deleted. The phrase "volume or", which had been inserted into the TS since the plant's alternate repair criteria for kinetic expansion indications utilized a leakage volume (vice rate) acceptance criterion, is deleted since that criterion will no longer be applicable to the replacement SGs. Since Section 6.19.c.1.b is deleted, the phrase "from all sources excluding the leakage attributed to the degradation described in TS Section 6.19.c.1.b is also" is deleted.

- On Page 6-28 of the TS, all discussion referring to the inspection and dispositioning criteria for sleeves is removed from TS 6.19.d, since the replacement SGs will not have sleeved tubes. The inspection requirements for Volumetric ID IGA and kinetic expansion alternate repair criteria contained in 6.19.d.4 and 6.19.d.5 are deleted, since these are not applicable to, or approved for, the replacement SGs. Since the TMI, Unit 1 replacement SGs will contain Alloy 690 tubes, TS 6.19.d.2 is revised to incorporate the prescriptive inspection intervals required for SGs with Alloy 690 tubing.

- On Pages 6-28 and 6-29 of the TS, TS 6.19.f pertaining to SG tube repair methods is deleted since, as described above, these will no longer be applicable to the replacement SGs.

3.0 TECHNICAL EVALUATION

TMI, Unit 1 currently utilizes two SGs that were supplied as a part of the original nuclear steam supply system by Babcock and Wilcox. The current and replacement TMI, Unit 1 SGs are straight-tube, vertical, counter-flow, once-through heat exchangers with shell-side boiling of secondary fluid. Primary fluid from the reactor enters through an inlet nozzle in the top head, flows down through the tubes, is collected in the bottom head, and exits through two primary outlet nozzles. The use of straight tubes results in almost pure counter-flow properties.

The TMI, Unit 1 replacement SGs are being manufactured by AREVA in Chalon, France and are being designed, manufactured, and tested in accordance with ASME Code Section III, Class 1 requirements. The design, procurement, and manufacturing processes are being performed under a Quality Assurance Program that complies with the requirements of 10 CFR 50, Appendix B and with the current NRC requirements that relate to SG design. Significant design changes for the replacement SGs include:

- Use of thermally treated Alloy 690 tube material with full depth tube sheet expansions

- Addition of an integral flow restrictor in each main steam nozzle

- Greater secondary side volumes due to thinner shells of higher strength shell material

- More corrosion resistant material for the tube support plates

- Higher design secondary side pressure and temperature ratings.

The replacement SGs occupy essentially the same physical space as the original SGs and are very similar in thermal and hydraulic performance to the original SGs. Therefore, the replacement SGs are similar to the original SGs and will be replaced under the requirements of 10 CFR 50.59.

The proposed TS changes do not affect primary coolant chemistry controls. The primary coolant activity limit and its assumptions are not affected by these proposed changes to the TSs.

The proposed TS changes include a change to the current TS limit on primary-to-secondary leakage of 144 GPD that was established in the 1980s due to kinetic expansion repairs of SG tube degradation. The basis for this limit will no longer be applicable with the installation of replacement SGs. The proposed limit of 150 gallons per day of primary-to-secondary leakage through any one SG is "standard" for the U. S. PWR industry and was implemented under TSTF-449 (Reference 6.1). This limit is based on operating experience as an indication of one or more tube leaks and provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident (Reference 6.1). Further, if it is not practical to assign the leakage to an individual SG, all the primary-to-secondary leakage is conservatively assumed to be from one SG. This operational leakage rate criterion, in conjunction with the implementation of the SG Program, is an effective measure for minimizing the frequency of SG tube ruptures. The primary-to-secondary leakage limit helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 100 and GDC 19 dose limits or other NRC approved licensing bases for postulated faulted events. This limit also contributes to meeting the GDC 14 requirement that the reactor coolant pressure boundary "have an extremely low probability of abnormal leakage, of rapidly propagating to failure, and of gross rupture."

