REPORT OF THIRD PARTY REVIEW OF

THREE MILE ISLAND, UNIT 1, STEAM GENERATOR REPAIR

TO

R. F. Wilson - Vice President, Technical Functions

GPU NUCLEAR

Prepared by

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date: ~~February 18~~, 1983

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CONTENTS

PURPOSE:

This is the report of the Third Party Review Group established by GPU Nuclear to provide a timely, independent and objective safety evaluation of the activities being conducted to repair and return the TMI-1 steam generators to operation.

An interim report of the Review Group was issued September 27, 1982. It reported the evaluation of part of the TMI-1 Steam Generator Repair Program - that relating to the safety of conducting the proposed repair of the steam generators while the plant is in a cold shutdown condition, including the effects of the repair on the steam generators and on the remainder of the TMI-1 plant. The Interim Report is Appendix A to this report. Appendix A is considered an integral part of this report and its content is not repeated here.

BACKGROUND:

On April 12, 1982, R. F. Wilson of GPU Nuclear established a Third Party Review of the TMI-1 steam generator repair program. Appendix B is the GPU Nuclear letter that established the Review Group. A Charter was supplied which defined the purpose, scope, membership and operations of the Review Group. During the first meeting of the Review Group on April 23, 1982, the Review Group discussed the Charter and suggested changes. The revised Charter dated April 27, 1982 is Appendix C. The evaluations of the Review Group have been conducted in conformance with Appendix C.

The membership of the Review Group, selected by GPU Nuclear for expertise in the following technical specialties is:

Specialty	Name	Affiliation
Steam Generator Design	E. J. Wagner	Burns and Roe, Inc.
Chemistry	D. J. Morgan	Pennsylvania Power & Light
Materials	R. W. Weeks	Argonne National Lab
Stress Analysis	A. Kalnins	Lehigh University
Safety Analysis	W. H. Layman	EPRI - NSAC
Plant Operations	S. A. Holland	Duke Power Co.
Non-Destructive Examinations	S. D. Brown	EPRI - NDE Center

E. G. Wallace of GPU Nuclear is a non-voting member who serves as liaison with GPU Nuclear and was assigned as Secretary. All members have been present and participated in all meetings of the Review Group. In its April 23, 1982 meeting, the Review Group elected Mr. E. J. Wagner to be Chairman.

As can be seen from the expertise of the membership, GPU Nuclear carefully selected the Review Group to obtain a broad range of competence in technical specialties important to the steam generator repair. In addition to obtaining competence, the formation of the Review Group focused on independence and objectivity. The individual members are not responsible for the performance of the design or development activities involved in the steam generator repair; nor are the organizations with which they are affiliated performing the steam generator repair activities.

It should be noted that the organizations with which the members are affiliated have many competent personnel and other resources applicable to the subject of this Review Group. The members have used these resources to conduct their evaluations.

The members also availed themselves of personal contacts with other experts in universities and laboratories. Notwithstanding, the members acted as independent individuals on this Review Group. Neither their individual statements nor their contributions to the reports of this Review Group are intended to represent the opinions or conclusions of the organizations with which they are affiliated or of the other experts they consulted.

The evaluation was conducted using reviews of pertinent documents, submittal of written questions to GPU Nuclear, written responses by GPU Nuclear, review of specialty topics by individual members, presentations by cognizant GPU Nuclear or contractor personnel, Review Group meetings and executive sessions of the Review Group members only. Appendix D is a bibliography of the documents supplied to and used by the Review Group. Appendix E is a compilation of the members' written questions and the GPU Nuclear responses.

Full day meetings of the entire Review Group were held on April 23, May 20 and 21, August 24 and 25, and December 7, 8 and 9, 1982. At each meeting, GPU Nuclear and its contractors presented status and plans, answered oral questions from the Review Group and were given suggestions and comments by the Review Group. The comments of the Review Group were summarized with senior GPU Nuclear Management at each meeting. These summaries were made to Mr. R. F. Wilson, Vice President, Technical Functions in the first three meetings and to both him and Mr. P. R. Clark, Executive Vice President at the December 9, 1982 meeting.

NRC staff members attended and participated in the Review Group meeting of August 25, 1982. GPU Nuclear, contractors and Review Group members answered questions from the NRC staff members. The summary meeting of

December 9, 1982 was attended by NRC staff members and a representative of the State of Pennsylvania. They heard oral presentations of the findings of the Review Group made by each of the members in his field of expertise. Questions from all attendees were responded to by Review Group members.

The principal foci of this review were the two Safety Evaluation Reports prepared by GPU Nuclear. The first treated only the repair of the steam generators. It is identified as item 10.1 in Appendix D. The Review Group's evaluation was reported in Appendix A and is not repeated here. The second Safety Evaluation Report treated the return of the TMI-1 plant to service after repair of the steam generators. It is item 17.9 of Appendix D. The following evaluation addresses this second Safety Evaluation Report and the activities it embodies through December 9, 1982.

CONCLUSION:

The Third Party Review Group recognizes that work has progressed to conclusions in many areas supporting the evaluation of safety, such as the cause of the steam generator tube cracking, corrective actions, and actions to prevent recurrence. Other supporting efforts have not yet neared completion. Recognizing the status of all the activities supporting the Safety Evaluation, safe operation of the TMI-1 plant after repair of the steam generators will be dependent on several remaining major activities:

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.

GPU Nuclear work in these areas has not reached the point where the results can receive a final safety evaluation. It would be premature to evaluate that when all existing GPU Nuclear plans are completed that the Third Party Review would conclude that the results will be positive and will ensure that plant operation would be without increased risk.

GPU Nuclear has performed sound technical work and the Third Party Review Group endorses the planned GPU Nuclear program to insure safe operation. We believe that there is a high probability that when the GPU nuclear program embodied in the Safety Evaluation Report is completed, that the TMI-1 plant with steam generators repaired in accordance with present plans will be able to operate without undue risk.

FINDINGS, COMMENTS AND RECOMMENDATIONS:

A. Steam Generator Inspection

Finding 1 - GPU Nuclear and their eddy current inspection contractor have done a thorough job of improving and qualifying eddy current inspection techniques for detecting the primary-side cracking condition which exists in the TMI-1 steam generators. A minimum detectable defect condition has been identified based on inputs from stress analysis and fracture mechanics considerations. Eddy current standards have been fabricated which duplicate the desired detection limits and eddy current system optimization studies have been

conducted which have identified test frequency, coil fill-factor and instrument sensitivity settings. Confirmatory detection studies have also been conducted using intergranular stress corrosion cracking (IGSCC) specimens which were fabricated in the laboratory. This qualification program, and the extensive inspection performed on both steam generators with the qualified techniques, provide confidence that the primary side cracks above the defined minimum detectable size have been detected.

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

Eddy current indications have been detected between the upper and lower tube sheet. 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.

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.

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.

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

B. Cause of Tube Cracking

Finding 1 - The Review Group is in agreement with the failure scenario presented by GPU Nuclear in Section II.D.2 of the Safety Evaluation Report for return to service Appendix D, item 17.9.

The probable mechanism for the tube cracking was IGSCC or possibly stress-assisted intergranular attack (IGA) resulting from exposure of the tube ID to sulfur and its lower oxidation states during cold shutdown with the reactor coolant system partially drained. The most plausible input corrodants were sodium thiosulfate which probably leaked from the containment spray system into the reactor coolant system during the extended shutdown and oxygen which was presented in the gas phase of the partially drained reactor coolant system.

The concentration of the damage at the upper tubesheet region can be explained by two mechanisms, both of which may have been present. Sodium thiosulfate or corrosive sulfur compounds from the oxidation of sodium thiosulfate may have been concentrated at the liquid-gas interface which was in the vicinity of the upper tubesheet by evaporation following

fluctuations in the reactor coolant level. Alternatively, the solution near the interface would have had better access to oxygen in the gas phase, resulting in a higher concentration of corrosive species in the water near the interface.

The spread in damage may be the result of fluctuations of reactor coolant level and/or the concentration of oxygen in the solution. The oxygen concentration would be expected to decrease with increasing distance from the interface.

Evidence supporting this scenario is as follows:

a. Destructive examination showed that the cracking was intergranular, ID initiated, and circumferentially oriented. This indicates a strong tensile stress component on the cracking mechanism. Since major tensile stresses in the axial direction are expected only during cool-down and cold shutdown, this implies that the cracking occurred under these conditions.
b. Areas of IGA were found on tube samples from the generator, both on the ID surface and in some cases on metal bordering the fracture surfaces. As GPU Nuclear pointed out, the surface IGA may have been generated by pickling during tube fabrication.
c. The tubing was sensitized, a condition known to be susceptible to IGSCC when exposed to the lower valence sulfur oxides (polythionic acids). These tend to be stable only at low temperatures, further evidence that the cracking may have occurred at low temperature.
d. Cracking has been induced by GPU Nuclear on stressed Inconel-600 specimens exposed to sodium thiosulfate and oxygen at low temperatures.

e. The damage was widespread and extensive on both generators. This implies that either relatively large quantities of corrodants or a highly corrosive species would be required. Sodium thiosulfate and oxygen appeared to be the only known corrodants which may have been present in sufficient quantities to produce the extent of attack observed.

Recommendation 1 - Although the Review Group believes that reduced sulfur forms were the most likely corrodant, we recommend that GPU Nuclear implement corrective measures or 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 from 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.

C. Materials Application

Finding 1 - GPU Nuclear is to be commended for their materials work to date associated with the steam generator repairs. They had two independent metallurgical failure analyses performed in addition to their own efforts, and they were diligent in resolving minor differences in the findings. They assembled

a separate panel of corrosion experts to advise them, and have pursued their advice by inspecting other areas of the reactor coolant systems considered especially susceptible to similar attack by establishing a test program to determine the feasibility of cleaning the residual sulfur out of the system, and by establishing short and long-term corrosion testing programs to help in identifying any residual corrosion problems prior to their possible occurrence in the plant. In general, they are doing a thorough job in attempting to assure the future reliability of the materials in the system.

In reviewing the foregoing efforts by GPU Nuclear and their contractors, the Review Group expressed the following notes of caution:

Comment 1 - 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 tend 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.

Comment 2 - Through a fracture mechanics analysis, GPU Nuclear arrived at a tentative 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.

