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Washington, D.C. 20006

In the Matter of
METROPOLITAN EDISON COMPANY, ET AL.
(Three Mile Island Nuclear Station, Unit No. 1)
Docket No. 50-289 OLA
(Steam Generator License Amendment)

Dear Mr. Kelly:

It has been brought to our attention that the normal three copies of NRC Staff Exhibits 1 and 2, which exhibits were offered and admitted into the evidentiary record at hearing in the referenced proceeding on July 18, 1984, may not have been provided to the court reporter at the time of the hearing. To remedy this apparent oversight, three copies each of NRC Staff Exhibit 1 (identified and marked at Tr. 576, 648; admitted into evidence at Tr. 583, 649) and NRC Staff Exhibit 2 (identified and marked at Tr. 576-77, 648; admitted into evidence at Tr. 584, 650) are enclosed.

Sincerely,

Joseph R. Gray

Joseph R. Gray
Assistant Chief Hearing Counsel
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission

Enclosures: As stated

cc w/ single copies of enclosures:
Service list

D507

STAFF EXHIBIT

NUREG-1019

Safety Evaluation Report

related to steam generator tube repair and return to operation
Three Mile Island Nuclear Station, Unit No. 1

Docket No. 50-289

GPU Nuclear Corporation, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1983

The licensee conducted an extensive program to return the OTSG to service whic includes the following aspects addressing staff concerns:

- determination of the causative agent(s)
- examination and repair of the remainder of the reactor coolant system (RCS)
- OTSG examinations to determine extent of degradation
- OTSG repair
- cleanup of the contaminant
- procedures to prevent a re-introduction of contaminants
- post-repair testing and operational crack arrest considerations
- occupational dose assessment
- plant performance evaluation subsequent to OTSG repairs

On October 13, 1982, the staff issued a safety evaluation (SE) for the OTSG kinetic expansion repair technique. In that SE, the staff found that the repair did not involve an unreviewed safety question or a modification to the technical specifications, and hence could be conducted without prior NRC approval. However, the staff stated that NRC review and approval of the overall program to return the OTSG to service was required prior to any subsequent power operation.

By letter of May 9, 1983, the licensee submitted Technical Specification Change Request No. 125 which would revise TS 4.19 to permit operation of the plant following repair of the steam generators by methods other than plugging, provided the repair methods are approved by the NRC. That letter also requested approval of the method used by the licensee to repair the steam generator in order to permit non-nuclear heatup for pre-critical testing, and subsequent operation using the repaired steam generators. The purpose of this SE is to evaluate the specific repair method used by the licensee, and to evaluate subsequent operation using the repaired steam generators.

2 DESCRIPTION OF THE REPAIR METHOD

The as-built OTSG has two 24-inch-thick tubesheets, one at the top (UTS) and one at the bottom. The once-through straight tubes are nominally 56 feet and 1 inch in length, with 52 feet of heat transfer surface between the tubesheets. The remaining 4 feet and 1 inch includes 2 feet in each tubesheet and ½ inch protruding into the primary head at each end of the OTSG. To provide structural integrity and leak tightness, during shop fabrication all tubes were hard rolled to a nominal depth of 1-¼ inches, and seal welded on the primary side of the tubesheet surfaces. The tubesheet has a nominal 22-3/4-inch long, 8-mil radial gap (crevice) between the outer tube surface and the drilled tubesheet hole. As stated in the introduction, the preponderance of defects are located within the top 2 inches of the UTS with a rapidly decreasing number of defects down through the depth of the UTS.

