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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289-OLA
	)	ASLBP 83-491-04-OLA
(Three Mile Island Nuclear	)	(Steam Generator Repair)
Station, Unit No. 1)	)	

LICENSEE'S TESTIMONY OF RICHARD F. WILSON,
DAVID G. SLEAR AND DON K. CRONEBERGER ON
ISSUE 1.a (CONTENTION 1.a)

To Mr. Wilson:

Q1. Please state your name and address, and describe your involvement with the TMI-1 steam generator tube repair program.

A1. My name is Richard F. Wilson. I am employed by GPU Nuclear Corporation, 100 Interpace Parkway, Parsippany, New Jersey 07054. As the Vice President of Technical Functions, I was responsible for the overall project and technical management of the TMI-1 steam generator tube repair program.

A statement of my professional qualifications is attached.

To Mr. Slear:

Q2. Please state your name and address ard describe your involvement with the TMI-1 steam generator tube repair program.

A2. My name is David G. Slear. I am employed by GPU Nuclear Corporation, 100 Interpace Parkway, Parsippany, New Jersey 07054. As the Manager of Engineering Projects for

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TMI-1, I was the overall task manager for the TMI-1 steam generator tube repair program, reporting directly to the Vice President of Technical Functions. My responsibilities included all activities associated with the evaluation and repair of the steam generators.

A statement of my professional qualifications is attached.

To Mr. Croneberger:

Q3. Please state your name and address and describe your involvement with the TMI-1 steam generator tube repair program.

A3. My name is Don K. Croneberger. I am employed by GPU Nuclear Corporation, 100 Interpace Parkway, Parsippany, New Jersey 07054. As the Director of Engineering and Design, I provided technical management oversight of the failure analysis and repair activities with special emphasis on evaluation of the steam generator's mechanical design and the impact of the repair on the response of the components. My department also provided engineering support in the areas of Materials Engineering/Failure Analysis, Chemical Engineering and Chemistry, Mechanical Engineering and Engineering Mechanics.

A statement of my professional qualifications is attached.

To all witnesses:

Q4. What is the purpose of your testimony?

A4. The purpose of this testimony is to address Issue 1.a of Contention 1.a as enumerated at page 23 of the Board's Memorandum and Order (Rulings on Motions for Summary Disposition, dated June 1, 1984) in which the Licensing Board stated:

1. The rationale underlying certain proposed license conditions should be addressed, with attention to:

a. Reliability of leak rate measurements.

Q5. Describe the TMI-1 license conditions for leak testing the steam generators.

A5. The existing license conditions related to primary-to-secondary (P-S) leakage through the TMI-1 once-through steam generator (OTSG) tubes are Technical Specifications (T.S.) 3.1.6.3 and 4.1.

Technical Specification 3.1.6.3 reads as follows:

> If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

Technical Specification 4.1. requires that leakage be evaluated daily.

In addition, the following proposed license condition dealing with leakage will be imposed:

> Repaired Steam Generators
>
> In order to confirm the leak-tight integrity of the Reactor Coolant System, including the steam generators, operation of the facility shall be in accordance with the following:
>
> * * * *
>
> 2. GPU Nuclear Corporation shall confirm the baseline primary-to-secondary leakage rate established during the steam generator hot test program. If leakage exceeds the baseline leakage rate by more than 0.1 GPM [6 GPH], the facility shall be shut down and leak tested. If any increased leakage above baseline is due to defects in the tube free span, the leaking tube(s) shall be removed from service. The

> baseline leakage shall be re-established, provided that the leakage limit of Tech. Spec. 3.1.6.3 is not exceeded.

The key points from this proposed condition are that: 1) Licensee was to establish its baseline leakage from the leak rate data obtained during the post repair OTSG hot test program; 2) an increase of more than 0.1 GPM (6 GPH) above this baseline at steady state operating conditions requires facility shutdown and leak testing; 3) if leakage is due to defects in the tube free span, the leaking tubes are to be removed from service; 4) leakage not identified as originating in the tube free span during this testing is deemed acceptable if it does not exceed the 1 GPM (60 GPH) limit of TMI-1 Technical Specification 3.1.6.3; 5) the baseline is re-established following shutdown and leak testing (possibly at a higher leak rate than the initial baseline); and 6) operation can then continue until the increase in leakage exceeds the new baseline by 0.1 GPM (6 GPH).

Licensee determined the baseline primary-to-secondary leakage to be 0.02 GPM (1 GPH) during the steam generator hot test program. This means that the facility is to be shut down if the leak rate reaches 7 GPH total for both steam generators, as compared to the existing limit of 60 GPH in Technical Specification 3.1.6.3.

Q6. How does this compare with the leak rate license conditions for other nuclear plants?

