REPORT OF THE THIRD PARTY REVIEW OF

THREE MILE ISLAND, UNIT 1, STEAM GENERATOR REPAIR

SUPPLEMENT 1

To

R. F. Wilson - Vice President, Technical Functions

GPU Nuclear

Prepared by

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date: 5/16/83

PURPOSE:

This report supplements the report of the Third Party Review of the TMI-1 steam generator repair dated February 18, 1983. The February 18 report contained findings, comments and recommendations to which GPU Nuclear responded. The GPU Nuclear written responses were further described and evaluated in a Review Group meeting of April 12 and 13, 1983. This supplementary report is intended to be the final report by this Review Group.

BACKGROUND:

The establishment and operation of the Third Party Review Group of the TMI-1 steam generator repair was previously discussed in the Review Group's Report of February 18, 1983. The report contained a conclusion concerning the adequacy of the steam generator repairs for safe operation of the TMI plant and findings, comments and recommendations about the steam generator repair and return of the plant to operation.

GPU Nuclear responded to the Review Group's report in an April 7, 1983 letter and submitted additional documents for the Review Group's use. The April 7, 1983 letter with its attachment 1 is included as Appendix A with this report. Also GPU Nuclear letter of April 4, 1983, which distributed additional documents for the Review Group's use, is Appendix B.

The focus of the Review Group's meeting on April 12 and 13, 1983 was the GPU Nuclear responses in Attachment 1 of Appendix A and the revised Safety Evaluation Reports 008 and 010. During this meeting, GPU Nuclear made presentations and responded to the Review Group's questions. The Review Group determined the adequacy of the GPU Nuclear responses and modified its conclusion.

At the close of this Review Group meeting, its revised conclusion was presented to Mr. R. F. Wilson, Vice President, Technical Functions, Mr. P. R. Clark, Executive Vice President, GPU Nuclear staff and an NRC staff representative.

CONCLUSION:

The February 18, 1983 report concluded that it would be premature to determine that, when all existing GPU Nuclear plans are completed, the Third Party Review would conclude that the results will be positive and will ensure that plant operation would be without increased risk.

Based upon the revised documents identified in Appendix A and B, and particularly the Safety Evaluation Reports 008 and 010, the Review Group modified its prior conclusion. The identified documents described substantial additional analyses and testing done by GPU Nuclear and its contractors on activities identified on the Review Group's prior report as incomplete including:

1. Post repair testing and hot functional operation of the systems.

2. Completion of analyses including leak before break and the contingency of multiple tube rupture.

3. Translation of analytical work such as leak before break and multiple tube rupture into useable plant guidance, procedures and training.

4. A conservative approach of power escalation after completion of repairs.

The Review Group now concludes that, upon satisfactory completion of the entire program as defined in the safety evaluations and as augmented by GPU Nuclear comments during and subsequent to the April 12 and 13 meeting, the TMI-1 plant can be operated safely with the repaired steam generators.

COMMENTS AND RECOMMENDATIONS:

The Review Group specifically addressed each of the GPU Nuclear responses of Appendix A and determined the acceptability of the responses. In the following section each of the GPU Nuclear responses to earlier Review Group comments and recommendations are addressed. The same format and numbering are followed as in Appendix A. The GPU Nuclear responses are not restated here. Refer to Appendix A.

A. Steam Generator Inspection

Recommendation 1 - The difference between the Review Group recommendation and the GPU Nuclear response concerned 17 tubes total in both steam generators. The 17 tubes have eddy current indications of less than 40 percent through wall and over two coils in extent. The Review Group had recommended plugging these tubes but GPU Nuclear responded that the 17 tubes would remain unplugged to provide information on crack growth between periodic eddy current examinations.

The Review Group considers the GPU Nuclear response to be satisfactory. It is noted that the indication size is substantially less than the critical crack size developed in Safety Evaluation Report 008 and thus would not present a safety risk.

Recommendation 2 - The Review Group previously recommended the plugging of all tubes with eddy current indications at or above the 15th tube support plate within three rows of the tube lane. GPU Nuclear responded that these tubes are or will be plugged except that tubes in the second or third row from the tube lane will not be plugged.

The Review Group was advised in the April 12 and 13 meeting that the extent of participation of tubes in the tube lanes was evaluated based upon prior eddy current examination of TMI-1 steam generators. Only tubes in the first row from the tube lane have shown indications associated with tube lane vibration. Based upon this specific information about vibration performance of these steam generators, the Review Group considered this response satisfactory.

Recommendation 3 - The GPU Nuclear response is satisfactory.

Recommendation 4 - The GPU Nuclear response is satisfactory.

B. Cause of Tube Cracking

Recommendation 1 - The Review Group had recommended earlier that GPU Nuclear implement corrective measures or verify their current programs for minimizing the ingress of all impurities (not just sulfur) into the reactor coolant system. The response addressed actions to protect from impurities. Although the GPU Nuclear actions are considered adequate for safety, the Review Group made the following comments concerning impurity control and related chemistry program.

Further Comment 1 - Identification of Sources - GPU Nuclear presented their plans for a chemical control program to be implemented over the next year or so. This program would identify paths for impurity input to the reactor coolant system and would impose controls to minimize input. We commend this effort and recommend that it be accelerated if possible.