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 loading which could make a significant non-conservative difference in the results once a crack is initiated.

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.

In recognition of these uncertainties, the Review Group recommends the following.

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.

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.

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.

D. Removal of Sulfur Residues

Finding 1 - GPU Nuclear has proposed that sulfur residues known to be present on reactor coolant system surfaces be removed by a hydrogen peroxide flushing process to eliminate the possibility that these could produce future attack. The Review Group recognizes the logic that, if sulfur caused the problem, it is conservative to remove it under controlled conditions rather than possibly letting it come off under uncontrolled conditions. However, the Review Group is not convinced that this sulfur removal process will be of much benefit. Further, we do not think it is essential for the return of the plant to power. This is based on the following considerations:

a. Eddy current testing indicates no progression of cracking since it was initially observed.

b. Although the data base is very limited, it appears that present sulfur concentrations on TMI-1 surfaces are comparable to those at other plants of this type.

c. The repair activities being conducted in the OTSG will result in substantial flushing and wiping of the tube surfaces (the major portion of the reactor coolant system surface area) as well as the upper head of the OTSG. These are the regions of maximum expected surface sulfur concentrations.

Finding 2- The sulfur removal process has not been developed to the point that the Review Group can assess whether it will accomplish its objectives. Review of the available information raised the following concerns:

a. Although beaker tests have shown that peroxide additions can oxidize and solubilize sulfur, scale-up from beaker testing to predict conditions under which this will work on the plant may be difficult. In contrast to the glass beaker, the metallic surfaces of the actual plant with their burdens of corrosion products could be expected to cause different process dynamics and results.

b. The beaker tests have shown some anomalies on peroxide concentration change and sulfate response which indicate the need for more understanding of the reactions occurring.

c. The process as currently envisioned will be protracted (about 400 hours) and will require significant cleanup efforts to remove the large concentrations of ammonia required for pH adjustment of the reactor coolant.

d. The concentration of peroxide envisioned (about 25 ppm H_2O_2) is significantly higher than that typically used in other nuclear plants to enhance crud removal. This higher concentration may require a more careful assessment of materials compatibility than presently given, especially in light of the long flush times expected.

e. These flushes are known to put significant amounts (1-2 ppm) of nickel into solution and to cause crud bursts. These effects should be considered.

f. Finally, although the corrosion testing program is considering this, there is concern that peroxide flushing could produce the corrosion that we are trying to prevent.

Recommendations are included in Section C. Materials Application of this report that take cognizance of these findings.

E. Stress Analysis of Steam Generators

Finding 1 - The main question that the Review Group addressed in this area is the following: did the events that occurred after the initial shutdown, including the repair, affect the OTSG tubes of TMI-1 in such a way that, during restart and subsequent operation, the stress levels can be expected to be significantly higher, or the strength of the tubes significantly lower, than those in a normal OTSG? From the information received, the answer appears to be negative, with a minor qualification regarding the undetected defects that are left in the tubes (see Comment 4 below). Therefore, the Review Group concludes that the integrity of the tubes in the OTSGs of the TMI-1 has not been significantly reduced to influence the restart and subsequent operation of the plant. This conclusion is based on the following comments:

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.

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 from 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.

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.

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.

GPU Nuclear has performed an extensive analysis of the possibility that these small defects could propagate as fatigue cracks if the tubes were subjected to flow induced vibration. This analysis concludes that the cracks will grow at a stable growth rate within the tube wall, and that the time that is required for a crack to reach the OD of the tube is longer than the lifetime of the OTSG. GPU Nuclear has calculated also the leakage rates through through-cracks and concluded that they are high enough, so that leaking tubes can be detected and taken out of service, before the cracks become unstable.

The Review Group finds these results reassuring, but has some reservations with regard to the limited data base for the crack propagation rates that has been used in the analysis. Comment 2 on Materials Application refers to this uncertainty. The Review Group recognizes that better data are not available at this time and acknowledges the difficulty of making firm conclusions in this case on fatigue-crack growth rates and their stability. The Review Group's reservation on this matter is somewhat mitigated by observing that the accuracy of the predicted growth rates, even though important, may not be as crucial for the present purpose as the analysis of the manner in which the cracks may grow. The Review Group recognizes

the possibility that among the undetected defects there may be some that are large enough to break through to the OD and propagate along the circumference in a stable manner, with the potential of breaking the tube when the crack becomes unstable. Such defects may have simply escaped detection and are the primary candidates for breaking some tubes. The crucial question is whether or not these tubes will leak before they break, so that they can be taken out of service before they do some damage.

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 in the free span (away from tube sheets), 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 F. 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 tube sheet. 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 tube sheet. 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 tube sheet, the end of the broken tube would be restrained in the hole, and a controlled leak would result. Based on this argument, 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.

F. Steam Generator Leak Tightness After Repair

Finding 1 - The Review Group expects that operational leak rates, principally from leakage through the explosively expanded tube joints, may exceed the process qualification requirements.

The steam generator repairs have been qualified to requirements for a maximum total leak rate of 1 lb/hr. The qualification program has demonstrated the capability of the process to meet this requirement. However, industry experience with high quality, expanded, unwelded tube-to-tubesheet joints would indicate a low probability that this leak tightness will be obtained in these steam generators. Each repaired steam generator will depend upon several thousand expanded tube joints. Even if this leak tightness is initially attained, leakage will tend to increase with future operation.

GPU Nuclear has further evaluated the effects of operation with a leak rate through the steam generators of 6 gal/hr. (about 50 times the

qualification maximum rate). Operation was found acceptable by GPU Nuclear. Plant modifications are in progress to facilitate operation with leak rates in this range and administrative controls will be imposed directly on leak rates or on related radiological conditions. We recognized that GPU Nuclear has evaluated and prepared for this eventuality.

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.

G. Plant Operations

Finding 1 - GPU Nuclear has satisfactorily addressed most of the operational considerations of concern to the Review Group. Others, particularly those relating to Leak-before-break and multiple tube failures are still in development and their adequacy cannot be adjudged.

Specific concerns that have been addressed include:

a. That normal operating parameters will need to be changed as a result of the OTSG repair program. GPU Nuclear has analyzed the plugging effects and has concluded that no significant operating parameter changes will need to be effected for normal operation.

b. That adequate procedures for abnormal operation are developed and operators are trained with them. GPU Nuclear is in the process of upgrading procedures. Operating training is planned following this upgrade. Multiple tube leak procedures are included in this upgrade. Operator training under steam generator leak conditions was completed following the GINNA tube rupture incident. Annual requalification (both class room and simulator)

training either has been completed or is presently scheduled.

c. That the plant will have capability for handling leakage into secondary systems in event of a steam generator tube leak. Various station modifications are presently being installed to address this concern.

   1. Turbine building sump will be fiberglass lined to facilitate decontamination.
   2. Additional processing equipment and a 250,000 gal. storage tank will be installed to provide reaction time and cleanup capability.
   3. Various drains will be re-routed to reduce water volume collected in the turbine building sump.
   4. Various drains that could become contaminated will be re-routed to the turbine building sump.
   5. A system will be installed to collect and dispose of condensate polisher resin as radwaste.

d. That adequate steam generator leak detection capability will be provided. Systems with the capability to identify and quantify steam generator leakage will be in place along with appropriate guidance to the operator.

The following operational considerations could be improved and are the subjects of recommendations:

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.

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.

We feel this will allow the operator time to feel comfortable and "get his arms around" the job responsibility. TMI-1 operating personnel have not been exposed to actual operating plant experience in over three years. During this period people have been promoted and transferred and now hold positions in which they have no real operating plant experience. Plant modifications have been installed during this period with no opportunity for the operators to gain "hands-on" experience in the use of this equipment.

APPENDIX A

INTERIM REPORT OF THIRD PARTY REVIEW OF THREE MILE ISLAND, UNIT 1, STEAM GENERATOR REPAIR

To

R. F. Wilson - Vice President, Technical Functions

GPU Nuclear

Prepared by

THIRD PARTY REVIEW GROUP:

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Richard W. Weeks
Edwin J. Wagner - Chairman

Submitted for the Review Group

by: E. J. Wagner

E. J. Wagner

date: 9/27/82

PURPOSE:

This is an interim report of the Third Party Review Group to evaluate a part of the TMI-1 Steam Generator Repair Program - that relating to the safety of conducting the proposed repair of the steam generators while the plant is in a cold shutdown condition, including the effects of the repair on the steam generators and on the remainder of the TMI-1 plant. This interim report was requested by R. F. Wilson, GPU Nuclear, on August 10, 1982 to obtain the Review Group's evaluation of this part of the steam generator program concurrent with decision making on conducting the repair in TMI-1. As GPU Nuclear completes the remainder of the overall repair program, the Review Group will report its evaluation of the remainder of Scope of Review defined in the Charter for this Third Party Review.

CONCLUSION:

Based upon the information developed by the repair program and summarized in the Safety Evaluation, the Review Group concludes that the proposed repair conducted on the TMI-1 steam generators in conformance with the control systems described will not have adverse effects on the nuclear safety related items of the plant (including the steam generators) in the cold shutdown condition. This includes consideration of potential hazards from:

1. Missiles generated by the explosive process.

2. Introduction of chemical residues in the steam generators, reactor coolant system or the reactor plant ambient.

3. Transmission of pressure pulses through air or structures to sensitive items such as previously expanded tubes, previously plugged tubes, other steam generator structures or safety related instruments and controls.

4. Handling of explosives in nuclear safety related equipment or structures (We note that no Review Group member is expert in handling explosives. Acceptance of the response to this potential hazard is based on the procedure controls described in the Safety Evaluation, their compliance with the Laws of the State of Pennsylvania and the exclusive use of blasters licensed in accordance with that Law).

5. Occupational radiation exposures.

This conclusion applies only to the safety of the plant, including the steam generators, of conducting the proposed repair while the plant is in the cold shutdown condition. This conclusion does not apply to the safety of returning the plant, with steam generators repaired by the proposed process, to service.

Although efficacy of the repair is not a consideration in the safety of conducting the repair, it will be important to the safety of returning the plant service. GPU Nuclear may elect to proceed with the proposed repair at substantial economic cost. The Review Group therefore considers it appropriate to render an opinion now about the efficacy of the repair as it may affect future safety considerations.