The current TSs contain prescriptive inspection intervals for Alloy 600 mill annealed tubing (600MA). The proposed TS inspection intervals for the replacement SGs reflect advanced materials consistent with TSTF-449. Following the 100% inspection requirements during the first refueling outage following SG replacement, the maximum inspection interval for the replacement SGs' Alloy 690 thermally treated tubing is:

> "Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first in-service inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected."

Longer inspection intervals for the replacement SGs are only achievable early in SG life and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. The performance of Alloy 690 tubing has been significantly better than the performance of 600MA tubing, the material used in the plant's current SGs.

The proposed change to TS 6.9.6 replaces the 90-day report with a report required within 180 days. The 180-day period is now industry "standard" practice per TSTF-449. Safety significant SG tube degradation would be reportable in accordance with 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requiring NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence.

The proposed TS revisions also eliminate a considerable amount of material from the plant's TS that is applicable to the current SGs, but which is not applicable to the replacement SGs. The current SG TSs include alternate repair criteria for Volumetric ID IGA indications and alternate repair criteria for indications within the upper tubesheet kinetic expansions. These alternate repair criteria are not applicable to the plant's replacement SGs. The current TSs also include kinetic expansion and sleeve repair techniques. These repair techniques are not applicable to the plant's replacement SGs. The analyses that formed the basis of approvals for these repair criteria, and these repair techniques, are not applicable to the replacement SGs.

The proposed TS changes delete the alternate repair criteria, as described above, that are applicable to the plant's current SGs and not applicable to the replacement SGs. The repair criterion that is left in the TS is the standard 40% through-wall criterion.

Revision of the TMI, Unit 1 UFSAR is required to document the replacement SGs. UFSAR changes are currently being drafted. The scope of the proposed UFSAR changes includes a number of similar items as are included in the scope of this TS change request (e.g., removal of kinetic expansions, removal of sleeves, tubing alloy changes, etc.).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

TSTF-449 (Reference 6.1) provides the applicable regulatory requirements for TS to implement the site's steam generator program. As described above in Section 3.0, the proposed TS changes modify the site's TSs to be consistent with those prescribed by TSTF-449 for sites with Alloy 690-tubed steam generators. This change does not affect the applicability of the following regulatory requirements:

- 10 CFR 50.55a, Codes and Standards - Section (b), ASME Code, c) *Reactor Coolant Pressure Boundary*
- 10 CFR 50.65 Maintenance Rule
- General Design Criteria (GDC) 14, 30, and 32 of 10 CFR Part 50, Appendix A

4.2 Precedent

The NRC issued License Amendment No. 223 for Crystal River, Unit 3 (CR, Unit 3) on May 16, 2007 (ML071340112). The issuance approved CR, Unit 3's May 25, 2006 License Amendment Request No. 264 (ML061500062) submittal in accordance with TSTF-449 for their current SG configuration. Following the NRC approval, CR, Unit 3 submitted License Amendment Request No. 301 dated August 28, 2008 (ML082460317) to update their TSs to support their planned SG replacement in 2009. The CR, Unit 3 license amendment request was submitted in accordance with TSTF-449, but not the Consolidated Line Item Improvement Process.

4.3 No Significant Hazards Consideration Determination

TMI Unit 1 has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed changes to the Technical Specifications (TSs) for the TMI, Unit 1 Steam Generator (SG) Program recognize that the TMI, Unit 1 SGs are being replaced and the standard industry performance criteria documented in TSTF-449 for Alloy 690-tubed SGs will apply. These changes eliminate criteria that were established to reflect the condition and materials of the current TMI, Unit 1 SGs, and add the requirements for inspection of Alloy 690-tubed SGs from TSTF-449.