Further Comment 2 - Resin Loss from Purification Ion Exchangers - We concur with GPU Nuclear's assessment that resin loss will be detected by pressure drop increases across the reactor coolant pump seal water inlet filters. These filters are of small pore size and receive a large fraction of the flow from the ion exchangers. GPU Nuclear's plans for changing out the purification ion exchanger post filters during the startup should permit identification of any physical problems and should reduce the potential for impurity input during the proposed peroxide cleanup. We concur that specific sampling and analyses for resin fines is not necessary.

Further Comment 3 - Control of Organics Input - Make-up water analyses presently specified will not detect organics. These materials can contain sulfur, chlorine, and other aggressive impurities which will be released to reactor coolant under heat and radiation. Also, carbonaceous material was found to be the major impurity near tube failure, and may have played a role in the failure which, in our ignorance, we do not understand. For these reasons, we recommend that specific analyses for organics be performed on the make-up water and other input streams to the reactor coolant system. GPU Nuclear indicated that they were in the process of purchasing a Total Organic Carbon (TOC) analyzer. This purchase should be expedited and analysis for TOC should be added to the Impurity Ingress Control Program. An initial guideline of 1 PPM TOC was suggested.

Further Comment 4 - Control of Recycle Streams - We understand that TMI-1 does not plan to recycle water or boric acid recovered from the Radioactive Waste Processing Systems, although the design permits it. We recommend that plant procedures be reviewed to assure that recycle is not an option until the recycle streams have been incorporated into the Impurity Ingress Control Program.

Further Comment 5 - Chemistry Specifications - With the exception below on sodium limit, the Review Group concurs with the chemistry control program planned by GPU Nuclear. Our exception on the sodium limit is not considered to be of significance to plant safety.

Further Comment 6 - Reactor Coolant Sodium Limit - GPU Nuclear has established a limit of 1.0 ppm for sodium in reactor coolant. The review Group suggests that this be reduced an order of magnitude to 0.1 ppm.

- With the removal of the sodium thiosulfate addition system, sodium is no longer likely to be present in the reactor coolant system.

- The purification ion exchangers are easily capable of maintaining this limit, even in plants which use sodium hydroxide to adjust the pH of reactor building spray (providing a possible source for sodium into reactor water).

- Sodium of 1 ppm will noticeably affect conductivity and pH, complicating the reconciliation of these measurements with lithium and boron concentrations.

- Lower sodium concentrations will be reflected in lower Na -24 activity in the coolant and waste streams.

- Sodium should be used as an indicator of potential problems with the purification ion exchangers or in one of the make-up streams (demineralized water, concentrated boric acid, etc.) to the reactor coolant system. As such, its limit should be only slightly above its nominal concentration in the reactor coolant system. The nominal or "background" sodium should be determined from plant experience.

C. Materials Application

Comment 1 - The Review Group previously cautioned GPU Nuclear to remain alert to the possibility that small cracks may have gone undetected in the reactor coolant system. Since then, additional inspections in the waste gas system and the pressurizer have located additional evidence of high sulfur concentrations and corrosive attack. Our previous note of caution continues to be valid although the Review Group does not consider further inspections to be necessary at this time.

Comments 2 and 3

The Review Group had commented previously on the sparcity of crack propagation rate data for the steam generator tubes, and the subsequent large extrapolation of the data required for the conditions of interest. Since then, more data were found to help substantiate GPU's analysis, although extrapolation is still required. One of the conclusions of the most recent GPU Nuclear analysis is that flow-induced vibrations may not play any role in propagating steam generator cracks. Nevertheless, if practical, for conservatism the Review Group still suggests that the long-term corrosion tests, which are designed to anticipate problems before their occurrence in the plant, should include a simulated flow-induced vibration loading. By simulating the actual loading, material conditions, and expected environment as closely as possible, these tests should help to warn of any problem that may have been missed in the analysis.

Comment 4 and Recommendation 1

The Review Group previously considered both the necessity or benefits of sulfur removal and the capability of the proposed peroxide flushing process for accomplishing sulfur removal. At that time we concluded that sulfur removal was not essential for the return of the plant to power. All available information indicated that the corrosion had stopped and that sulfur residues following completion of the repair would be comparable to other plants. The primary benefit of sulfur removal was intangible; the potential for reactivation of the corrosion from these surface residues would be reduced in proportion to the degree of effectiveness of removal. However, there was (and is) no quantitative measure of the potential for reactivation.

The proposed peroxide flushing process was at a very preliminary stage of development at that time. The Review Group expressed concerns over potential corrosion due to this process (alone and with the presence of sulfur), given that peroxide concentrations and exposure times exceeded current industry experience. We also pointed the difficulties of scale-up and the need for better understanding of the process. We could not assess whether the proposed process would accomplish its objective of sulfur removal without harm to the plant.

Since that time GPU Nuclear has generated additional data showing that sulfur concentrations are an order of magnitude higher than originally reported, that the sulfur exists as a sulfate at the surface of the oxide, and that nickel sulfide exists near the metal surface. They were unable to develop comparative data for other plants, however, so that we still do not know if these observations are unique to TMI-1. GPU Nuclear has also consulted with other experts concerning the desirability or necessity of sulfur removal, and has received strong recommendations to remove sulfur.