The Review Group believes that the proposed repair, after completion of the on-going qualification and when conducted in accordance with the control procedures, has a high probability of producing tube-to-tubesheet joints adequate for a safe operation of the plant. However, based upon industrial experience with expanded, unwelded tube-to-tubesheet joints in high pressure heat exchangers, it is expected that greater leak rates will occur during normal operation than typically experienced on new nuclear plant steam generators. The Safety Evaluation covering return of the plant to service should consider this possibility

APPROACH:

On April 12, 1982, R. F. Wilson of GPU Nuclear established a Third Party Review of the TMI-1 steam generator repair program. A Charter was supplied which defined the purpose, scope, membership and operations of the Review Group. The evaluations of the Review Group have been conducted in conformance with the Charter.

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E. G. Wallace of GPU Nuclear is a non-voting member who serves as liaison with GPU Nuclear and was assigned as Secretary. All members have been present and participated in all meetings of the Review Group.

It should be noted that the members act as independent individuals on this Review Group. Neither their individual statements nor their contributions to any reports of this Review Group are intended to represent the opinions or conclusions of the organizations with which they are affiliated.

The evaluation reported in this interim report was conducted concurrently with evaluation of the full Scope of Review defined in the Charter. The evaluation was conducted using reviews of pertinent documents, submittal of written questions to GPU Nuclear, written responses by GPU Nuclear, review of specialty topics by individual members, presentation by cognizant GPU Nuclear or contractor personnel, Review Group meetings and Executive Sessions of the Review Group members only. Full day meetings of the entire Review Group were held on April 23, May 20 and 21, and August 24 and 25, 1982.

The focus of the evaluation of this interim report was the Safety Evaluation of the TMI-1 steam generator repair distributed to the Review Group (E. G. Wallace letter of August 20, 1982 and of its reference documents 1, 2 and 4). The proposed repair described in this Safety Evaluation is the explosive expansion of the top 17 or 22 inches of the tubes within the top tubesheets of both steam generators. The repair will be made on all tubes which will be returned to service. The explosive expansion creates new pressure boundary joints between the reactor coolant and steam-side of the steam generators.

COMMENTS:

During the review of the Safety Evaluation and supporting documents, certain observations were made by the Review Group. These comments and the GPU Nuclear responses could be pertinent to the conclusions of the Review Group in assessing the return of the plant to service. They are therefore documented as follows:

1. The Safety Evaluation states that the repair joints will be leak tight and meet the design bases of the original joints. As the Review Group stated under Conclusion, the repaired joints will probably be adequately leak tight for safe operation. However, the joints should not be expected to be as leak tight in normal operation as those of typical new steam generators.

The Safety Evaluation for the return of the plant to service should consider the potential for higher leak rates from reactor coolant to steam systems and the handling of resultant radioactivity discharges such as from the condenser air ejectors. GPU Nuclear agreed.

2. Although indirect measurements provided some indication, the repair process qualification does not contain a direct metallographic examination to verify that the metallurgical structure of the tube material in the expanded region is not degraded by the expansion.

   GPU Nuclear said that such a metallographic examination would be included in the qualification.

3. Paragraph 6.1 of the Safety Evaluation discusses residues left on the steam generator surfaces by explosive expansion. The Review Group understands that the testing at Mt. Vernon showed greater amounts of residue than expected based upon mock-up tests. Some cleaning is now expected to be necessary. GPU Nuclear further advised that a material called Immunol is under consideration as a coating to facilitate removal of residues. It would be applied to the tube surfaces before the explosive expanding.

   The Review Group suggested that specific limits and appropriate check methods be included in procedures to preclude existence of detrimental contamination from either the explosive residues or Immunol after completion of the repair. GPU Nuclear agreed.

4. Paragraph 6.5 of the Safety Evaluation indicates that the steam generators will be isolated by temporary plugs from the reactor coolant system during repair. However, a cognizant contractor person stated that, based upon the Mt. Vernon testing, temporary plugs might not be used.

   The Review Group considered it important to assure that contaminants from explosives not be allowed to travel into the reactor coolant system. GPU Nuclear agreed and reiterated that the temporary plugs will be used.

5. Paragraph 8.0 of the Safety Evaluation discusses the quality assurance and quality control for the repair.

   The Review Group asked whether a quality plan existed specifically for the repair. GPU Nuclear advised that the quality actions are an integral part of each procedure

and that all the repair activities would be conducted in accordance with written procedures and the TMI-1 Quality Assurance Plan.

A specific quality check of the adequacy of the overall explosive process (type of explosives, amount of charge, charge condition, correctness of assembly, etc.) was suggested by the Review Group as the type of check that might be identified by a specific quality plan for the repair. The intent of such a check would be to detect any repair process failure early rather than during testing at the completion of repair. This specific check might be conducted by actually explosively expanding a test joint periodically with production explosives and equipment. GPU Nuclear advised they would assure that the integral quality provisions of their procedures constitute an adequate quality plan for the repair and that an overall quality control check such as suggested would be included.

GPU Nuclear
100 Interpace Parkway
Parsippany, New Jersey 07054
201 263-6500
TELEX 136-482
Writer's Direct Dial Number:

April 12, 1982

Members, TMI-1 OTSG Repair Program
Review Group

SUBJECT: TMI-1 Steam Generator Repair Third Party Review

The corrosion problem that has developed in the TMI-1 steam generators represents a first-of-a-kind condition for commercial reactors. As in any new problem, the opportunity for overlooking important relationships between elements of the problem is greater, therefore, additional care is required to assure that all important links are identified and dealt with. It is for that reason GPU Nuclear felt that an independent third party review effort would be warranted to provide additional assurance that the failure mechanisms and repair proposal are compatible and appropriate, and do not represent a safety concern when the plant returns to power.

To achieve the desired result, GPUN Management devel^ped a charter for the review which I have attached for your information and guidance. The membership of the group was developed to bring a broad range of individuals with the necessary expertise related to the steam generator failure together for the review function. Where available I have provided copies of resumes of the members so that you may become familiar with your fellow members' capabilities. I think that you will agree that we have been fortunate in bringing together the quality people needed to review this unique problem.

It is envisioned that a minimum of two meetings will be necessary to complete the review. The first meeting has been scheduled for April 23 at the GPU Headquarters, Parsippany, New Jersey. For your assistance, I have included a simplified map of the routes from major highways and airports. If you would like some assistance with local accommodations, please let me know. The first meeting will begin at 10:00 a.m. and will cover a number of organizational details first. The technical presentation will start about 11:00 a.m. and be scheduled to last until 4:00 p.m. so that people may make evening travel connections. Subsequent meetings will be scheduled on the 23rd with the final meeting being near the end of May.

In preparation for the first meeting, I have enclosed a number of documents which will provide you with a general background of the program and progress as things now stand. Your review of these documents will undoubtedly raise questions that you would like addressed at some point. So to try and make the first meeting productive as possible, I would request that you send or call in questions in advance of the meeting and they will be factored into the agenda. The base agenda that is envisioned will be the same as the one used in the April 7th NRC status meeting. The material from that meeting is in one of the attachments, and will be updated for the 23rd.

I hope that this package will answer a number of your questions and get the review off to a good start. If you have any further questions, please call Ed Wallace at (201) 299-2191, who has been assigned as Secretary of the group.

Very truly yours,

E. G. Wallace /mt
for
R. F. Wilson
Vice President - Technical Functions

/mt
Attachments

LIST OF ATTACHMENTS

1. Charter, Third Party Review of TMI-1 OTSG Repair Program
2. Membership List and Resumes
3. Map of Routes to GPU Headquarters
4. OTSG Repair Task Force Organization
5. Preliminary Report - Failure Analysis
6. Preliminary Report - Eddy Current Examination Program
7. Agenda (Proposed) for April 23rd Meeting
8. Handouts from April 7 and January 25 Status Meetings with NRC

ATTACHMENT 1

THIRD PARTY REVIEW OF TMI-1 OTSG REPAIR PROGRAM

CHARTER

I. PURPOSE

It is the intended purpose of this "Third Party Review" (TPR) to provide a timely, independent, objective, safety evaluation of all activities defined in this charter for conformance to: 1) the NRC rules & regulations governing the operation of TMI-1; and 2) the adequacy of the steam generator repair program that will allow safe operation of the nuclear unit.

II. SCOPE

The scope of this review is generally limited to activities associated with the identification of failure mechanisms and repairs of the TMI-1 Once Through Steam Generators (OTSG's). The specific task areas to be reviewed are described in more detail in Section IV of this charter. It is the intent of GPUN Managemnt to fully develop and implement repairs to the TMI-1 OTSG's within the provisions of 10CFR50.59. It is expected that the TPR will promptly notify GPUN Management of any circumstance not already identified by GPUN, that fails to meet these standards.

III. MEMBERSHIP

The membership of the TPR body shall include individuals with expertise in the following specialty areas:

A. Steam Generator Design and Performance	- 1 member
B. Chemistry	- 1 member
C. Materials	- 1 member
D. Stress Analysis	- 1 member
E. Safety Analysis	- 1 member
F. Plant Operations	- 1 member
G. Non-Destructive Examination	- 1 member

The TPR shall have a GPUN individual assigned as Secretary for the review. He will be the general interface for the TPR membership and GPUN. A Chairman will be elected and report the results of the review directly to the Vice President - Technical Functions in this assignment. The Secretary will arrange for all review material, meetings and recordkeeping for the team. Any specific information requests within the scope of the TPR should be to the Secretary. The Secretary will be a non-voting member in any matters of the TPR seeking concensus opinions.

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IV. SPECIFIC REVIEW AREAS

A. Failure Analysis Program

This program is intended to identify the cause of tube cracking and means to arrest it.

B. Eddy Current Examination Program

This program is to develop and implement an eddy current examination method to identify the extent of the tube cracking problem.

C. OTSG Performance Evaluation

It is the object of this effort to evaluate the impact of the repair procedures on the performance of the steam generators, especially in the area of safety analysis.

D. Repair Criteria

This program is intended to provide guidance concerning the type of repair to be done on the damaged tubes.

E. OTSG Repair Program

This program covers the actual repair of the steam generators.