The repair method utilizes a kinetic expansion process to form the tube against the tubesheet; i.e., close the 8-mil radial gap. The kinetic expansion process closes the 8-mil gap and produces an interference fit between the tube OD and tubesheet drilled holes ID to achieve a leak-tight, load-carrying joint. The tube repair procedure requires that all repaired tubes have a 2 inch defect-free unexpanded section within the UTS above the secondary side tubesheet interface. This unexpanded section will prevent tube pullout in the event a tube is severed at the repair transition zone. Developmental testing has been conducted to demonstrate that a kinetically expanded 6-inch long defect-free section of tube (qualification zone) can provide the necessary leak tightness and load carrying capabilities required for operation. Therefore, all tubes which have defects down to a depth of 16 inches into the tubesheet can be repaired. The 16-inch section plus the 2-inch unexpanded zone and the 6-inch qualification zone account for the full depth of the 24-inch thick tubesheet. The forming technique consists of inserting a polypropylene sheath into each tube. The polypropylene sheath contains a prima-cord which, when ignited, forces the polypropylene sheath against the tube. The resultant force expands the tube. The polypropylene sheath and prima-cord assembly is called a candle. The candles are detonated by a blasting cap which is maintained outside the steam generator in a sealed container and ignited electrically by a licensed blaster. The two OTSG have a total of approximately 31,000 tubes, all of which have been expanded, except those previously plugged. After all tubes were expanded, those tubes which contained non-repairable defects were plugged.

Prior to kinetic expansion the crevices between the tube and tubesheet were flushed with hydrazine-treated water and then dried out with electrical heaters to remove moisture.

3 EVALUATION OF REPAIR

By letter dated March 31, 1983, the licensee provided Topical Report 008, Rev. 2, Assessment of TMI-1 Plant Safety for Return to Service after Steam Generator Repair. Topical Report 008, Rev. 2, superseded Topical Report 008, Rev. 1 that was submitted by letter dated December 10, 1982. By telephone on July 25, 1983 and in its letter of July 29, 1983, the licensee stated that it intends to further update Topical Report 008 prior to its final determination that the steam generators are operable. The licensee also stated, and repeated in its letter of August 3, 1983, that the fundamental conclusions of Topical Report 008, Rev. 2, are not expected to be changed in that update. Our evaluation herein is based on Topical Report 008, Rev. 2. Any subsequent revision of the Topical Report will be evaluated in a supplement to this SE as part of any amendment package which would permit power operation of TMI-1 with the repaired steam generators.

Several consultants helped NRC in the evaluation of the licensee's program for return to service. The staff consultants included representatives from the following organizations; their technical evaluation reports are attached, except for that of Franklin Research Center, which is proprietary. A non-proprietary version is being prepared and will be provided when it is available.

Franklin Research Center (FRC; Attachment No. 1 - proprietary, not attached)
Brookhaven National Laboratory (BNL; Attachment No. 2)
Pacific Northwest National Laboratory (PNL; Attachment No. 3)
Ohio State University (OSU; Attachment No. 4)
Oak Ridge National Laboratory (ORNL; Attachment No. 5)

In addition to the evaluation conducted by the staff and staff consultants, the licensee established an independent third party review group (TPR) of industry experts who are not employed by the licensee or its affiliated companies. By letters dated April 4 and May 20, 1983, the licensee submitted copies of the

TPR final reports. The TPR provided an independent operational and safety evaluation which is enclosed as Attachment No. 6 and consists of an original report dated February 18, 1983 and a supplemental report dated May 16, 1983. In the cover memo for the May 16, 1983 supplemental report, the TPR concludes that "comments and recommendations relating to safety of the steam generator repair have been satisfactorily resolved by GPU nuclear." Some additional comments by the TPR which the staff has determined are not related to safety issues are being considered by the licensee. However, resolution of these comments will not have a negative effect on public health and safety.

Our evaluation is divided into eight areas associated with the cause, repair and recovery from the OTSG corrosion problems:

1. Determination of causative agents
2. Examination and repair of the remainder of the RCS
3. OTSG examinations to determine the extent of degradation
4. OTSG repair
5. Cleanup of contaminants
6. Procedures to prevent re-introduction of contaminants
7. Post-repair testing and operational crack arrest considerations
8. Occupational dose assessment

Plant performance with the repaired steam generators is discussed in Section 4 below.

3.1 Determination of Causative Agent(s)

The licensee and its consultants conducted extensive microstructural and fractographic examinations on Inconel 600 tubing specimens taken from the TMI-1 OTSG. Cracks in the observed specimens exhibited a morphology characteristic of stress-assisted intergranular attack. Austenitic stainless steels and certain nickel-base alloys, such as Inconel 600, under certain conditions, are known to be susceptible to intergranular stress corrosion cracking (IGSCC). However, the occurrence of IGSCC requires that three conditions be present simultaneously: 1) a high tensile stress, 2) a susceptible alloy microstructure, and 3) an aggressive environment.