GPU Nuclear has also aggressively pursued development of the peroxide flushing process. Additional materials have been added to corrosion test programs to determine if peroxide flushing would have detrimental affects on core materials. None were found. The cause of anomalous data from the beaker test was determined. Loop testing has been performed which has defined the process parameters and documented the performance of the process. Ion exchange resins have been selected and tested for sulfate and ammonia removal. These results significantly reduce our prior concerns about the peroxide flushing process itself.

The Review Group continues to believe, however, that sulfur removal is not essential for safe operation of the plant, and that the costs and residual risks in uncertainty over peroxide flushing outweigh any benefit. We believe that the corrosion process is presently passive and will remain passive with good chemistry control even though sulfur residues will be available. We note that tests show 20-50% of the sulfur will not be removed by the process, so that sulfur residues will still be available after the flush. This process will be costly in time, chemicals, ion exchange resins, radioactive waste generation and man-Rems. In any complicated process, upsets can occur which could result in exposure of system materials to conditions not enveloped by testing. Finally, there is much about the reactions between peroxides and system materials which is not understood, so that (in spite of testing) there remains a risk that the process could be detrimental.

We therefore believe that peroxide flushing to remove sulfur is not essential to plant safety nor is peroxide flushing expected to have an adverse effect on plant safety.

Irrespective of our beliefs concerning the necessity for and questionable benefits of sulfur removal, we commend GPU Nuclear's efforts toward developing the peroxide flushing process. We have the following comments to offer.

Further Comment 1 - Peroxide Flush Process Control - Table B-1 of TR-010, Safety Evaluation of TMI-1 Reactor Coolant System Cleaning, presents the anticipated control parameters and and frequency of sampling for the process. The sampling frequency is too low to provide adequate control, especially during initiation and transients such as sulfate removal, residual peroxide/oxygen removal and ammonia removal. GPU Nuclear indicated that this Table is considered a minimal frequency, and that the procedure will address these concerns.

Further Comment 2 - Performance of Sulfate Removal Resin - The sulfate removal resin was tested for efficiency at a pH of 7.9, in contrast to the maximum pH of 8.5 for the process. Higher pHs will be more limiting, possibly resulting in lower sulfate removal efficiencies and chloride throw from the resin. It is suggested that analyses or testing be performed to evaluate the performance of the resin over the full range of process conditions.

Further, the potential for chloride throw from the resin should be evaluated and, if necessary, additional analyses performed to monitor chlorides during the process.

Recommendation 2 - The GPU Nuclear response is satisfactory.

Recommendation 3 - The GPU Nuclear response is satisfactory.

D. Removal of Sulfur Residues

This subject was covered under Section C above.

E. Stress Analysis of Steam Generators

Comments 1 through 4 - No GPU Nuclear response was required.

Comment 5 - Leak Before Break - The Review Group previously commented in part "In the transition zone, two additional stress states are superimposed on the axial tensile stresses. One stress state is caused by the axial load, and the other is caused by the interfacial pressure between the expanded tube and the tubesheet".

The sum of these three states is the stress state in the transition zone. As far as leak before break is concerned, the precise description of these states is not necessary. But if conclusions on main steam line break are drawn, they should include the transition zone.

For this reason, the Review Group recommended that a detailed stress analysis of the transition zone be made including the loading of the main steam line break.

Subsequent to the Review Group meeting, GPU Nuclear advised that they had completed the stress analysis of the transition zone, and the revising the stress report to include this analysis and that the analysis shows an acceptable stress condition. This resolves the Review Group's comment.

F. Steam Generator Leak Tightness After Repair

Recommendation 1 - GPU Nuclear establishes administrative limits on leakage which consider the threshold level of detectibility. The response resolved the Review Group's comment.

However, the additional analysis of leak before break conducted by GPU Nuclear and reported in Safety Evaluation Report 008 raised a further question. Warning before break is an important safety consideration in the operation of TMI-1 steam generators. It is important that flaws be detected by inspection or leakage before they can grow to critical sizes which would be unstable under normal operating conditions or accidents such as a main steam line break.

Usually, eddy current inspection and conservative tube plugging criteria are relied upon in demonstrating integrity for steam generator tubes. In addition to taking these steps, GPU Nuclear has performed a fracture mechanics analysis to show that the growth of flaws due to mechanical effects is slow (i.e., growth to a critical size takes many operating cycles) and that eddy current is capable of detecting flaws before they approach critical size.

Using the results of fracture mechanics analysis, GPU Nuclear also calculated leak rates through flaws. Their calculations show large margins in both time and flaw size between detection of leakage and failure of a tube during normal full power operation. The same calculations show that a flaw which is of a critical size for a main steam line break will leak approximately 0.1 gpm during normal full power operation.

GPU Nuclear has established an administrative limit of 0.1 gpm primary-to-secondary leakage for shutdown. On the basis of fracture mechanics and leak rate calculations, GPU Nuclear believes that this administrative limit assures leak before break for all normal and abnormal operating conditions including the limiting case of a main steam line break.