With these proposed TS changes, the operational primary-to-secondary leakage rate limit established for the original TMI, Unit 1 SGs is replaced with the standard industry primary-to-secondary leakage rate limit. The standard industry limit is that limit provided in TSTF-449. The current, reduced limit in the TMI, Unit 1 TS was implemented in response to upper tubesheet tube expansion degradation, and repairs, in the original TMI, Unit 1 SGs. A reduced limit is not required for the replacement SGs since they are fabricated from advanced materials and were not subjected to the degradation mechanisms that influenced the original TMI, Unit 1 SGs. Thus, reverting to the standard industry limit is appropriate. The slightly higher, industry standard, leak rate limit is still low enough to provide assurance that the probability of tube ruptures, or of rapidly propagating tube leaks, remains acceptably low. Thus, the probability of a previously evaluated accident is not increased.

The installation of the new SGs, with improved materials, will decrease the consequences of SG-related accidents. The removal of accident-induced leakage attributable to the current degradation mechanisms from TS 6.19.c.1.b reduces the accident induced leakage limit to 1 gpm per SG. SG accident-induced leakage is proportional to dose; a lower accident-induced leakage limit will result in lower dose than previously evaluated accident consequences.

The proposed change to replace the 90-day report with a report required within 180 days is a change to an administrative requirement and does not affect the probability or consequences of an accident. The 180-day period is now industry "standard" practice per TSTF-449.

These changes continue to provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). With the proposed changes, the SG performance criteria (based on tube structural integrity, accident-induced leakage, and operational leakage) and SG Program are updated to reflect the replacement SGs while remaining consistent with TSTF-449.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident that was previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed TS changes recognize an improvement in SG design as a result of SG replacement. The replacement SGs contain a number of design improvements with respect to the plant's original SGs. However, even with the design improvements, the replacement SGs are very similar to the original SGs and new types of accidents are not created. There are no new design functions for the Alloy 690 tubing in the replacement SGs. The proposed new leakage and inspection requirements are the standard industry requirements for Alloy 690 tubing.

Primary-to-secondary leakage monitoring equipment is not affected by the proposed changes, and primary-to-secondary leakage will continue to be monitored to ensure it remains within current accident analysis assumptions and limits. The proposed changes implement the industry "standard" TSTF-449 primary-to-secondary leak limits for the plant's Alloy 690-tubed replacement SGs. No new types of primary-to-secondary leak accidents are created.

The proposed change to replace the 90-day report with a report required within 180 days is a change to an administrative requirement and does not create a new or different kind of accident. The 180-day period is now industry "standard" practice per TSTF-449.

Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. The SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

SG tube integrity is a function of the design, environment, and physical condition of the tubing. The proposed changes do not affect the operating environment but do recognize the improved tube material as a result of replacing the SGs. The proposed TS changes for inspection, repair, and leakage requirements are consistent with industry codes and standards for replacement SGs with Alloy 690 tubing material. The requirements established by the SG Program are consistent with those in the applicable design codes and standards. The proposed changes update the requirements in the current TSs to reflect SG replacement.

The proposed TS changes include a change to the current TS limit on primary-to-secondary leakage of 144 GPD that was established in the 1980s due to SG tube degradation. The basis for this limit will no longer be applicable with the installation of replacement SGs. The proposed limit of 150 gallons per day of primary-to-secondary leakage through any one SG is "standard" for the U. S. PWR industry. This limit is based on operating experience with SG tube degradation mechanisms that result in leakage and provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. Further, if it is not practical to assign the leakage to an individual SG, all the primary-to-secondary leakage is conservatively assumed to be from one SG. This operational leakage rate criterion, in conjunction with the implementation of the SG Program, is an effective measure for minimizing the frequency of SG tube ruptures.

The proposed change to replace the 90-day report with a report required within 180 days is a change to an administrative requirement and does not affect the margin of safety. The 180-day period is now industry "standard" practice per TSTF-449. Additionally, this TS requirement is significantly less than the conditions assumed in the safety analysis.

For the above reasons, the margin of safety is not reduced.

4.4 Conclusions

Based on the above, AmerGen concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.0 Environmental Consideration

A review has determined that the proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 References

6.1 TSTF-449, Revision 4, "Steam Generator Tube Integrity."

6.2 U.S.N.R.C. Letter, "Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Generic Letter 2006-01 (TAC Nos. MD1807 and MD0115)," P. Bamford to C. Crane, September 27, 2007.