Stress analysis on the OTSG tubes conducted by the licensee indicates that: 1) axial tensile stresses in tubing are largest during cooldown 2) axial tensile stresses also exist during cold shutdown, 3) locally high axial tensile stresses are possible in the seal weld heat affected zone and in the vicinity of the roll transition, 4) under heatup at full operating temperature, the hoop stress generally is larger than the axial stress, and 5) the axial stresses are generally larger at the periphery tubes than those in the center of the tube bundle. Based on this analysis, the licensee concluded that the tube cracking occurred during cooldown or cold shutdown because tensile stresses were highest under these conditions. By independent analysis (Attachment 1), the staff consultant confirmed the licensee's finding that high tensile stresses exist in the cooldown and cold shutdown conditions and their distribution is consistent with that reported by the licensee. Consequently, the staff agrees with the licensee's conclusion that IGSCC of the OTSG tubes occurred during the cooldown or the cold shutdown conditions. Furthermore, the stress level and its distribution are consistent with the observed failure pattern.

Conclusion

The staff finds that the PORV and safety relief valves, which exhibited pitting corrosion, were replaced with fully qualified non-corroded valves and therefore are now acceptable. The waste gas system was found to be affected, and all corroded portions of this system were replaced. The remainder of the reactor coolant system and interfacing systems which were inspected, within the limitations of the inspection methods employed, disclosed no defects attributable to sulfur-induced corrosion. Therefore, based on the above, the staff finds that GDC 1, 14, 15 and 31 have been met, and that reasonable assurance exists that the public health and safety is protected.

3.3 OTSG Examinations to Determine Extent of Degradation

Initial eddy-current examinations of the steam generators with a 0.510-inch diameter standard differential probe indicated that tube wall defects were on the inside surfaces of tubes in both steam generators. These defects were distributed radially throughout the steam generators and predominantly in the upper portions axially. The vast majority of the defect indications were in the UTS region and particularly confined to the tube roll transition zone.

In order to better quantify and characterize the defects, the licensee established a comprehensive testing program whereby special eddy-current testing (ECT) techniques were developed so that a more accurate picture of the extent of damage could be developed.

Using machined notches and laboratory-grown cracks as standards for calibration, and comparing field data to laboratory ECT and metallurgical examinations of tubes removed from service, the licensee used a self-developed eddy-current system for 100% full length inservice inspection of the tubes in both steam generators.

The eddy-current system which evolved from the testing program utilized a standard differential probe of 0.540-inch diameter, with an effective gain of approximately 60. Two base frequencies (400 KHz and 200 KHz) and an "ID" mix to enhance detection of ID defects and minimize the effect of chatter and tube noise were used. The testing program compared absolute systems to the 0.540-inch diameter standard differential high gain probe and found the latter system as sensitive as the absolute systems for detecting circumferential cracks.

The licensee's inspection results, with the 0.540-inch high gain probe eddy-current inspection system, revealed indications in addition to those previously identified by the 0.510-inch standard gain probe. However, no significant pattern of crack growth was apparent in the six-month interval between the initial 0.510-inch standard gain inspections and the 0.540-inch high gain inspections.

The licensee interpreted the eddy-current test measurements of the through-wall depth of the indications in the UTS transition zone regions as greater than 40% through-wall, and hence tubes with such indications were characterized as defective tubes. Metallurgical examination conducted by the licensee on tubes removed from service confirmed that the flaws in the transition zone region all exceeded 50% through-wall, with the majority 100% through-wall.

Based on the eddy-current results indicating the depth of the defects and the extent of the observed degradation, the licensee decided that all unplugged tubes would be kinetically expanded within the tubesheet to establish a new primary system pressure boundary below the defect area.