V. MEETING FREQUENCY

The TPR shall initially meet to receive a presentation by GPUN and its consultants on the current status and direction of the OTSG Repair Program. At that meeting, they will be presented with initial reference material that will enable them to assess products then available.

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Other meetings may be held at the discretion of the TPR.

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The Secretary shall maintain records of all meetings of the TPR. The Secretary shall also maintain a record of all documents reviewed by membership. Portions of the material may be proprietary in nature. Appropriate arrangements shall be made to protect proprietary information when it is used.

Transcripts may be taken of the final review meeting and used as part of the documentation package available to the NRC in support of the OTSG Repair Program.

A final report will be prepared by the review team which summarizes its findings and conclusions regarding the safety adequacy of the repair program. The report should make an explicit finding that the approach proposed by GPUN is adequate if that is the conclusion of the review.

MEMBERS, TMI-1 Third Party Review

- STEAM GENERATOR DESIGN AND PERFORMANCE

  ED. J. WAGNER - Director, Engineering and Design, Breeder Reactor Division, Burns and Roe, Inc.

- CHEMISTRY

  DAVID J. MORGAN - Plant Systems Analysis Section, P.P.& L.

- MATERIALS

  DR. R. W. WEEKS - Associate Director Materials Science Division, Argonne National Laboratory

- STRESS ANALYSIS

  DR. ARTURS KALNINS - Professor of Mechanics, Lehigh University

- SAFETY ANALYSIS

  WILLIAM LAYMAN - Electric Power Research Institute

- PLANT OPERATIONS

  STAN HOLLAND - System Production Engineer - Duke Power Co.

- NON-DESTRUCTIVE EXAMINATION

  STEVE BROWN - Electric Power Research Institute

- SECRETARY

  ED WALLACE - Manager, PWR Licensing, GPU Nuclear

E. J. WAGNER

Education: BSME Carnegie-Mellon University
Post Graduate Education at Case Western Reserve, George Washington University and Ohio State University.

Current Position:

Director Engineering and Design
Breeder Reactor Division
Burns and Roe, Inc.

Current Additional Assignments:

(1) Member, EPRI Steam Generator Owners Group Architect/Engineer Advisory Committee

(2) Member, Three Mile Island Unit #2 Technical Assistance and Advisory Group

Historical Technical Background:

- Deputy Director for Engineering, Burns and Roe, Inc.; Deputy Director Technical Evaluation - 1975-1981

- Division Engineering Manager, Westinghouse Electric Corporation - 1970-1975

- Chief, Nuclear Components Branch, Naval Nuclear Propulsion Program (USAEC [USDOE] and US Navy); Senior Naval and AEC Manager of Numerous Steam Generator Development Programs - 1955-1970

- Test Engineer, Babcock and Wilcox Company - Development of Steam-Water Separators

EXPERTISE: Coolant technology of pressurized water power reactors; water chemistry theory, radio-chemistry, corrosion control, and chemical engineering analysis of reactor coolant and steam generator chemistry control programs, systems, and equipment.

PENNSYLVANIA POWER & LIGHT, NUCLEAR PLANT ENGINEERING

Nuclear Analyst, Plant Systems Analysis

Provide technical direction to Systems Engineering Group, which is responsible for evaluations of the adequacy of Susquehanna Steam Electric Station (SSES) for scenarios involving failures of equipment or operations, for changes to the plant or for questions relating to the adequacy of the plant as designed.

COMBUSTION ENGINEERING, NUCLEAR POWER SYSTEMS

4/81-7/81: Senior Consulting Engineer, Systems Chemistry

Responsible for incorporating plant performance observations, research and development test results into water chemistry specifications and control programs. Provided consulting support to operating plants.

6/77-4/81: Supervisor, Coolant Technology Group

Directed up to five professionals in science and engineering. Group conducted in-house and contract research into nuclear steam generator secondary side corrosion problems, evaluated plant chemistry data, maintained CE's Chemistry Manual, and responded to plant problems.

6/73-6/77: Senior Engineer, Chemistry Development

Performed theoretical analyses of chemistry/chemical engineering/ corrosion problems. Planned and conducted special procedures at operating plants to evaluate system performance and to resolve problems.

U.S. NAVY, NUCLEAR POWER SCHOOL, BAINBRIDGE, MD.

4/69-4/73 Naval Officer

Instructor of Chemistry, Metallurgy, and Radiation Fundamentals

EDUCATION: CLEVELAND STATE UNIVERSITY, 9/62 - 6/67
BS Chemical Engineering (Magna cum Laude)
Co-op Experience: B. F. Goodrich Chemical Co.

UNIVERSITY OF MENNESOTA Fall 1967
UNIVERSITY OF DELAWARE Evening), 9/71 - 6/73
Graduate studies in Chemical Engineering

AFFILIATIONS: Tau Beta Pi-Honorary Engineering Society
American Institute of Chemical Engineers
American Nuclear Society

REPORTS

- Member, Steam Generator Owners Group Water Chemistry Guidelines Committee.
  Contributor, "PWR Secondary Water Chemistry Guidelines," September 1981.

- Beineke, Morgan, Hall, Marugg, Wiatrowski, "Test of Isothermal Soaking Procedures for Limiting Tube Denting in Nuclear Steam Generators, "EPRI NP-1761, April 1981.

- EPRI Quarterly Reports, RP623-2 Neutralization, RP623-3 Condensate Polishing, 1979-1981.

- "Evaluation of Reactor Coolant Crud Samples," CE Internal Report, 3/80.

- "Test Report, Hydrogen Absorption (Efficiency in the VCT)," CE Internal Report, 11/76.

- "An Evaluation of the Influence of Letdown Degassification on the Control of RCS Hydrogen Inventory with Hydrogen Overpressure on the Volume Control Tank," CE Internal Report, April 1976.

- "An Evaluation of Sodium Pass through in Nuclear Once-Through Superheating Steam Generators Relative to Solids Deposition in Turbine Cycles," CE Internal Report, 12/75.

- "An Evaluation of Core Crud Deposition and its Effect on Core Pressure Drop," CE Internal Report, 4/75.

- "Fort Calhoun Reactor Coolant System Peroxide Treatment, November 1974, "CE Internal Report, 2/75.

- "Analysis of Core Crud Samples," Various CE Internal Reports, 1973-1976.

Richard W. Weeks
Associate Director
Materials Science Division
Argonne National Laboratory
312-972-4931

Responsible for the materials technology programs at Argonne currently involving the efforts of about 120 people. Materials reliability problems are addressed in a wide variety of energy systems including conventional and advanced nuclear reactors, and coal conversion and combustion plants. Work on stress corrosion cracking in LWR systems has been conducted for more than six years under EPRI, NRC, and direct utility sponsorship. Member of EPRI Corrosion Advisory Committee from 1975 to 1980 and currently a member of the EPRI Materials and Corrosion Committee. Member of the DOE Nuclear Systems Handbook Advisory Committee. Author or co-author of over twenty technical publications and member of the Editorial Board of the Journal of Nuclear Engineering and Design. Registered Professional Engineer in Illinois. B.S.M.E. Swarthmore College, 1964; M.S.M.E. Caltech, 1965; Ph.D. Theoretical and Applied Mechanics U of Illinois, 1968.

RESUME OF ARTURS KALNINS

PERSONAL DATA

Date of Birth: February 13, 1931

EDUCATION

B.S.	Eng. Mech.	The University of Michigan	1955
M.S.	Eng. Mech.	The University of Michigan	1956
Ph.D.	Eng. Mech.	The University of Michigan	1960

EMPLOYMENT

1958 - 1960	University of California, Berkeley; Research Engineer
1960 - 1965	Yale University; Assistant Professor
1965 - Present	Lehigh University; 1965 - 1967, Associate Professor; since 1967 Professor of Mechanics

PROFESSIONAL SOCIETIES

Member ASME
Fellow Acoustical Society of America
Founding Member Academy of Mechanics

EDITORIAL BOARD

Associate Editor, Journal of the Acoustical Society of America, since 1970

CONSULTANTSHIPS

Has served as a consultant on shell analysis to more than 50 organizations.

MAJOR SCIENTIFIC ACCOMPLISHMENT

Has invented (in 1964) a numerical method for solving boundary value problems, governed by N first-order, ordinary differential equations, and applied it to the stress analysis problem of axi-symmetric shells.

PUBLICATIONS

One book and 46 articles in recognized journals.

PERSONAL DATA SHEET

Name: Stanley Austin Holland   Social Security Number: 237-40-7221
Title: System Production Engineer   Date of Employment: 10/16/48

Education:

High School - Cool Springs High School
Forest City, North Carolina

ICS Correspondence Schools - Power Plant Engineering

Full time curriculum in physics, nuclear physics, trigonometry, and chemistry prior to entering reactor operator training - 7 months.

Full time participant in the Westinghouse reactor operators training program covering pressurized water reactor technology, nuclear physics radiation control, calculus, chemistry, and PWR design - received operator license certificate - 9 months.

B&W pressurized water reactor design, strength of material, basis for design, operating characteristics - 4 weeks.

General Electric Company - steam turbine and generator operation and design criteria.

Full time training associated with receiving NRC reactor operator and senior reactor operators license. Various engineering related studies associated with maintaining license.

Extensive training in health physics, system operation, rad waste operation, simulator and control room operation.

Numerous supervisory and management development programs sponsored by Duke Power Company and International Management Council of Charlotte, N.C. Participating in EIT review course sponsored by UNCC.

Experience:

1976 to Present - Approximately 6 years experience working with the General Office nuclear station operation support group. Provides day-to-day operation and maintenance support, active involvement in outage planning and execution, vendor contract negotiation and support coordination, station interface with Design Engineering and other groups within Duke Power Company, interface with other utilities,

implementation of a computerized outage management program for Duke Power Company nuclear stations, implementation of the B&W ATOG program, working with IMPO to develop an Emergency Operating Procedure Writers Guide - holds Crisis Management Team position. Responsible for review and implementation of vendor recommendations concerned with station operation.

1968-1976 - Various start up responsibilities for all three Oconee units - assigned full responsibility for completing construction, hot functional testing, and start up of Oconee Unit 2. Held reactor operator and senior reactor operator license. Held positions of shift supervisor, assistant operating engineer, and operating engineer. Assisted in the formation and development of the Oconee operation organization and operating procedures.

