

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

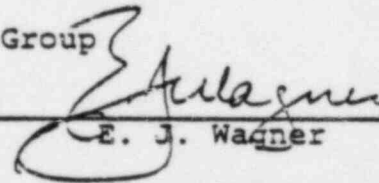
THIRD PARTY REVIEW GROUP:

Stephen D. Brown
Stanley A. Holland
Arturs Kalnins
William H. Layman
David J. Morgan
Richard W. Weeks
Edwin J. Wagner - Chairman

XA Copy Has Been Sent to PDR

Submitted for the Review Group

by:


E. J. Wagner

date:

9/27/82

8211040575-XA

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.

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

<u>Specialty</u>	<u>Name</u>	<u>Affiliation</u>
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.

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 developed 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 Management 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 consensus opinions.

Members of the TPR will either be from outside the GPUN organization or from the portion of GPUN not responsible for the steam generator repair or TMI-1 operations.

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.

A second meeting will be scheduled by GPUN to make a final presentation to the TPR prior to initiation of the final production repairs.

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

VI. RECORDS

The Secretary shall maintain records of all meetings of the TPR. The Secretary shall also maintain a record of all documents reviewed by the 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 MINNESOTA, 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

- o Member, Steam Generator Owners Group Water Chemistry Guidelines Committee.
Contributor, "PWR Secondary Water Chemistry Guidelines," September 1981.
- o Beineke, Morgan, Hall, Marugg, Wiatrowski, "Test of Isothermal Soaking Procedures for Limiting Tube Denting in Nuclear Steam Generators, "EPRI NP-1761, April 1981.
- o EPRI Quarterly Reports, RP623-2 Neutralization, RP623-3 Condensate Polishing, 1979-1981.
- o "Evaluation of Reactor Coolant Crud Samples," CE Internal Report, 3/80.
- o "Test Report, Hydrogen Absorption (Efficiency in the VCT)," CE Internal Report, 11/76.
- o "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.
- o "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.
- o "An Evaluation of Core Crud Deposition and its Effect on Core Pressure Drop," CE Internal Report, 4/75.
- o "Fort Calhoun Reactor Coolant System Peroxide Treatment, November 1974, "CE Internal Report, 2/75.
- o "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 axisymmetric shells.

PUBLICATIONS

One book and 46 articles in recognized journals.

PERSONAL DATA SHEET

Name: Stanley Austin Holland
Title: System Production Engineer

Social Security Number:
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.

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 multiparameter 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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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.

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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 Bibliography

- 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 I 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 Task 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 17, 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/GORE 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 Transparencies 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 within 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 MC 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 highest 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 susceptibility.

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 would 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 susceptibility 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 susceptibility. 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 Battelle 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 threshold 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 4xl) 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 reliability 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 $S_2 O_3$ 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 ($\sim 3000 \mu\text{g}/\text{ft}^2$). 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 peroxide 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 NH₃; 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 SO₄ concentration in the coolant while cleaning. When SO₄ 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 peroxide 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 H₂O₂ 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 $H_2 O_2$ 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 peroxide 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	530-700 micrograms SO_4/ft^2
Source Retainer	144 micrograms SO_4/ft^2
RNS Spring	

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 conditions. 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 labeling

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 inadvertent 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 sections 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 polyethylene 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 boric acid 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.

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