After kinetic expansion, additional eddy current examinations, using an 8 x 1 absolute probe, revealed 9 of 151 tubes examined in steam generator A, and 6 of 284 tubes examined in steam generator B, with indications not previously seen by the 0.540-inch standard differential high gain probe. Fiberscope examinations of the new eddy current indications revealed small pits and scratches which were below the sensitivity of the 0.540-inch standard differential high gain probe. These indications, which are not considered to be of safety significance, will be reexamined at the mid-cycle ECT inspection and evaluated to confirm the decision that they are acceptable. Any future eddy current examinations in the kinetic expansion region will be performed with the 8 x 1 absolute probe.

Of the 31,062 tubes in both steam generators, 29,838 with no known defects below 16 inches from the primary (top) surface of the UTS were repaired by kinetic expansion and returned to service. A total of 347 tubes had been removed from service by plugging prior to the start of the kinetic expansion repair program. An additional 811 tubes with greater than 40% through-wall indications 16 inches or more below the primary surface of the UTS were removed from service by plugging after kinetic expansion. Approximately 475 of the plugged tubes were also stabilized, i.e, staked, with internal rods to prevent damage to adjacent tubes in the event the degradation continues and the tube severs.

The purpose of the stabilization of plugged tubes is to reduce the risk due to propagation of tube defects located in regions with high potential for flow induced vibration, which could result in circumferential tube severance, and thus cause damage to adjacent tubes or create loose parts. The affected tubes, which are in the area of high steam cross flow (16th span) were stabilized to the 14th tube support plate.

Fracture mechanics analysis of circumferential tube defects conducted by the licensee shows that circumferential defects less than 40% through-wall are acceptable because they would not propagate due to vibration during normal operation or accident conditions. The staff consultant's analysis (Attachment 1) confirmed the licensee's conclusion and, therefore, the staff agrees with the licensee that circumferential defects less than 40% through-wall are acceptable, and they would not propagate due to vibration during normal or accident conditions.

Approximately 66 tubes with ECT indications between 20-40% through-wall, as verified by an 8 x 1 absolute probe and located 16 inches or more below the primary surface of the UTS, are considered degraded tubes. Approximately seventeen of these degraded tubes have ID indications which are attributable to the sulfur-induced corrosion problem. To verify that the corrosion mechanism has been arrested, these 66 tubes will be left in service and monitored in the post-repair extended Inservice Inspection (ISI) program. The extended ISI program will include 100% reinspection of tubes with 40% and less through-wall indications as a separate subset during subsequent examinations. If eddy-current examinations show no substantial growth in the cracks, they will be left in service. Tubes showing signs of crack propagation based on previously established acceptance criteria will be taken out of service. Lack of defect propagation will give additional assurance that the corrosion mechanism has been arrested for the long term.

A summary of the licensee's extended post-repair eddy current inspection plan is shown in Table 3.3-1.

Table 3.3-1

Summary of Extended Post-Repair Eddy Current Inspection

DESCRIPTION	SCOPE	PROBE	TOTAL NUMBER OF TUBES BASELINE	TOTAL NUMBER OF TUBES AFTER 90 DAYS
Kinetic Expansion (6" Qual. Length)	- 15 Tubes with Previous Indications	8 x 1	15	15
	- 3% Baseline/OTSG	8 x 1	930	930
Wear (Inservice Tubes Adjacent to Unstabilized Plugged Tubes)	- 10 Peripheral/OTSG	0.540 SD	60	60
	- 10 with Defect in 15th, 10th, or 1st Span/OTSG	0.540 SD	120	120
	- 5 with .540 SD>3V	0.540 SD	60	60
Inservice (F40% TW)	- Defect Location - Full Length	8 x 1 0.540 SD	66	66
High Plugging Density	- 50 Full Length/OTSG	0.540 SD	100	100
Standard Inspection	- 3% Full Length/OTSG	0.540 SD	Completed in 1982	930
		TOTAL	1350	2300

CONCLUSIONS

Based on the above evaluation, the staff concludes that the eddy-current techniques developed and qualified for inspection of the OTSG tubing demonstrated the ability to reliably detect and size, with a high degree of sensitivity, the defects that were present in the tubing. The 100% tube inspection using these techniques, tube repair, and preventive tube plugging and staking of critical defective tubes give reasonable assurance that defective tubes have been identified and repaired or removed from service.