Prior to 1968 - Approximately 20 years fossil station experience in boiler, turbine, and control room operation.

SAH/kcc
3/26/82

STEPHEN D. BROWN

CURRENT POSITION

1980 - Present PRINCIPAL ENGINEER - INSPECTION APPLICATIONS DIVISION - J. A. Jones Applied Research Company, EPRI NDE Center, Charlotte, North Carolina. Responsible for Eddy Current and Acoustical Holography technology transfer.

CAREER EXPERIENCE

1974 - 1980 GROUP LEADER, FABRICATION & QUALITY ASSURANCE - Battelle-Columbus Laboratories, Columbus, Ohio

1967 - 1974 RESEARCH ENGINEER, ELECTRONIC COUNTER MEASURES GROUP - Rockwell International, Columbus, Ohio

EDUCATION

The Ohio State University, 1974, Degree Electrical Engineering

The Ohio State University, 1967, B.S. Physics

PROFESSIONAL ASSOCIATIONS

Chairman of ASME Multifrequency Eddy Current Task Group

Alternate on ASME Working Group on NDE

Stephen D. Brown

Mr. Brown has 14 years experience specifically related to NDE technology and closely related disciplines. As a Group Leader at the Battelle-Columbus Laboratories he was responsible for the development and implementation of eddy current research programs which included the technology transfer of multifrequency/multiparameter eddy current techniques for the inservice inspection of PWR steam generator tubing; the evaluation and quantification of existing single-frequency eddy current technology for PWR steam generator tubing inspection; design and construction of fixed-site and air-transportable PWR steam generator mockups for the training of inservice inspection personnel and evaluation of existing and potential inspection techniques. Mr. Brown has also been instrumental in the application of computer modeling techniques for the optimization of planar multilayered eddy current problems and the selection of optimal mixing frequencies for the multiparamter eddy current inspection of tubing.

Earlier at Battelle, Mr. Brown accrued extensive experience in the boresonic inspection of steam turbine rotors and is co-author of a report documenting boresonic state-of-the-art capability. He was also a member of the Battelle rotor analysis group responsible for the third-party assessment of rotor remaining lifetime and has analyzed approximately twenty-five rotors to-date. Mr. Brown was also involved in the early assessment of BWR piping ultrasonics inspection methods and has extensive experience in the development of ultrasonic inspection techniques and systems for the inspection of depleted uranium penetrators. Mr. Brown has also conducted independent research activities in the area of coherent optics and has implemented speckle diffraction interferometry techniques for the measurement of in-plane displacements and strain and time-average holography for the measurement of resonant modes of vibrating structures.

At Rockwell International, Mr. Brown was a microwave systems engineer responsible for the analysis of foreign radar-associated weapon systems, the assessment of U.S. derived electronic countermeasures, and the development of advanced countermeasure systems.

W. H. Layman is at present the Department Manager for Generic Safety Analysis in the Nuclear Safety Analysis Center (NSAC) operated by the Electric Power Research Institute (EPRI). He has been in the nuclear power industry since 1952. During his nine years in the nuclear submarine program Bill served as Chief Operator in starting up the first submarine prototype nuclear power plant in Idaho, then served as Assistant Engineer Officer in the initial crew of the first nuclear submarine, Engineer Officer and Executive Officer of later nuclear submarines and Squadron Engineering and Material Officer for the first squadron of missile firing submarines.

In 1961 Bill joined the Pennsylvania Electric Company where he served as General Manager of the Saxton Nuclear Experimental Corporation and then as Generation Division Manager of Pennsylvania Electric Co.

In 1968 Bill joined the Atomic Energy Commission as Chief, Water Reactor Branch, Division of Reactor Development Technology. In 1973 he was promoted to the position of Assistant Director for Operations, Division of Reactor Safety Research.

From 1975 to the present time Bill has been a member of the EPRI organization where he has served in a number of capacities including Director, Steam Generator Project Office; Associate Director of NSAC and Department Manager of the Plant Engineering Department.

THIRD PARTY REVIEW OF
TMI-1 OTSG REPAIR PROGRAM

CHARTER

I. PURPOSE

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E.	Safety Analysis	- 1 member
F.	Plant Operations	- 1 member
G.	Non-Destructive Examination	- 1 member

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IV. SPECIFIC REVIEW AREAS

A. Failure Analysis Program

This program is intended to identify the cause of tube cracking and means to arrest it. The program will also include an evaluation of other portions of the reactor coolant system to determine if corrosion mechanism extended out of the steam generator boundaries.

B. Eddy Current Examination Program

This program is to develop and implement an eddy current examination method to identify the extent of the tube cracking problem.

C. OTSG Performance Evaluation

It is the object of this effort to evaluate the impact of the repair procedures on the performance of the steam generators, especially in the area of safety analysis.

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Rev. 3,4/27/82

TPR Bibilography

- 1. Date: April 12, 1982
  Subject: Third Party Review
  To: TPR Members

  Material Transmitted:

  .1 OTSG CHARTER
  .2 Membership List and Resumes of TPR
  .3 Map to GPUN
  .4 OTSG Repair Task Organization
  .5 Preliminary Report Failure Analysis
  .6 Preliminary Report - Eddy Current Exam Program
  .7 Agenda - April 23 - TMI-1 OTSG Review Outline
  .8 Handout from April 7 Status meeting with NRC
  .9 Handout from January 25, TMI-1 OTSG Status Review (NRC)

- 2. Date: April 16, 1982
  Subject: TPR
  To: TPR Members

  Material Transmitted:

  .1 Attachment 8, Part I, Handouts from April 7th Status Meeting with NRC
  .2 Two Additional Resumes of Members of TPR Team

- 3. Date: April 23, 1982 (No Cover Letter)
  Subject: TPR
  To: TPR Members

  Material Transmitted:

  .1 TMI-1 Steam Generator Recovery Program Task 7
  .2 Reactor Coolant System Inspection and Requalification (April 16, 1982) B&W
  .3 Handout from NRC April 23, 1982 from N. Kazanas
  .4 Reactor Coolant System Review Task 7 4/23/82
  .5 TMI-1 OTSG Tube Failure Probability from D. Slear 4/23
  .6 OTSG-B E/C Absolute (4 x 1) Results Summary from D. Slear 4/23
  .7 TMI-1 OTSG Tube Making Process 4/23

• 4. Date: May 13, 1982
Subject: Draft Minutes of First Meeting
To: TPR Members

Material Transmitted:

.1 OTSG Charter (Revised)
.2 Preliminary Specification for the Repair
.3 TMI-1 Tube Preliminary Stress Report (4/21/82)
.4 MPR Associates - draft Residual Stress Report (April 27, 1982)
.5 EPRI Reference Material

a. EPRI Activities in Support of TMI-1 OTSG Recovery Memorandum Report April 1982
b. APPENDIX 1 Status Report 2/24/82
c. APPENDIX VI Laboratory Experience of Cracking of Materials (Other than IN600) in Solutions Containing Sulfur Species

• 5. Date: June 1, 1982
Subject: Draft Minutes of May 20, 21 TPR Meeting
To: TPR Members

Material Transmitted:

.1 Telecon M. Graham w/R. Jacobs (NRC) 5/24 re: NRC Participation
.2 Memo: OTSG Testing Program 4/1/82 TMI-E3914 from D. Slear

• 6. Date: June 21, 1982
Subject: TPR
To: Prof. Arturs Kalnins

Material Transmitted:

.1 EPRI Letter: Status of Tasks Assigned to EPRI by GPUN Failure Analysis Task Group on 2/11/82

• 7. Date: July 2, 1982
Subject: TPR
To: Prof. Arturs Kalnins

Material Transmitted:

.1 Stress report Work by Jim Moore (Updated Copy)

• 8. Date: July 26, 1982
Subject: TPR
To: Prof. Arturs Kalnins

Material Transmitted:

.1 BAW - 10146 (October, 1980), "Determination of Minimum Required Tube Wall Thickness for 177 FA Once Through Steam Generators"

- 9. Date: August 10, 1982
  Subject: TPR
  To: TPR Members

  Material Transmitted:

  .1 TMI-1 OTSG Failure Analysis Report, 7/82
  .2 TMI-1 OTSG As-Built Tube Stress Analysis
  .3 TMI-1 OTSG Recovery
  .4 B&W Evaluation of Tube Samples from TMI-1
  .5 Battelle final report on Failure Analysis of Inconel 600 Tubes from OTSG A and B of TMI-1

- 10. Date: August 20, 1982
  Subject: TMI-1 OTSG Repair Safety Evaluation
  To: TPR Members

  Material Transmitted

  .1 TMI-1 OTSG Repair Safety Evaluation

- 11. Date: September, 1982
  Subject: Minutes of Meeting
  To: TPR Members

  Material Transmitted

  .1 Sulfur Cleanup Handout
  .2 OTSG Logic Diagram
  .3 B&W Memo of July 30, 1982 from J. F. Pearson: Report on Prequalification Charge Sizing for TMI-1 Steam Generator Tube Expansion
  .4 Kazanas Handout: August 9, 1982 TMI-1 OTSG Task 4 Eddy Current Presentation to the NRC.

- 12. Date: September 17, 1982
  Subject: TMI-1 OTSG Repair Process Description and Qualification
  To: TPR/GORB Members

  Material Transmitted:

  .1 Handout from September 15, meeting (in conjunction with Foster Wheeler and Babcock & Wilcox) with NRC.

- 13. Date: September 24, 1982
  Subject: TPR
  To: TPR Members

Material Transmitted:

.1 Revision 4 of OTSG Tube Repair Specification

- 14. Date: October 5, 1982
  Subject: TPR
  To: W. H. Layman (cc: TPR)

Material Transmitted:

.1 Transparancies used during TPR Meeting held on 8/25/82.
.2 Summary text used as mechanical input to SER.
.3 Prof. F. Erdogan's (Lehigh University), Part I of II.

- 15. Date: October 20, 1982
  Subject: TPR
  To: TPR Members

Material Transmitted:

.1 Revision 1 to the Steam Generator Repair SER.
.2 NRC's SER on the Explosive Repair Process.
.3 (Via D. G. Slear) Presentation material given to NRC during 10/18-10/19 meeting.