A6. The TMI-1 leakage limitations in Technical Specification 3.1.6.3 are comparable to those at most other pressurized water reactors (PWRs) in the United States. A recent survey by Licensee of approximately 30 PWRs showed that the vast majority of the plants have limits similar to TMI-1's current 1 GPM limit. One plant has a limit three times the current TMI-1 limit. A few of the more recently licensed plants have limits lower than T.S. 3.1.6.3. However, the proposed TMI-1 license condition of 0.1 GPM is more stringent than that for any other operating PWR in the United States.

Q7. What is the purpose of measuring primary to secondary leakage?

A7. Primary-to-secondary leak rate measurements are made periodically for all operating PWRs in the United States in order to confirm that the steam generators are performing as anticipated. TMI-1 is no different than other operating PWRs in this respect. These measurements are one aspect of an overall defense in depth approach to maintaining OTSG integrity. The program includes leak rate monitoring during operation, and periodic eddy current testing, and leak tests while shut down at cold conditions.

The leakage measurements during operation are made both to document the absolute value of leakage and to document any trends which may be cause for concern. The absolute value is

required to both assess the performance of the steam generators and to ensure that technical specification limits are not exceeded. Trends are monitored because increasing leakage may indicate ongoing chemical or mechanical degradation of the tube. Increasing leak rates are investigated further to identify leak locations and take appropriate corrective action.

The intent of the overall defense in depth program is to correct defects in tubes in order to ensure that the steam generator tubes satisfy the licensing basis specified in General Design Criterion 14, 10 C.F.R Part 50, Appendix A, _i.e._, "to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture".

Q8. How were the leakage limits in the proposed license condition for TMI-1 established?

A8. The proposed license condition is based upon stringent administrative limits imposed by Licensee as part of its own program. Licensee included a number of considerations in establishing the absolute value of the leak rate increase during steady state operating conditions which would dictate further action. These considerations are summarized as follows:

1. Establish a leak rate monitoring capability sensitive enough to detect a leak rate as low as 0.5 GPH (about 1% of the Technical Specification 3.1.6.3 limit) during power operations.

2. Establish a baseline leakage rate to take into account the anticipated, low level leakage from the mechanical plugs and the kinetically expanded joint. The current baseline

leak rate of 0.02 GPM (1 GPH) was based on monitored leakage during OTSG hot (pre-critical) testing.

3. Establish a shutdown limit sufficiently above the pre-established baseline so that we can have confidence that the change is significant as compared with the anticipated variation in the nominal monitored leak rate. The OTSG hot testing results indicate that the monitored leak rate statistical variation (twice the standard deviation from the mean value) of approximately $\pm$ 0.01 GPM ($\pm$ 0.5GPH) can be expected during steady state operation.

4. Establish a shutdown limit low enough to ensure conformance with the off-site exposure limits of 10 CFR Part 50, Appendix I. These limits are based on off-site exposure to various body organs over a one year period. Licensee has evaluated off-site releases. These evaluations are based on 0.03% failed fuel. This is the failed fuel percentage prior to the last refueling, so we anticipate the actual failed fuel percentage to be less when we restart. We have determined that the gaseous release mode results in the limiting off-site exposure dose closest to an Appendix I limit. This limit is 15 mr/year exposure to the thyroid due to iodine releases. A continuous 0.1 GPM primary-to-secondary leak rate contributes about 5 mr/year to the off-site thyroid dose rate.

5. Recognize the probability of multiple leakpaths within the OTSC contributing to the aggregate leakage. The baseline leak rate value was determined at operating conditions

following an OTSG inspection and leak testing with a drip and bubble test. These cold leak tests conducted before the hot test program demonstrate that no single tube is causing all of the current 0.02 GPM (1 GPH) leakage. The results from these sensitive cold leak tests showed that the baseline leak rate value is and will be in the future the sum of multiple minor leakpaths which would not be expected to individually jeopardize the integrity of any OTSG tube.

Based on these considerations, a nominal leak rate of 0.1 GPM, above a baseline value, was established as the limit at which the plant is to initiate an orderly shutdown for OTSG inspection and identification of the leak source.

Q9. Can leakage commensurate with the license condition limit be realiably measured during plant operation?

A9. Yes. Primary-to-secondary leakage is indicated by several diverse methods at TMI-1. These methods include measuring radionoble gas concentrations on the secondary side, and measuring chemistry and radio-chemistry in secondary side OTSG water. The radionoble gas concentration measurement is the most sensitive method of quantifying the primary-to-secondary leakage rate. The leakage rate is calculated periodically by utilizing data from on-line continuous monitors and grab samples analysis. The following describes the measurement technique and our evaluation of the sensitivity of this measurement. The purpose of this description is to demonstrate that the leak rate value obtained by this measurement technique is

sufficiently sensitive relative to the proposed license condition limit.

Primary side activity is transported to the secondary system via the OTSG leakage pathway, and then carried over into the main steam system. The main steam, condensate and feedwater systems distribute the primary leakage throughout the secondary side of the plant. Non-condensable gases entrained in the steam and condensate are concentrated and removed from the system via the condenser air removal system. A measurement of radionoble gas activity in the discharge of the air removal system can be correlated with primary to secondary side OTSG leakage.