As discussed above, the staff further finds the post-repair extended ISI program instituted by the licensee acceptable. However, to ensure that the potential for primary to secondary leakage remains acceptably low, the following actions which the licensee has stated are to be implemented will be required by license conditions: (1) the licensee shall conduct extended post-repair eddy-current examinations, consistent with the inspection plan defined in Table 3.3-1, either 90 calendar days after reaching full power, or 120 calendar days after exceeding 50% power operation whichever comes first and (2) the licensee shall establish a baseline primary-to-secondary leakage rate as early as feasible in the steam generator hot functional test program. As early as feasible in post-critical operation, the licensee shall confirm the baseline primary-to-secondary leakage rate, and establish the minimum increase in such leakage rate which can reliably be measured (expected to be about 0.1 GPM). If leakage exceeds the baseline leakage rate by that minimum increase, the plant 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. Prior to restart after removing leaking tubes from service, the baseline leakage shall be re-established, provided that the present technical specification limit of 1.0 GPM is not exceeded.

3.4 Once Through Steam Generator Repair

3.4.1 Requirements of the Repaired Joint

To establish acceptability of the repaired OTSG for return to service, the licensee instituted a test program to demonstrate that the repaired joint would meet the original design basis. The following is a summary of the goals established by the licensee, which were used to qualify the repaired joint. These goals meet or exceed the comparable original licensing requirements for the steam generator, and are the only requirements which could be affected by the repair process.

a. Axial Load

The joint should be able to sustain the original design basis tensile load of 3140 pounds with no slippage between the expanded area and the tubesheet at an axial strain corresponding to this load. This criterion envelopes loads caused by all design basis accidents, including the main steam line break.

b. Thermal and Pressure Cycle Loading

The goal for the repaired joint is to maintain its load carrying and leak-tight capabilities for the remaining plant design life of 35 years, assuming design basis thermal cycling and transients. To demonstrate design life capabilities, the licensee instituted a multi-phase test program. The first phase of the test program, as discussed in Topical Report 008, Rev. 2, includes qualification testing to show capability to operate for the first five years, to justify restart. The second testing phase includes extended confirmatory tests, which will be completed subsequent to restart, to verify continued acceptability of the repaired joint for periods in excess of five years.

As a result, subsequent elongation of the tube is not sufficient to cause buckling.

The licensee conducted induced strain tests on test blocks to determine tube lengths before and after the expansion. Results show that the expansion process has a minimal effect on the overall longitudinal tube strain and as-fabricated preload induced strain. Measurements taken before and after expansion indicated maximum changes in longitudinal strain values of less than 0.04%. This relates to a maximum reduction in the tube pretension of about 16 pounds for all tubes except those which had lost pre-tension prior to being kinetically expanded.

The licensee has recently indicated that during the kinetic expansion process, an estimated 600 tubes lost pre-tension due to slight downward movement of as yet unexpanded tubes which had corrosion-caused full circumferential cracks. For tubes which have lost pre-tension, this would result in a maximum cold compressive load of 16 pounds. Although this deviates from the licensee's repair goal, it is insignificant compared to the 1025 pounds necessary to cause tube bowing and approximately 1500 pounds estimated to cause tube buckling. Therefore, reasonable assurance is provided that the repaired tubes are not in significant compression while cold and will not buckle during hot operations.

In addition to the mechanical tests discussed above to qualify the expanded joint, the effects of the explosive expansion on the tubesheet ligaments and welds were determined. The dimensions of adjacent holes in the tubesheet were measured before and after the expansion and compared. Results show that there is practically no effect on the diameter of adjacent holes due to tube expansion.

Full scale testing conducted by the licensee and its consultant in a steam generator at B&W's Mt. Vernon facility using strain gages and profilometry showed no degradation of the tubesheet ligaments.[6]

During preliminary and qualification testing, some candles fractured, i.e. blow throughs occurred, creating a concern that parts of the polypropylene cartridges could be left in OTSG tubes. To ensure that fragments of cartridges do not remain trapped or wedged in tubes, free flow air tests were conducted for each tube after the expansion. The final cleaning involved blowing felt plugs through each individual tube. The head and the tubesheet were manually wiped down and then the generator was flushed to remove any remnants of the repair process. Non-destructive and visual examinations revealed no tube deformation as a consequence of the fractured candles.