Comparative analyses by others show that fracture mechanics and leak rate calculations are sensitive to assumptions such as loads in steam generator tubes, flaw shape, flaw surface roughness and the existence of threshold stress intensities for crack growth. Because of these uncertainties in the analyses, the Review Group questioned whether the results of the GPU Nuclear analysis of leak before break had sufficient margin for the limiting case of a main steam line break.

Subsequent to the meeting GPU Nuclear has pursued this issue and advised they have come to the following conclusions:

- The GPU Nuclear study on crack preparation and their interpretation of the draft analysis done by others suggests that cracks will not grow or not grow rapidly as a result of flow induced vibration. Although growth rate is a function of the assumed threshold stress intensity, even the extreme case of no threshold revealed long time periods for crack growth to a critical size and therefore ample time for operator action to shut down the reactor prior to a tube failure either at power or during shutdown.

- The tube loads associated with the steam main line break case were calculated using a generic analysis for B&W plants. The assumptions used in this analysis are very conservative with respect to the particular plant parameters for TMI-1 and result in calculated tube loads substantially greater than would actually occur. When more realistic tube loads are taken into account, the critical crack size is estimated to be significantly larger and the corresponding leak rate is increased by approximately a factor of two. Thus GPU Nuclear concludes that because significant margin exists in the tube loading used to determine critical crack size, no further conservatisms need be added in designating administrative limits which take credit for leak before break.

On the other hand, the Review Group considered that eddy current inspection should give adequate warning of flaws which could become unstable before the next inspection for a main steam line break. If further assurance of leak before break is required for the main steam line break, then more sensitive leak detection techniques might be applied. GPU Nuclear should consider the following:

- Leak rate measurements during cooldown loads as a sensitive way to detect through wall cracks.

- Sensitive secondary-to-primary leakage measurements in steam generators during shutdown.

GPU Nuclear agrees that eddy current testing will identify cracks that could, before the next inspection, become unstable under main steam line break loading. As previously described the threshold for detectability for eddy current testing has been determined using notched calibration standards and laboratory-grown sulfur-induced intergranular cracks. This threshold has been found to be below the crack size that would rupture under main steam line break loads.

In addition, in response to Review Group suggestions, GPU Nuclear will record the condenser offgas activity data during cooldown and evaluate the feasibility of using this data for determining primary-to secondary leak rates during conditions of higher tube-to-shell delta T.

GPU Nuclear also plans to use secondary-to-primary bubble testing tube as one technique for locating leaking tubes whenever the plant is shut down in response to an increase to a leak rate 0.1 gpm or more. The high sensitivity of this measurement technique provides additional assurance that flaws that could become unstable before the next eddy current inspection will be detected.

The Review Group considers these actions satisfactory.

G. Plant Operations

Recommendations 1 and 2 - The GPU responses were satisfactory.

Attachments:
Appendix A - GPU Nuclear letter to Members, TMI-1 OTSG Repair Program Review Group; Subject: TMI-1 Steam Generator Repair; dated April 7, 1983 (with attachment 1 only)
Appendix B - GPU Nuclear letter to same; Subject: same; dated April 4, 1983

APPENDIX A

GPU Nuclear

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April 7, 1983
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Members, TMI-1 OTSG Repair
Program Review Group

SUBJECT: TMI-1 Steam Generator Repair

Enclosed for your review are several additional documents related to the TMI-1 Steam Generator Repair.

1. GPUN Responses to Third Party Review Findings and Recommendations, April, 1983.
2. TDR 400, Rev. 1. Guidelines for Plant Operation with Steam Generator Tube Leakage, DRAFT.
3. TDR 406, Rev. 0. Tube Rupture Guidelines, March 28, 1983.
4. TDR 417, Rev. 0. DRAFT. OTSG Leakage and Operating Limits.
5. TR-007, BAW-1760, Rev. 1. Kinetic Expansion Technical Report, March, 1983.
6. Handouts, NRC Briefing, April 5, 1983.

If you have any questions on these documents or any others we have sent you in the past, we will plan to discuss them at the April 12 meeting.

Sincerely,

Mary Jane Graham

M. J. Graham
Licensing Engineer

MJG:mt
Encl.

RECOMMENDATIONS AND COMMENTS
of the THIRD PARTY REVIEW

A. Steam Generator Inspection

Recommendation 1 - The Review Group recommends the plugging of all tubes which contain ID indications between the upper and lower tubesheet in both steam generators.

Eddy current indications have been detected between the upper and lower tubesheet. These indications are a mix of ID and OD and in general have an eddy current estimated depth of less than 40 percent through-wall. GPU Nuclear stated that they plan to leave some or all of these tubes with ID indications in service. About 60 tubes may be involved. The development of a plugging limit requires knowledge of eddy current measurement accuracy, defect growth rate and transient steady/state tube load conditions. Each of these factors has uncertainties which are not known with confidence. Although sufficient operating experience with other once-thru steam generators (OTSGs) would justify allowing the OD indications less than 40 percent through-wall to remain in service, the ID indications are most probably stress corrosion cracks and should be plugged.