- 16. Date: November 19, 1982
  Subject: TPR
  To: TPR Members

Material Transmitted:

.1 Draft Safety Evaluation for Return to Service after OTSG Repair
.2 Results of Argonne National Laboratory's analysis of the Sludge in the "C" Reactor Coolant Bleed Tank"
.3 Draft Guidelines for procedures to deal with beyond design basis tube ruptures.

- 17. Date: December 1, 1982
  Subject: TPR
  To: TPR Members

Material Transmitted:

.1 TMI-1 OTSG Tubing Eddy Current Program Draft Qualification Report (October, 1982)
.2 GPU Nuclear Memo: Procedures Plan for Issuance of Guidelines for Single and Multiple Tube Ruptures, Dated: October 13, 1982.
.3 GPU Nuclear Memo: Progress Report on OTSG Tube Rupture Procedure Development, Dated: November 19, 1982.

.4 B&W Final Draft Report; TMI-1 OTSG Repair Kinetic Expansion Technical Report, Rev. 0, Dated: November, 1982.
.5 Response to TPR Questions.
.6 Results of Supplemental ISI.
.7 B&W, Research & Development Div Letter, Report No. 5433-01, July 13, 1982.
.8 TPR Bibliography.
.9 Safety Evaluation for Return to Service After Steam Generator Repair, Rev. 0.
.10 SDD 232-C, Rev.0, TMI-1 Primary to Secondary Leak Temporary Waste Processing System.

COMPILATION OF REVIEW GROUP QUESTIONS & GPU NUCLEAR ANSWERS

I. PLANT OPERATIONS

1. Are there plans to augment operating shifts with additional people specifically for responding to steam generator tube leak incidents?

There is no need to assign additional personnel for responding to steam generator tube leak incidents. This incident can be handled by normal staff on duty. A tube leak incident of greater than 1 gpm but less than 50 gpm would be categorized as an alert by the unit Emergency Plan. A rupture of greater than 50 gpm is a site emergency. Both levels of event require implementation of the emergency plant and organization. The supplemental staff will be available on site ·ithin 60 minutes.

2. Is additional training required for operating shift and support personnel to better prepare them for managing tube leak incidents?

Yes. The following training has been conducted or is planned to improve handling of steam generator tube leaks:

a. Following the Ginna steam generator tube rupture, all operators were trained in accordance with NRC guidelines and using the lesson program developed at Rochester Gas & Electric.

b. The annual simulator requalification training for operators is scheduled for January and February, 1983. As was done last year, both simulator and classroom time have been set aside for steam generator tube leak incidents. This year additional classroom hours are scheduled to discuss the upgraded steam generator tube rupture procedures.

c. Training is planned at TMI on the upgraded procedures as they are finalized.

d. Training will be provided on operation of all new equipment.

3. Are there presently any indications that failed fuel may be a problem? Is there a procedure that will identify RC system activity increases during low power testing?

At the end of the last cycle we had 0.03% failed fuel compared with 1.0% failed fuel used in accident analyses. This was the higest experienced at TMI-1. With a beginning-of-cycle core now in place, the percent failed fuel should be substantially lower. RCS activity is routinely monitored and an increase in activity would be evident during low power testing.

4. Station capability for handling leakage from the secondary system in event of a significant steam generator tube leak experience.

a. Installed and proposed processing equipment to handle contaminated water input to the turbine building sumps (valve or other leakage from the secondary system).

Additional processing equipment and a holdup tank are being installed to provide sufficient capacity and operational flexibility in case of

significant S. G. tube leak. This modification includes installing pipes from the TMI-1 Turbine Building sump to a 250,000 gallon tank.

The system and its intended operation are described in detail in Ref. 17.10.

b. Capability to reduce, eliminate, or re-direct cooling water or other inputs to the turbine building sumps during shutdown and cooldown of the unit to minimize the volume of water to be processed.

The following drains have been or will be rerouted prior to return to service:

1. Industrial Cooler Blowdown Drain.
2. Main Steam Safety Valve Drain.
3. Auxiliary Boiler Blowdown Sump Pump Discharge Drain.
4. Secondary Plant Sample Sink Drain.

These modifications reduce the volume of water collected in the turbine building sump. In the event of a large steam generator tube rupture, the fluid that would normally be processed through the turbine building sump will be pumped to a large hold up tank. The new tankage will hold up to 30 days fluid for processing.

c. Are there installed systems that will minimize contamination of the auxiliary steam system and boiler?

No. Aux boiler uses condensate. Leakage from aux boiler to its sump is to be directed to Turbine Building sump for monitoring prior to discharge and processing, if required.

d. Capability for inservice cleanup of secondary system contaminated water prior to restart of the unit.

Yes. This capability will be provided using Powdex Resin System and disposal of Powdex resin as radwaste. See additional information in answer to I.4.a.

e. Approved procedures and operator training required to minimize off site releases from secondary system contamination.

Procedure guidelines which address normal offsite releases and offsite releases due to tube ruptures are planned for completion prior to 12/31/82. Training for the procedures is planned to begin prior to 1/15/82.

5. Will procedures dealing with multiple tube failures be available?

Yes. See draft procedure guidelines for beyond DBA procedure development plans (Ref. 16.3, 17.2 and 17.3).

6. Do you plan to address simultaneous leaks in both generators?

Yes. See draft procedure guidelines (Ref. 16.3, 17.2 and 17.3).

7. In event of a tube rupture and auto initiation of high pressure injection

normally all RC pumps will be removed from service. Will natural circulation provide sufficient cooling to prevent steam reliefs from opening and what is the impact of this steam release?

Response to this item will be supplied at the December 7th meeting.

8. Is it possible to completely cooldown and depressurize the primary system, without reactor building entry, to minimize primary leakage into the secondary system due to differential pressure?

The equipment used to minimize RCS/OTSG differential pressure is dependent on the RC pump status. With RC pumps available pressurizer spray would be used to maintain the minimum allowable subcooling margin (i.e., minimize OTSG leakage).

With RC pumps unavailable, then RCS pressure would be controlled by either the pressurizer vent line or the PORV. Both of these valves are remotely operated from control room. Therefore, containment entry would not be required to minimize OTSG leakage.

9. The OTSG Task Organization included in Task 8 an evaluation of modifications that may be necessitated by operation with small primary to secondary leaks. However, the agenda for the review appears not to include discussion of the modifications that may be required.

The three categories of planned modifications are as follows:

TMI-1 OTSG Task 8 Modifications

A. Radiation & Contamination Control Modifications

Temporary shielding (as required), lining of sumps, reroute of some drains, airborne & liquid leak detection/monitoring upgrade of industrial waste treatment system and filter system (IWTS/IWFS).

B. POWDEX Resin Processing/Backwash Water Recovery Modifications

Portable skid mounted.

C. Increased Water Storage Modification

Holding tank for abnormal/casualty conditions (250,000 gallons) with connections for portable demineralizers.

These three modifications are described in detail in Ref. 17.10.

II. SAFETY ANALYSIS

1. Reactor Control and Shutdown

a. Has in-core instrumentation been affected by the corrosion process? (in a way which might invalidate safety setpoints, or prevent a safe shutdown).

No. It has not been affected. The Task 7 final report (Ref. 3.1) reports the results. See Table V-1 sections:

VI., i.e., VI.2.b, VI, 2.d, VI.3.e

b. Have control rod drive mechanisms been affected so that they will not perform adequately their control and shutdown functions?

The Task 7 final report (Ref.3.1) reports the results.
See Table V-1 sections: VI.1.b, VI.1.c, VI.2.a.

c. Has corrosion increased the probability of a control rod ejection accident? The control rod drive mechanism is mounted on an inconel nozzle welded to the carbon steel reactor vessel head. Stainless steel clad is buttered over the bi-metallic weld.

We have concluded that this is unlikely based on our inspection program. We have found no damage in that area.

See inspection of CRDM In Task 7 final report (Ref. 3.1 Section V).

d. Could the corrosion process have degraded the ability to shut down the reactor by boron injection?

Charging system (LPI/HPI) nozzles were inspected. No damage was found. These areas represent conditions of highest susceptability.

2. Reactor Coolant System Inventory

a. Has the primary coolant boundary been degraded in areas in addition to the steam generator tubing? Examples of susceptible areas include reactor vessel nozzles, safety valves, PORV, reactor coolant pump seals, and isolation valves.

Nothing has been identified. See Task 7 report (Ref.3.1 Section V) for details.

b. Has the ability to make up fluid inventory been affected? (tanks, piping, valves, pumps)

In addition to the Task 7 inspection program, supplemental ISI has been performed on Spent Fuel, Decay Heat and Building Spray systems. No degradation that would lead to an inability to make up fluid inventory has been found (Ref. 17.6).

3. Beyond Design Basis Tube Rupture

a. Should safety analyses be performed assuming multiple steam generator tube ruptures? The normal design basis analysis considers the guillotine rupture of only a single tube. The basis for performing analyses of only a single tube rupture has been that it would be caused by a random failure and that operation would not be allowed with the general condition of the tube bundle degraded. With slower developing failures it has been demonstrated that inspections and tight leakage limits allowed degraded tubes to be plugged before serious failure and before a large number of tubes could degrade to the point where multiple tube failures would be considered to be credible. With the potential for incipient cracks in a large number of tubes we should evaluate the probability that the plant operators might be faced with multiple tube failure in a single accident. It is probable that simple analyses will show that the consequences of multiple tube failures (dozens or more) within the tubesheet area are bounded by the design basis single tube rupture out of the tubesheet area. This result could give us high confidence that even if the proposed repair were to be faulty, the great majority of cracked tubes which are within the tubesheet area would not pose an unacceptable safety problem. Incipient cracks below the upper tubesheet pose a more difficult problem.

Based on our Safety Evaluation (Ref. 17.9, Section IX), we have concluded that ECT has detected those cracks with the potential for propagation to failure during a transient or due to flow-induced vibration. We conclude that any failure which might occur in the future could be characterized as random. In addition, leakage through cracks of various circumferential extent has been analyzed to determine a threshold of detectability. Analysis shows that a crack wuld be identifiable due to leakage before it propagated to a size that would rupture under accident loading. (See also Ref. 17.9, Chapter IX).