To determine the effect of the kinetic expansion process on the welded connections in the vicinity of the tubesheet, the licensee obtained strain and acceleration data during a kinetic expansion on a full scale steam generator at Mt. Vernon. One measurement was made at the junction between the inlet header and the tubesheet and the other at the welded location underneath the tubesheet. The strain gage measurements were taken at the two ends of a diametral row of 132 tubes. On the basis of these data, the peak stresses and stress intensities were calculated for the fatigue evaluation. The cumulative usage factor was determined to be 0.12 at these locations. Based on these data, the licensee concluded that the welded connections in the tubesheet/shell section will not be affected by the expansion process. Independent analysis by the staff consultant (Attachment 1) confirmed the licensee's calculation. Therefore, we agree with the licensee's conclusion.

STAFF EXHIBIT 2

NUREG-1019
Supp. No. 1

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

FACILITY OPERATING LICENSE NO. DPR-50

GPU NUCLEAR CORPORATION, ET AL

THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)

DOCKET NO. 50-289

STEAM GENERATOR TUBE REPAIR

AND RETURN TO OPERATION

1. INTRODUCTION

On August 25, 1983, the Nuclear Regulatory Commission issued NUREG-1019, its Safety Evaluation dealing with the steam generator tube repair and return to operation of TMI-1. The staff concluded that the steam generator repair program was acceptable, that appropriate General Design Criteria (GDC) had been met, and that subject to resolution of open items identified in Section 5.3, there is reasonable assurance that the health and safety of the public will not be endangered by operation of TMI-1 with the repaired steam generators.

Since issuance of NUREG-1019, the licensee has provided additional information in Revision 3 to its Topical Report 008 and in its letter of September 30, 1983, which included Revision 2 of TDR-406 (SGTR Guidelines) as well as GPU comments on NUREG-1019. Updated versions of Emergency Procedures 1202-5 (OTSG Tube Leak/Rupture)) and Emergency Plan Implementing Procedure 1004.7 (Offsite/Onsite Dose Projections) have been made available to the staff. In addition, by letter of October 25, 1983, the licensee submitted TDR-488, TMI-1 OTSG Hot Testing Results and Evaluation.

The purpose of this Supplement is to update NUREG-1019 by addressing the above information.

In the SER, we provided a description of the repair method which focused on the 22-inch kinetic expansions which are limiting in determining that tube pullout from the tubesheet cannot occur under design basis accident conditions as a consequence of severance of the tubes at the tube repair transition zone. By letter dated September 30, 1983, the licensee noted that our SER did not clearly indicate that tubes were repaired using both 22-inch and 17-inch kinetic expansions. We agree with the licensee's comments.

The majority of tubes were repaired using a 17-inch expansion because the vast majority of defects were located near the top of the upper tubesheet. The 17-inch expansions provided for repair of tubes with defects down to 11 inches from the top of the tubesheet while retaining the 6-inch qualification zone. Tubes with defects between 11 inches and 16 inches were repaired using 22-inch expansions. Because the 22-inch expansion, which is the limiting case, was already addressed, the information does not alter the conclusions in our SER.

3 EVALUATION OF REPAIR

3.1 Determination of Causative Agent(s)

In the SER, at the top of page 8, we stated that "The thiosulfate tanks have also been physically removed." By letter dated September 30, 1983, the licensee pointed out that the lines which connect the thiosulfate tank to the reactor coolant system have been physically severed and sealed but the tank has not been removed. This information does not alter our conclusion in the SER.

3.2 Examination and Repair of the Remainder of the RCS

In the SER on page 9, we stated that "all corrosion-affected sections in the waste gas system have been replaced." By letter dated September 30, 1983, the licensee noted that only sections of the waste gas system with unacceptable corrosion have been replaced. Piping with minor corrosion indications will be placed on an augmented inspection list. We agree with the licensee, because sections of this low pressure system in which the corrosion indications were not significant need not be replaced.

to a maximum of approximately 2.6 GPH during the third cooldown transient when tube tension was maximized. If a throughwall crack of sufficient length to propagate due to flow induced vibration existed, a minimum leakage rate of 23 GPH is predicted for the most limiting tube. Leakage rates for nonlimiting tubes are predicted at up to 80 GPH. Therefore, because of the low primary to secondary leakage rates during steady state and transient conditions, we find that there is reasonable assurance that the OTSG's do not contain critical size defects which could jeopardize the tube integrity when subjected to postulated accident design basis tube loadings.