Response:

Tubes with any ECT indication, ID or OD, in the 16th span and the bottom 4" inch upper tubesheet were both plugged and stabilized. GPUN plugged all tubes with eddy current indications of <40% T.W. (differential probe) and ≥3 coils (absolute probe) between the lower tubesheet and the 15th tube support plate.

Only tubes with ID indications of <40% TW and 2 coils or less circumferential extent were left in service. Engineering evaluations were performed on a tube-by-tube basis to ensure that the crack would not be expected to propagate to a size which would fail mechanically through the life of the plant. Tube stress evaluations are discussed in TDR 388 and Topical Report 008. There are 3 tubes in the "A" OTSG and 14 in the "B" OTSG with ID indications left in service. Allowing these 17 tubes to remain unplugged has been found acceptable, and will further provide information on crack growth during periodic ECT examination that would not otherwise be available.

Recommendation 2 - Tubes within three rows of the lane region and in the wedge-shaped region at the periphery which have OD indications at the 15th support plate or above, should be plugged as has been done in other OTSGs.

Response:

Tubes with any indications (OD and ID) from the bottom of the 15 TSP to 4" from secondary surface of the upper tubesheet have been plugged and stabilized. Two of previously plugged tubes not located in the lane region which have eddy current indications in the 16th span were replugged and stabilized to the 14th support plate.

Recommendation 3 - The eddy current inspection baseline performed after expanding tube ends should be extended to the full tube length on a selected number of tubes to detect possible deleterious effects of the explosive expansion. Tubes which contain defects less than 40 percent through-wall mentioned above should be considered as the sample set to detect evidence of defect growth or initiation as a result of the explosive expansion.

Response:

GPUN plans a post-repair eddy current inspection program which will include approximately 50 tubes full length. If results of this examination are acceptable, the 1982 full length examination of all tubes will be taken as the baseline. The 90-day ECT examination, and subsequent examinations will include a minimum of 3% of the tubes full length.

Recommendation 4 - The Review Group recommends selected non-destructive examination of seal welds and tube ends on the lower tubesheets to confirm the absence of any deleterious effects of the explosive expansion of the upper tube ends.

Response:

In early January selected dye penetrant examination was performed on "A" unit lower tubesheet. This informal inspection was documented by photographs. Interpretation of these photographs by Engineering disclosed no relevant indications. A formal dye penetrant inspection of selected tube ends and welds in the "B" lower tubesheet has also been scheduled.

B. Cause of Tube Cracking

Recommendation 1 - Although the Review Group believes that reduced sulfur forms were the most likely corrodant, we recommend that GPU Nuclear implement corrective measures to verify their existing programs for minimizing ingress of all impurities (not just sulfur) into the reactor coolant system. For example, failure analyses have consistently reported carbon as the major impurity on fracture surfaces, followed by sulfur. Carbonates in the presence of oxidants at high temperature can produce IGA and IGSCC of Inconel-600. Other contaminants (lead, mercury, phosphorus) can also induce IGSCC. Specific areas where attention is recommended are as follows:

a. Resin loss from the purification ion exchangers, since thermal decomposition of cation resins can release sulfur acids.

b. Contamination of the reactor coolant system makeup by greases, oils, organic solvents, or possibly resin fines form the makeup demineralizers or from water recovered from liquid radioactive waste treatment.

c. Contamination of the reactor coolant system or its makeup from peripheral, connected systems such as sulfur in the waste gas vent collection system.

Response:

a. Transfer of significant quantities of resin into the reactor coolant system is highly unlikely. The effluent from the purification ion exchangers (and makeup demineralizers) as well as coolant from the RC Bleed Tanks, Deborating demineralizers are filtered through makeup and purification filters (MU - FIA + MU - FIB) prior to return to the RCS. Resin beads, fractured resins particles and resin fines will be removed by these 1 micron filters. No further action is planned.

b. A Technical Specification has been written describing the development of a chemical control program. This control program which is scheduled to be prepared by Chemical Engineering for use at TMI-1 will deal with resins, greases, oils, organic solvents, and normal chemical additions.

Currently, the Chemical Technical Specification SP 1101-28-001 defines the acceptable chemical parameters for all water added to the primary system. Also controls are applied by the TMI-1 plant personnel for the addition of chemicals. These controls are a composite of B&W, EPRI and GPUN experience. Chemical additions require operations personnel to rack in chemical addition pumps when the additions are being made and to document these additions.

c. Wipe samples on peripheral connected systems (spent fuel cooling system) building spray system, decay heat removal system, chemical addition system, core flooding, fluid block, nuclear nitrogen, and waste disposal-liquid bleed tank portion) have been performed to detect other potential contaminated areas. Liquid samples from systems have been analyzed to determine presence of sulfur. Action is being taken to flush system tank and dead legs from normally isolated and stagnant locations. Periodic sampling of these systems for sulfur will be performed and chemistry will be controlled. Corrosion of portion of the waste gas piping attributed to sulfur has been identified. In areas where stress corrosion cracking of the piping was observed, this portion of the piping was replaced. Need for removal of surface sulfur from the remainder of this system is being investigated.