However, we are developing a program for multiple tube ruptures and ruptures in both generators. See draft guidelines (Ref. 16.3, 17.2 and 17.3).

b. Are existing design basis safety analyses affected by this corrosion problem? Examples could be main steam line break analyses and loss of coolant accident analyses. A pertinent question is about the possibility of one of these design basis accidents being complicated by having the force of that accident cause simultaneous tube failures in a degraded steam generator tube bundle.

The design basis analyses have been applied to both the new joint and the tube portion left in service. The new joint has been qualified to hold under maximum accident loading (main steam line break) with margin. Thus multiple tube ruptures in the tubesheet caused by joint loosening under accident condition are considered unlikely (See Ref. 17.9 Chapter V and Ref. 17.4). Analyses have also been done to determine the size of crack which would rupture under accident loading.

Analyses of cracks caused by this corrosion problem indicate that a propagating crack would be identifiable due to leakage before reaching a size that would ductilely fail during a main steam line break. (See Return to Service SER (Ref. 17.9) Chapter IX for further details.)

c. There are a number of hypothetical "what if" questions that need to be

addressed. For example:

How would TMI-1 respond if (a) several tubes burst essentially simultaneously in either of its steam generators, or (b) tube failures occurred in both of its steam generators simultaneously? I realize these questions may be outside the current design basis, but in view of the extensive tube damage in both steam generators and the novel "fix" proposed, these questions should at least be discussed.

A program addressing TMI-1 response to multiple tube ruptures and ruptures in both generators is under development. See draft guidelines (Ref. 16.3, 17.2 and 17.3).

4. Has the corrosion process degraded the ability of the plant to remove heat from the reactor and transport it to heat sinks? This includes the ability to shutdown from operation, cool down, and maintain the system in a cold shutdown condition (heat exchangers, pumps, piping, and valves.)

Task 7 examined a sampling of areas of expected highest susceptability in the primary system. See Ref. 3.1, Chapter III, Table III-1 for a listing. In addition, supplemental ISI was performed on several supporting systems. (Ref. 17.6). No damage was found. Based on these results, it was projected that no damage has occurred to degrade the ability of the plant to remove heat from the reactor.

5. Has the corrosion process degraded the ability to isolate the reactor containment building? Examples might be fluid system containment isolation valves.

Based on Task 7 (Ref. 3.1 Chapter V) results, no damage is projected.

6. Is sodium thiosulfate going to be eliminated from the site? Since the thiosulfate was present for the purpose of post accident radioactive iodine control, have analyses confirmed the adequacy of iodine control without the use of sodium thiosulfate?

Yes, thiosulfate has been eliminated from the unit. The required changes (Tech. Spec. Change Request and associated Safety Analysis) justifying the adequacy of spray iodine retention, have been submitted to NRC. Resolution is required prior to restart.

7. Radioactivity Releases

(a) Has any equipment been affected that is used to mitigate accidents?

In the Task 7 sampling of equipment (Ref. 3.1 Chapter V), no damage was found in areas of the primary system of suspected highest susceptability. Based on these results, no damage to any equipment has been projected.

(b) Has the corrosion process caused any condition which could increase significantly the release of radioactivity?

Based on our safety evaluation (Ref. 17.9), we do not anticipate an increased probability of events leading to significant releases of

radioactivity. The repair process itself will leave the steam generators with mechanical rather than welded joints which can be expected to result in a slight increase in primary to secondary leakage. Thus routine offsite releases may increase slightly, but should remain within Appendix I limits.

(c) Will the plant condition have any effect on emergency guidelines or procedures?

Possible effects on emergency guidelines or procedures are being evaluated as part of the overall program to upgrade tube rupture procedures. See Guidelines (Ref. 16.3, 17.2 and 17.3).

III. NDE

1. GPU believes that if the generator tubing is cracked then the crack is through-wall. Physically, I find this hard to believe. The Battlelle metallography shows intergranular penetrations 10%-50% through wall. This would seem to demonstrate that we in fact can have a continuum of depths.

   Although ECT probes used have been qualified to find much smaller intergranular cracks (Ref. 17.1), the vast majority of cracks identified are through-wall. However, the existence of unidentified cracks below the threshhold ECT detectability has been assumed in the safety analysis. See SER (Ref. 17.9) Chapter IX.

2. What is the largest crack (length-depth) that can be allowed to remain in a tube?

   Curves have been developed for length vs. depth, indicating the largest crack that can remain in service. See SER (Ref. 17.9) Chapter IX.

3. What is the reliability of the existing eddy current NDE methods (differential coil or 4x1) in detecting the crack mentioned in 2)?

   Both the .540 differential core and the 8 x 1 absolute probe have been demonstrated to have adequate reliabiliy for detecting the cracks described above. See SER (Ref. 17.9) Chapter IX.

4. The lane region in OTSG's is subject to cross-flow conditions. Shallow intergranular penetrations may act as seed cracks which may rapidly propagate to failure in a high-cycle fatigue mode. Should special consideration be given with regards to tube stabilization.?

   Tubes to be plugged will also be stabilized if they have cracks in areas of high cross flow. See SER (Ref. 17.9) Chapter IX for a detailed discussion of areas to be stabilized.

5. Provide the basis for the sensitivities used during the in-generator eddy current tests.

   See the draft ECT qualification report (Ref. 17.1).

6. Similar chemical conditions appear to cause both IGA and IGSCC in sensitized I-600, and IGA has been observed in cracked areas. Are techniques available to periodically monitor for increases in the extent of IGA as an indicator of potentially corrosive conditions?

   No, not with nondestructive techniques. Leakage data in operation and ECT will detect major changes in tube integrity.

7. Are techniques available to determine the effect of explosive expansion on existing cracks in the tubing?

   The effects of explosive expansion on existing cracks are being evaluated

outside the OTSGs. Tubes removed from the generators and sulfur induced IGSCC grown in the lab have been expanded and examined ECT dye penetrant and, destructive techniques. No ductile tearing of cracks has been identified. See answer to question V.1.C.

8. Are techniques available to periodically monitor for tubesheet corrosion under cracked areas of tubing repaired by explosive expansion?

No, not with nondestructive techniques. See answer to question V.1.C.

IV. Failure Analysis Program (cause & cure)

1. Update of Failure Scenario

Has the scenario been defined more completely since our last meeting? Have the important parameters been identified? Has the scenario been verified by duplication in laboratory testing?

The final Failure Analysis Report is Ref. 9.1. Cracking of stressed sensitized I-600 in boric acid solutions with sodium thiosulfate concentrations of 1-10 ppm S_2O_3 at 130°F has been duplicated in the laboratory. The actual TMI tubing was found to be exceptionally susceptible to this type of attack when compared to archive samples of the same heat. (See B&W, Research & Development Div. letter Report Number 5433-01 dated July 13, 1982 (Ref. 17.7) and Failure Analysis report (Ref. 9.1) for further details.)

2. Update on Proposed Sulfur Removal Procedures

Peroxide treatment has been proposed as a means of removing residual sulfur from RCS system surfaces. Given what is known about the failure mechanism, I think this concept needs critical review. My questions and their implications are as follows.

a) What is the basis for believing that sulfur removal is necessary?

Sulfur exists on the steam generator tube surfaces in rather large concentrations (~3000 $\mu g/ft^2$). Analysis of the tube surfaces has shown the sulfur in reduced states as well as fully oxidized. Sulfur is also present on other plant surfaces although at lower concentrations. The oxidation state of the other sulfur is unknown. If during plant startup the reduced sulfur is slowly oxidized it may pass through a very corrosive stage where additional corrosion damage may occur. If the sulfur is oxidized, solubilized and removed by the perioxide treatment under protective high pH conditions the potential for corrosion damage is less than if nothing is done.

b) What is the basis for believing that peroxide can oxidize sulfur residue all the way to sulfate without pausing at an intermediate, possible corrosive state?

The sulfur undoubtedly does pass through the corrosive intermediate state during the peroxide treatment. The breaker experiments conducted at Batelle indicated that this transition is rapid with peroxide present so that no new attack is initiated. In addition, the pH will be slightly alkaline (8.0-8.5) in an effort to further reduce corrosion potential. Corrosion tests are being run to demonstrate the capability of the process to safely remove the sulfur without adverse effects.

c) What are the application conditions (concentration, pH, other chemicals, time, temperature, flow) and what are they based upon? Are they consistent with the oxidation kinetics?

Tests are being carried out to identify the optimum application conditions. Based on tests with actual TMI-1 tubing NiS; the time period appears to be

200-400 hrs; the temperature 130°F; the pH adjusted to 8.0-8.5 with NH ; the peroxide concentration at about 20 ppm. 130°F is the temperature often used in operating plants for peroxide addition. Higher temperatures might cause concern for plant safety while lower temperatures will reduce reaction rates.

d) How is residual peroxide to be removed? If by hydrazine, what is the risk of reducing non-removed sulfur to a possibly corrosive state?

Residual peroxide will decompose to oxygen which will be reduced by degassing in the normal manner prior to startup. Hydrazine additions will be made in small increments to remove residual oxygen to avoid establishing a strongly reducing atmosphere. It may be possible to maintain pH high until deoxygenation has been at least partially completed.

e) How is the end point to be determined?

End point determination will be based on time and growth of SO_4 concentration in the coolant while cleaning. When SO_4 analyses are essentially constant, the cleaning process is considered complete.

f) What steps will be taken to preclude chloride SCC of stainless component while peroxide is present?

Chloride will be kept at $<$0.5 ppm during the cleaning by operation of the normal purification system. It will be reduced to $<$0.1 ppm prior to heatup.

g) What inspections will be performed following this treatment to check that further corrosion has not occurred?

Further corrosion is expected to be identifiable by increases in leakage.

h) My basic problem is that the failure scenario identifies oxidation and/or reduction steps as actors in the corrosion process. Peroxide treatment may replicate these processes. If the residual sulfur is present as sulfide how do we know that perioxide treatment will not convert it to corrosive sulfur species on the way to forming water soluble sulfates?

See previous response to question IV, 2.b and c.

3. It would be of use to hear an update on any new information regarding the organic residue in the bleed tank.

See results of Argonne National Laboratory analysis (Ref. 16.2).