In Attachment No. 7 to this supplement, our consultant indicated that the effect on crack propagation of residual stress fields in the formed tubes and the effect during heatup of tube bowing on vibrational charac teristics should be further addressed by the licensee. As discussed below, the staff finds that additional discussion of the topics by the licensee is not required.

In Section V.C.1.c of the Topical Report 008, Rev. 3, the licensee stated that a transition length between 0.125 and 0.25 inch would be a goal, with a minimum acceptable transition length of 0.1 inch for the kinetically expanded tubes. This transition length is significantly longer than the original as-fabricated transition length of 0.0625-inch stated in Reference 17 of the Topical Report. The increase in transition length will cause a corresponding decrease in strain in the transition zone. Therefore, we find that the residual stresses in the transition zone of the kinetically expanded tubes would be lower than in the as-fabricated condition. Consequently, we conclude the transition zones should not be more susceptible to failure than the original as-fabricated transition zones.

Tube bowing is only of concern during plant heatup because the OTSG tubes which are relatively thin reach temperatures in thermal equilibrium with the coolant more rapidly than the OTSG shell and thus expand proportionally more rapidly than the shell during heatup. By letter dated September 30, 1983, the licensee indicated that tubes which have experienced a loss of pre-tension may exhibit bowing deflections during heatup which may allow them to touch adjacent tubes. During heatup, stress in bowed tubes will remain compressive and, therefore, the loading will not accelerate crack propagation. Since there is little or no flow during heatup, little or no flow-induced vibration exists. Consequently, the excitation force is miminal during heatup, and the flow-induced vibration of these tubes should remain below the levels exhibited by nominal tubes at full power.

Based on the above evaluation the staff finds:

1. Cracks which are large enough, i.e., critical size, to propagate due to flow-induced vibration are readily detectable by ECT;

2. Cracks which are below the threshold of ECT detectability will not propagate under combined cyclic, flow-induced and thermal loadings;

3. The maximum crack size which will remain stable during a MSLB has been determined;

4. Through-wall defects which may propagate during operation can be detected well below the threshold size that could fail during a MSLB.

coolant is continuously removed by the letdown system purification ion-exchangers and the actual concentrations of sulfur should remain at less than 0.1 ppm. The staff, therefore, finds that there is reasonable assurance that the peroxide treatment has effectively reduced the sulfur contamination of the reactor surfaces to an acceptable extent. The potential for sulfur-assisted corrosion during subsequent reactor operation is further diminished by the measures described in the Safety Evaluation for monitoring the sulfate concentration and adding lithium to the coolant.

Subsequent to the desulfurization treatment, the licensee carried out a pre-critical steam generator hot functional test program, the results of which were reported in TDR-488, TMI-1 Hot Testing Results and evaluation. This program included a series of rapid cooldown tests of the steam generators from 530°F to 350°F. Axial stress on the steam generator tubes is at a maximum during cooldown. Therefore, through-wall circumferential cracks which may exist can be predicted to open wider and increase in leakage rate. The condenser exhaust was monitored for Krypton-85, which had been added to the primary coolant as a leak indicator using two calibrated independent analyzers and grab samples analyzed off-site. The primary-to-secondary leak rate was well below the Technical Specification limit during all phases of the pre-critical steam generator hot functional test. The rapid cooldown did not result in significant additional leakage, as indicated by Krypton-85 analyses and by analyses of the steam generator water for boron and other primary coolant constituents. We independently verified the licensee's analytical results, the method of calculation and the degree of agreement among the different measurement methods. We find that the licensee's leak detection methods will detect primary to secondary leakage at levels significantly below the shutdown limit of 0.1 GPM above background.

These results provide added assurance that the repaired tubes are leak-tight and the contaminant has been reduced to concentrations