C. Materials Application

<u>Comment 1</u> - Although nondestructive and limited destructive tests were carried out in looking for possible stress corrosion cracking in the rest of the reactor coolant system and none was found, such cracks tends to be very tight and are indeed very difficult to detect. Yet all the ingredients to generate such cracks were apparently present; i.e., sensitized and stressed susceptible materials (due to welds), and presumably a thiosulfate-contaminated aqueous environment. Therefore, GPU Nuclear should remain alert to the possibility that small cracks may, in fact, be present in susceptible components of the reactor coolant.

<u>Response</u>:

It should be noted that most of the cracks identified at TMI-1 were in vapor spaces, and none in submerged, circulating portions of the system. Considering these differences in environment, large portions of the reactor coolant system would not be expected to have cracking. The tests performed as part of Task 7, and subsequently as part of a system review after the PORV and waste gas system defects were found, examined equipment representative of a range of conditions. However, the program concentrated on environments and materials which were the most susceptable. Included were ultrasonic and dye penetrant testing, as well as visual and, in some cases destructive testing. With the large number and variety of tests performed, the presence of cracking in additional areas should have been identified. Based on the test program, GPUN concluded that the likelihood of cracked components remaining in service at TMI-1 is no greater than at any operating plant.

Comment 2 - Through a fracture mechanics analysis, GPU Nuclear arrived at a alternative conclusion that steam generator tubing defects below a certain size range will not propagate due to flow-induced vibrations. The analysis which led to this conclusion depends on a large extrapolation of a limited crack-propagation-rate data base. This makes it hard to substantiate a firm conclusion.

Response:

More data is now available, as summarized in TDR 388, Tube Stress Report. The conclusions drawn are thus more firmly substantiated.

Comment 3 - The long-term corrosion tests, which are designed to anticipate problems before their possible occurrence in the plant, do include most of the right ingredients and should be very helpful. However, they do not include a flow-induced vibration type of loadings which could make a significant non-conservative difference in the results once a crack is initiated.

Response:

It is correct that flow-induced vibration loading was not simulated in the corrosion test program. Based on the evaluation documented in TDR 388, this does not appear to introduce a significant non-conservatism.

Comment 4 - Cleaning of the residual sulfur in the system poses a dilemma since even the laboratory-scale beaker test results are apparently not fully understood at this time. Some of the test results have shown erratic cleaning and peroxide consumption. The time required to remove sulfur is greater than expected, and the peroxide concentrations are much greater than previously used to encourage crud removal in other nuclear plants.

Response:

The laboratory scale beaker tests contain few unknowns at this time. Peroxide consumption, although lower than that experienced in the Westinghouse loop tests is predictable. Cleaning results, after the contribution from atmospheric sources was defined and appropriate total sulfur analyses developed, are also predictable. Significant progress has been made in quantifying polypropylene behavior and a final test employing a representative, plastic-coated, sulfur-containing tube is in progress. This test should give a more representative estimate of required cleaning time.

The beaker corrosion tests conducted at Battelle as well as the loop test at Westinghouse provide confidence that corrosion damage will not occur due to exposure to the peroxide cleaning solution at the concentrations to be employed in the cleaning.

Recommendation 1 - If GPU Nuclear pursues development of the peroxide process for sulfur removal, the effect of greater than 400 hours exposure of core materials (e.g., Zircaloy) to hydrogen peroxide at the anticipated concentration and pH conditions should be included in the test program. Also, scaled-up tests should be done in metal systems at least somewhat more closely simulating the reactor cooling system environment.

Response:

In order to characterize the hydrogen peroxide process under simulated plant operating conditions, a pilot simulation was run using a recirculating autoclave system at the Westinghouse Research & Development Center. Temperature, pressure and system chemistry was identical to that planned for the actual in-plant use of the process; the pilot loop was run for 500 hours. Actual OTSG tubes were exposed in three forms: expanded tubes, unexpanded tubes, and C-rings with the ID stressed in tension. Seven materials representative of primary system materials were also exposed in the stressed condition. These materials were: Zircaloy, 410, 17-4 pH and 204 stainless steel, Inconel X-750 and 308 S.S. and Inconel 82 weld metal.

Conclusions drawn from the pilot simulation are: 1) No attack due to peroxide exposure was observed on any specimens; 2) Continuous hydrogen peroxide consumption occurred throughout the operation; and 3) System pH is controllable within specified limits.

Recommendation 2 - To gain experience in operating the unit while keeping the risks as low as possible, GPU Nuclear should consider substantially extended operation at low power during a slow and deliberate power escalation the first time the plant goes critical. Although we do not have an analytical basis for a specific duration, a hold period of perhaps a month or more at 40 percent power should be considered before the Loss of Feedwater/Turbine Trip test is performed. This might be followed by another month or more at 70 percent power before final escalation to 100 percent power. Also, this first power operation might better be terminated by a normal cooldown procedure rather than by the Overcooling Control Test which is currently planned. This last test could be done during subsequent operations.

Response:

GPUN agrees that power escalation should be slow and deliberate and to that end the test program allows one (1) week for zero power physics testing, another week at a very low power level for natural circulation testing, a slow increase to approximately 40% power which will take approximately four (4) more days, a stop at 40% power for approximately one (1) week to perform various tests, i.e., ICS tuning, core power distribution, incore thermocouple testing and OTSG leak rate determination. At the end of this testing it is planned to perform the loss of feedwater/reactor trip test, after which the plant will be brought back to 48% power for approximately four (4) weeks to allow for OTSG leak rate monitoring, operator familiarity and surveillance testing. During this four (4) week period, the test group will review and confirm test data and make any adjustments or repairs that are necessary.