6.3 "TMI, UNIT 1 Steam Generator Repair Safety Evaluation Report (NUREG 1019)," August 1983, Page 46.

ATTACHMENT 2

TMI, Unit 1 Technical Specification Change Request No. 343

Markup of Proposed Technical Specifications and Bases Page Changes

Revised Technical Specifications & Bases Pages

3-12
3-15a
4-2b
4-78
4-79
4-80
4-82
4-83
6-19
6-20
6-27
6-28
6-29

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 through any one (1) ~~from both~~ steam generator~~s~~ exceeds 150 GPD, ~~0.1 gpm (144 GPD)~~, the reactor shall be placed in hot shutdown within 6 hours, and in cold shutdown within 36 hours.

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.

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~~ through any one (1) SG to less than or equal to ~~144~~ 150 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~~ 150 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.

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 4.53 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.

REFERENCES

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

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.

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~~ 150 gallons per day through any one (1) SG. 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~~ 150 gallons per day limit is measured at room temperature as described in Reference 5. ~~The operational leakage rate limit applies to the sum of the leakage through both SGs.~~ INSERT A

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).

The surveillance test procedures for the Variable Low Pressure Trip Setpoint do not compare the as-found Trip Setpoint (TSP) to the previous surveillance test as-left TSP. Basing operability determinations for the as-found TSP on the Nominal Setpoint (NSP) is acceptable because:

1. The NSP as-left tolerance specified in the surveillance test procedures is less than or equal to the calculated NSP as-left tolerance.

2. The NSP as-left tolerance is not included in the Total Loop Uncertainty (TLU) calculation. This is acceptable because the NSP as-left tolerance specified in the surveillance test procedures is less than half of the calculated NSP as-left tolerance. This prevents masking of excessive drift from one side of the tolerance band to the other.

3. The pre-defined NSP as-found tolerance is based on the square root of the sum of the square of the instrument accuracy, M&TE accuracy and drift. The NSP as-left tolerance is not included in this calculation.

Credible uncertainties for the Variable Low Pressure Trip Setpoint include instrument uncertainties during normal operation including drift and measurement and test equipment uncertainties. In no case shall the pre-defined as-found acceptance criteria band overlap the Allowable Value. If one end of the pre-defined as-found acceptance criteria band is truncated due to its proximity to the Allowable Value, this does not affect the other end of the pre-defined as-found acceptance criteria band. If equipment is replaced, such that the previous as-left value is not applicable to the current configuration, the as-found acceptance criteria band is not applicable to calibration activities performed immediately following the equipment replacement.

INSERT A to TS Page 4-2b

The operational leakage rate limit applies to leakage through any one SG. If it is not practical to assign the leakage to an individual SG, all the primary to secondary leakage should be conservatively assumed to be from one SG

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.3~~ 1 gallon per minute as a result of accident-induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is 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~~

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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~~ primary to secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced 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.

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

A report shall be submitted within ~~90~~ 180 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.~~

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records of normal station operation including power levels and periods of operation at each power level.

b. Records of principal maintenance activities, including inspection, repairs, substitution, or replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of periodic checks, tests and calibrations.

e. Records of reactor physics tests and other special tests related to nuclear safety.

f. Changes to procedures required by Specification 6.8.1.

g. Deleted

h. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

i. Results of annual physical inventory verifying accountability of licensed sources on record.

j. Control Room Log Book.

k. Control Room Supervisor Log Book.

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 assumed 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~~ is 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 top of the middle sleeve roll, 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 parent tube between the top of the middle sleeve roll to the bottom of the uppermost sleeve roll (upper tubesheet roll) is not an area requiring inspection.~~ e In addition to meeting the requirements of d.1, d.2, and 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 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. INSERT A No SG shall operate for more than ~~24~~ 72 effective full power months or ~~one~~ three refueling outage (whichever is less) without being inspected. 5

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

e ~~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.

e ~~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.~~

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INSERT A to TS Page 6-28

In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period.

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.