4. How are the concerns regarding sulfur species remaining in the primary circuit (Ref. GPU Appendix XII) to be resolved so that the problem is not re-initiated during service? Oxidation with H_2O_2 was suggested. Has an experimental program been initiated to investigate this approach along with any possible side effects? What are the levels of sulfur species in the primary system (on metal surfaces and in the water) at this time? Would it be possible to monitor the electrochemical potential of inconel and stainless steel in the primary system water as a function of time during requalification testing?

An extensive investigation has been undertaken to investigate the removal of sulfur using an H_2O_2 process. The cleaning procedure has been designed so that the plant coolant conditions will not be greatly different from that of many Westinghouse plants that add perioxide at shutdown. Results indicate that complete sulfur removal can be accomplished in about 200-400 hours. Tests were also conducted with Immunol on the tubes, and after kinetic expansion, to evaluate the impact of these variables on removal.

The Immunol has no appreciable effect on the cleaning process. It does tend to retard the peroxide decomposition, an unexpected benefit. However, the explosive expansion process or uncleaned residue does slow down the sulfur removal process from about 200 hours to 400 hours.

Corrosion specimens expanded to the cleaning process (304 SS-sensitized and Inconel 600 - sensitized - U bends and C-rings made from TMI-1 tubes) have shown no evidence of corrosive attack. Sulfur removal has been ~95% by EDAX measurement.

Samples of water taken from the RCS on 9/25/82 have been analyzed for thiosulfate and sulfate. Thiosulfate was undetectable (<4 ppb) and sulfate was 30 ppb.

Swipes of plant surfaces have shown the following contaminate levels.

Fuel Rod	522 micrograms SO_4/ft^2
Grid	418 micrograms SO_4/ft^2
Regenerative Neutron Source Retainer	530-700 micrograms SO_4/ft^2
RNS Spring	144 micrograms SO_4/ft^2

These levels in some cases are even lower in sulfate contamination than new steam generator tubing but in other cases show some level of contamination, although it is still below that now found on the steam generator tubes.

Very little work has been done using electrochemical potential devices in operating plants at operating conditons. Some work was done in a BWR in investigations of pipe cracking but special sidestream loops must be constructed and specialized probes built. In addition, an extensive research program is needed to determine the meaning of potential variations with various plant conditions changes before the work is done in plant. This was done over the period of one or two years for the BWR work. The extent of this effort makes it impractical for this application.

5. The sodium thiosulfate system was part of the emergency response system. Now that it has been eliminated, what will perform its previous function?

See response to Section II Question II.6.

6. The scenario of the failure mechanism identifies several possible sources of reduced sulfur species. Were considerations given to corrective modifications or procedures to preclude presence of these species in the future?

New administrative controls which are in effect include (1) clearer label

of tanks in the Chemical Addition Room, (2) locking open the breakers to pumps CA-P-2, 3 & 4 and placing them under the administrative control of the Locked Valve and Component List and (3) review of applicable procedures to insure that adequate guidance is provided. Assuming the effectiveness of administrative controls, the only chemical which has a potential for inadvertant induction into the RCS is Sodium Hydroxide and this appears possible only when the RCS is depressurized. Under these conditions, additions potentially would not reach OTSG tubing and even in the event that very dilute caustics did reach the tubes, damages would not be expected since the increase in pH value is toward a more benign condition.

7. The failure scenario concludes that the time of attack was after hot functional testing in September 1981. No direct evidence is cited to establish that corrosion cracking of the tubes occurred then. Were there prior surveillance tests that establish the cracking did not exist at some earlier time?

In 1980 a 3% ECT was performed without evidence of a problem. Indirect evidence that the Unit was able to function during pressure.

8. Did the hot functional tests include any operations that could impose high thermal stresses on the OTSGs, such as boil dry and refill or steam generator relief valve testing?

No.

9. The Failure Analysis Report indicates that OTSG performance history was reviewed in detail. It does not state the worst cases found. Did instances, such as in 6. above occur?

The review indicated no severe transients such as in Question 6 above. On 6/22/78, during a cooldown, the rate briefly exceeded the allowable 100°F/hour reaching 10°F/minutes for about 3 minutes. In no hour period was the temperature reduced by more than 100°F.

V. Repair Criteria

1. Explosive Expansion Process

a. Update of testing and analyses to determine the effect of secondary side tube-to-tubesheet contaminants on the quality of the joint formed and on potential corrosion processes.

Testing to date indicates that tubesheet corrosion serves to enhance the quality of the expanded tube-to-tubesheet joint. The tubesheets used in the qualification program were corroded to conditions simulating actual TMI-1 tubesheet material. In addition, a testing program is in progress using four tubesheet mockups corroded in varying degrees to provide a sensitivity study.

The B&W Kinetic Expansion Report (Ref. 17.4) discusses these programs in detail in secions 2.2.3, 2.5.1.4, 2.8.1, and 2.8.2.

b. Update of activities to determine the residue left on the tube ID by the process, the means of removing it, and/or the implications of not removing it (corrosion or other concerns).

Part of the qualification program involves a detailed evaluation of the materials present in the kinetic expansion process down to a level of one hundred part per million or the minimum detectable concentration. All materials selected for use in the kinetic expansion process have been found chemically acceptable for use in the primary system. As part of our effort to minimize the addition of contaminants, we have selected organic explosives to use during the process, and ordnance cord which contains the debris associated with transmitting the detonation energy to the kinetic expansion insert. Subsequent to kinetic expansion, pieces of polyethelene will be present, and some dark colored deposits on the tubes and tubesheet area. A detailed analysis of the deposits is documented as a part of the B&W Kinetic Expansion Report (Ref. 17.4, Section 2.7.1).

Our approach to evaluating possible effects of this residue is twofold. First, following identification of composition of the debris we consulted both internal and external experts in the field of corrosion of reactor coolant system materials and attempted to identify by literature search and experience what, if any, deleterious effects we might expect from the residue which will remain in steam generator, albeit at very low levels. No adverse effects are anticipated.

Recognizing that this is not absolutely conclusive, we have implemented a long term corrosion test program using actual TMI-1 tube samples subject to the kinetic expansion process and the cleanup process planned for TMI-1. These corrosion test samples have been placed in a chemistry environment comparable to that expected during the first cycle of operation. These samples are essentially lead samples which we anticipate will give us an indication if adverse effects can be expected to occur. See Ref. 17.9, Chapter III.

The OTSG tubes have been coated with Immunol prior to expansion to facilitate cleaning. Testing to determine the effectiveness of cleaning is

discussed in Ref. 17.4, Sections 1.4.3 and 2.7.2. After expansion, felt plugs will be forced through each tube, and a water flush will complete the cleaning. Immunol and felt plugs have been evaluated for possible adverse chemical effects and found acceptable. Any residual Immunol left after cleanup will be removed by the normal purification system.

c. Expansion of cracked tubing to the tubesheet wall may widen or lengthen existing cracks, forming a communication path between the primary coolant and the tubesheet metal. Update testing and analyses to determine the implications of this, or summarize basis for concluding that there is no concern.

Work in this area has included evaluation of both the behavior of cracks undergoing expansion and the tubesheet behind an expanded crack.

1) To date, one defect removed from the OTSGs has been expanded using 25 gr/ft and examined. The defect was not visible to the naked eye on the OD of the tube prior to kinetic expansion whereas subsequent to kinetic expansion it had opened up approximately .03 mils and was visible under 63 power magnification. There was no ductile tearing as shown by metallography. It is our opinion that this defect was already 100% thru wall. (Ref. 17.4, Section 2.6.4). Expansion and examination of several other cracks is planned. In addition, several samples of sulfur-induced IGSCC grown in the laboratory have been expanded using TMI's procedures and examined, included were cracks less than 100% through wall. They also show no ductile tearing, but a slight opening (fishmouthing) of the crack. Since the repair process is qualifying 6" of the 17" or 22" long kinetic expansion which are free of significant defects, the presence of small cracks below ECT detectability which have opened somewhat but not propagated circumferentially or through wall should not affect the leak tightness or load carrying capability of the new joint.

2) There is a concern for boric acid attack of A508 steel if cracks do open as we currently anticipate. In reviewing previous testing we have uncovered sufficient information from B&W tube sheet crevice corrosion tests in the late 1960's and B&W model boiler tests for Florida Power Corporation in 1980 and 1981 to indicate that we should not expect tube sheet degradation due to corrosion at operating temperatures and continuous exposure to borated water. The testing that was done involved varied boric acid concentrations and in the case of Florida Power Corporation testing approximately 6,500 hours of operation. We have, therefore, concluded that the slight opening we may get in through wall cracks that exist in the tube above the bottom 6" of the kinetic expansion will not have any deleterious effects on the A508 steel tube sheet.

2. Tube Plugging

What are the implications of continued cracking of plugged tubes (say due to sulfur on the crack tips or thermal cycling). If the tube completely parts, is the "free end" of concern due to vibration? Should tubes cracked outside of the tube-sheet region be completely removed or cut off below the lowest crack, and the free end "staked" to a support plate?

Tubes with cracks in areas of high cross flow will be stabilized. A summary of the criteria for using various kinds of plugs including those with

stabilizing bars is addressed in the return to service SER Chapter VII (Ref. 17.9).

3. One action of the plan is to expand existing tubes into the tubesheet to provide a leaktight seal. Do the qualifications for this expansion consider that the tubing is not new, e.g., trial expansion of sections of removed tube? It may contain incipient weaknesses not yet identified as defects. Expansion might open potential defects.

Yes. The tube material has been examined metallurgically and found to retain its normal characteristics in areas away from cracks. (Ref. 17.4, sections 2.2.3 and 2.6.4). In addition, evaluation of cracks which have been expanded shows no ductile tearing to enlarge the crack. ECT calibrations indicated that cracks which will propagate under normal operating or accident loads have already been identified. Any incipient defects undergoing expansion are not expected to grow into the range that would propagate. (Ref. 17.9, Chapter IX.)

4. Another corrective action is the plugging of certain tubes with a rolled plug. Has consideration been given to the need to intentionally defect each plugged tube to preclude the existence of a pressure building up with the trapped volume inside plugged tubes?

The activity was considered but is not planned. All plugs, including the Westinghouse rolled plugs, are qualified for use with greater differential pressures than those anticipated due to pressure buildup in a plugged tube.