One of the tests to be run during low power test phase of the program is the Natural Circulation Test. This test is a mandatory requirement and its sequence in the program is dictated by the NRC. During the Natural Circulation Testing, the emergency feedwater will be in manual and under control of the operators. The operator training acquired in this program and the approximately four week time period from criticality to the loss of feedwater test are expected to provide adequate plant familiarization for the operators. It must be recognized that during the plant startup, very close communication with the plant supervisory staff will be maintained and if at any time they feel they need more time between evolutions, that recommendation will be evaluated and a decision made.

The reasons for performing the overcooling control test at this point in the power escalation test program are:

1. Test should be run at a low value of decay heat.

2. Test should be combined with the operation of the emergency feedwater system in order to reduce the number of thermal cycles to the OTSG's.

3. This is the latest point in the test program that emergency feedwater will be utilized.

It is felt that the performance of the test at this point in the program satisfies the above criteria as it will be run after the loss of feedwater test which will require the use of the emergency feedwater system and will only require one thermal cycle to the steam generators. Therefore, we would propose to perform this test where it is presently scheduled. A formal management review will be performed at the end of this four (4) week period before proceeding to the 76% power plateau. Once at 76% power, we plan to stop and allow the plant to come to equilibrium condition and provide another four (4) week period for OTSG leak rate monitoring, surveillance testing and operator familiarty.

Recommendation 3

GPU Nuclear should consider the possibility of deliberately running one steam generator at a higher power than the other during the first power escalation hold periods. The objective of this would be to force any possible operating problems to occur in the higher power steam generator before such problems affect both units. We understand a substantial power unbalance between loops is within the range of plant design. We recognize, however, that this recommendation may involve other operating considerations which would have to be weighed before a decision could be made.

Response:

We have reviewed the possibility of operating with one steam generator at a higher power than the other during startup and have decided that the possible benefits that might be derived from this are outweighed by the disadvantage of operating the plant in an abnormal configuration. Specifically, the mismatch can only be implemented by operating a single pump in one loop, which would cause mismatched RCS flow, unbalanced feed flows and different levels in each steam generator. This would mask abnormal indications of plant response during transients. This abnormal configuration conflicts with the intent of conducting the startup in a slow, deliberate manner to gain operator experience under normal operating conditions.

D. Removal of Sulfur Residues

Response:

No recommendations or comments.

E. Stress Analysis of Steam Generators

Comment 1 - Effect of cooldowns and cold shutdowns on strength of tubes. The Review Group has found no evidence to suspect that the cooldowns, starting from the one in April 1979, have subjected the tubes to stresses that are higher than the design stresses. During the two cold shutdown periods, the one after April 1979 and the other after September 1981, the tubes have been in a state of tensile stress, although, from the information received, the levels of these stresses could not be determined with any accuracy. An indication of the stresses comes from another plant in which the gap between ends of a broken tube translated to a tensile stress of about 4000 psi, well below the allowable stress. Since creep at the cold shutdown temperatures and stresses should be insignificant, the Review Group concludes that the cooldowns and cold shutdowns have not left the tubes in a weaker state than in any other OTSG.

Response:

GPUN agrees with this comment.

Comment 2 - Effect of repair on strength of tubes - The explosive expansion of the tubes could affect the stress levels, if the process would change the strength or some dimensions of the tubes. From the information that the Review Group has received, from the reports on the qualification tests, and for the statements made in publications issued by the tube expansion contractor, the Review Group concludes that the repair process is not expected to affect significantly the stress levels in the tubes in the restart and subsequent operation periods.

Response:

GPUN agrees with this comment.

Comment 3 - Effect of environment on strength of tubes - Since the Review Group has found no indication of significantly higher stress levels than in normal OTSG tubes, it concludes that a corrosive environment, and not abnormal stress levels, must have been responsible for the appearance of the cracks. It concludes also that, if the environment is made more favorable, then, at the same stress levels, cracks should not propagate in the OTSG tubes if they were in the same condition as in any other OTSG.

Response:

GPUN agrees with this comment.

Comment 4 - Effect of defects on strength of tubes - The Review Group recognizes that, at this time, the tubes probably have some small defects that were not detected by the eddy current tests and were not eliminated by the repair. These defects present the potential for leaving the tubes in a weaker condition than those in a normal OTSG. Both GPU Nuclear and the Review Group have addressed the question of what could happen to these small defects during the restart and operation.

Response:

Considerable additional work has been done in this area since the TPR findings were formulated. This work is documented in TDR 388.

Comment 5 - Leak-Before-Break - In the event that a defect does grow in the form of a crack through the wall, and eventually breaks the tube, the Review Group has addressed the following question: Will the fracture process ensure Leak-Before-Break? The Review Group concludes that the answer is positive and offers the following arguments. EPRI Report NP-2399, dated May 1982, states that small defects at the ID of a tube have about the same fatigue-crack growth rates toward the OD as along the circumference, provided that the stresses throughout the wall are axisymmetric and tensile. Since the free span (away from tubesheets), the relevant stresses in the tubes are axisymmetric and tensile throughout the wall, then a fatigue crack in the free span will break through the wall and produce leakage before it grows around the circumference and breaks the tube, thus ensuring Leak-Before Break. (Note that the question of threshold detectability is treated in Section 5. Steam Generator Tightness After Repair.)

However, in the expansion transition zone of the tube, in the vicinity where the expanded tube diameter changes to the nominal diameter, two additional stress states are superimposed on the axial tensile stresses. One stress state is caused by the bending stresses in the transition zone, that are produced by the axial load, and the other is caused by the interfacial pressure between the expanded tube and the tubesheet. GPU Nuclear has performed calculations of these stress states, and they show a rapid decay from the transition point. This means that the transition zone lies well within the tubesheet. The important point is that these stresses introduce compression within the tube wall in the transition zone, which, according to EPRI NP-2399, can make fatigue cracks grow faster along the circumference than toward the OD. This means that in the transition zone the tube could break before it leaks. However, since the break would occur within the tubesheet, the end of the broken tube would be restrained in the hole, and a controlled leak would result. Based on the argumant, the Review Group has concluded that such leaking tubes, broken in the transition zone, could be detected and removed from service before an excessive reactor coolant leak rate results.

Response:

Considerable additional work has been done in this area and is documented in TDR 388. Included is an evaluation of crack behavior in the transition region.

The superposition of operating loads (1107) on the bending stress from fabrication almost eliminates the compression. The value of the residual bending stress from fabrication that was used in TDR 388 was conservative because it reflects as-built conditions. The as-repaired configuration should result in a lower residual stress because the candle ends were specially chamfered to achieve more gradual blend radii.

As said in the TDR, MSLB loads superimposed on the residual fabrication bending will cause the material to flow and thus to release the residual bending by formation of a plastic hinge.

The higher tension resulting from superposition will drive the crack through-wall rather than circumferentially which is the preferential direction for leak detection.

The differences between GPUN's evaluation and the reference cited (NP-2399) by the TPR may in part be due because the reference addresses the interaction of growing stress corrosion cracks with different stress fields. We have no growing stress corrosion cracks.

F. Steam Generator Leak Tightness After Repair

Recommendation 1 - The Leak-Before-Break evaluation should include consideration of a realistically high steam generator leak rate and this rate should be consistent with the administratively imposed controls on operation with leakage through the steam generators.

GPU Nuclear did not explicitly treat this operational leak rate in the Leak-Before-Break evaluation. This evaluation is sensitive to the threshold detectability of a leak from a tube defect. This threshold is in turn dependent upon the total leak rate from all sources through the steam generators during operation.

Response:

In TDR 400, GPUN addresses operational leakage limits based on the minimum leakage anticipated from a critical-size crack. This leakage is above the threshold for detectability of plant monitoring. The operational leakage limit also addresses the possibility of leakage from other sources by providing for measuring a baseline leakage at the beginning of operations. Limits are set on leakage changes above the baseline.

G. Plant Operations

Recommendation 1 - Inspection of the waste gas system vent header branch piping identified a pipe cracking problem in the heat affected zone of butt welds. The failures were attributed to sulfur contamination. From a safety and operational point of view we recommend this inspection be expanded to include the waste gas decay tanks and associated isolation valves. Reactor building isolation valves associated with this system should also be included.

Response:

The waste gas decay tanks and their associated isolation valves need not be inspected since they are of carbon steel construction and not susceptible to intergranular stress corrosion attack. However, one of the decay tanks was visually inspected without any significant damage observed.

The welds of the containment isolation valves for the WDG system (WDS-V3 and V4) were inspected and no other failures or indications were identified other than those originally identified.

The piping in the vicinity of WDG-V4 and the WDG-V4 itself were replaced.

Recommendation 2 - We recommend that during the "slow" approach to power escalation after repairs (Recommendation C.2 above), a planned program of operator training in the plant be conducted.

Response:

Time is allocated in the Startup Test program for training, and the plant staff intend to provide operator training during the various phases of power escalation and testing. The training program is being coordinated between the plant staff and training department.

Inter-Office Memorandum

GPU Nuclear

Date April 4, 1983
E&L: 4878

File: 2252.6.9

Subject TMI-1 Steam Generator Re, air

To Members, TMI-1 OTSG Repair Program Review Group

Location

Enclosed for your review are several additional documents related to the TMI-1 Steam Generator repair:

1. Topical Report 008, Rev. 2. Safety Evaluation for Return to Service, dated March 29, 1983.

2. Topical Report 010, Rev. 0. Safety Evaluation for RCS Cleaning, dated March 3, 1983.

3. TDR 388. Mechanical Integrity Analysis of TMI-1 OTSG Unplugged Tubes, Rev. 2., dated March 30, 1983.

Additional documents will be sent in the near future for your use prior to the April 12 meeting, i cluding responses to your earlier comments and recommendations.

M J Graham

M. J. Graham
Licensing Engineer

MJG:mt
Encl.

cc: R. M. Milford