The measurement of gaseous activity is accomplished by instrumenting the vacuum pump discharge stream and providing a direct readout of condenser air removal system rate and activity concentration, and/or by taking local samples and then determining the OTSG P-S leak rate via calculation. At TMI-1 the radiation monitoring instrument provided to determine the activity measurement is a Beta scintilation detector designated RM-A5L. The instrument is located in the main condenser air removal system discharge common eight-inch diameter header. The monitor is manufactured by Victoreen, Inc., and includes a detector assembly consisting of a beta sensitive plastic crystal, optically coupled to a photomultiplier tube. The readout associated with the monitor is located in the control room. Based upon the control room readout and condenser air removal

system flow rate, leakage can be calculated as a function of RM-A5L effiency and reactor coolant system activity.

Licensee has evaluated the sensitivity of the RM-A5L monitor to determine its suitability for measuring primary to secondary leakage. For the expected ranges of condenser offgas flow, reactor power and failed fuel, we have concluded that the sensitivity is at least 0.001 GPM (0.07 GPH) during steady state operation (power operation) and 0.003 GPM (0.2 GPH) during plant cooldown (sub-critical conditions). The higher sensitivity during power operation is due to higher concentration of short half life radioisotopes in the reactor coolant system when the reactor is in operation. Thus, the measurement technique being utilized at TMI-1 is sufficiently sensitive to support the 0.1 GPM licensing condition.

Q10. What cold leak tests are utilized to determine the location of leaks and what is their sensitivity?

A10. There are two cold leak tests used to locate leaking tubes, the bubble test and the drip test. The bubble test is conducted by pressurizing the secondary side of the OTSG with nitrogen at about 135 psig. During this test the secondary side is partially drained and primary side water is maintained a few inches above the upper tubesheet. The inspector then looks for gas bubbles at the upper tubesheet bubbling through primary side water which is being maintained several inches above the upper tubesheet. Licensee has evaluated bubble test sensitivity and determined it is the most sensitive cold leak

test. Based on bubble test experience, an 80 mil diameter bubble originating once every five seconds can be located during the bubble test. This correlates to a leak rate sensitivity of 0.000005 GPM for any individual leak. The bubble test was used to test about the top 18 feet of the 56 foot long OTSG tubes. Testing this upper portion of the OTSG tubes results in testing 100% of the new kinetic expansion joints.

The entire OTSG tube length is leak tested by the drip test. The drip test is conducted by pressurizing the secondary side to approximately 150 psig. During this test, the OTSG is full of water on the secondary side and drained on the primary side. The inspector looks for drops of water coming from individual tubes on the primary side of the lower tubesheet. Based on the ability to locate one drop every three seconds, the sensitivity of the drip test is as low as 0.0002 GPM for any individual leak located at or near the lower tubesheet. For leak locations higher in the OTSG, the drop has further to travel before it can be observed at the lower tubesheet. This allows more time for evaporation of the leakage water before the water can drip down and out the bottom of the tube. This evaporation will reduce the drip test sensitivity somewhat. Even so, the drip test sensitivity for leak locations high in the OTSG remains quite good, and is estimated to be about 0.002 GPM (three drops per second).

Q11. What is the relationship or relevance of the leakrate measurements to the repairs made on the TMI-1 OTSG tubes?

A11. The leak rate measurements made at TMI-1 measure total P-S leakage from the OTSGs. This would include the contribution from leakage through the joints. As previously described, if the nominal leak rate increases by 0.1 GPM, the plant will be shut down and the individual tubes, plugs and/or joints will be identified by the nitrogen bubble test and drip tests which we discussed earlier.

Q12. Could leaks be self-sealing?

A12. Yes, in certain limited circumstances. We believe there may be a tendency for some leaks to be self-sealing, but only for leakage pathways between the expanded portion of the joint and the tubesheet. The joint is formed between the Inconel tube and the carbon steel tubesheet. Since carbon steel has a propensity for general corrosion in a normal RCS chemistry environment, corrosion products are formed in the tube-to-tubesheet joint. Industry experience indicates that these corrosion products tend to plug up leakage paths in the tight tube-to-tubesheet crevice and to stop or slow (i.e., self-seal) leakage. A trend of decreasing leakage with time for joints tested in the qualification program further confirmed this industry experience.

To be self-sealing, a leak past the joint would have to have a very small flow through a pathway sufficiently tight to

enable the build-up of corrosion products adequate to seal the leak. A leak of this size would not adversely affect the load bearing capability of the joint, or increase the probability of rupture within the joint.

Q13. Would the loss of pretension affect the usefulness of leak testing of the repaired joint?

A13. No. Leakage past a repaired joint is independent of the loss of pretension.

Pretension, or preload, was originally placed on the tubes during the manufacturing of the steam generators. The tubes were heated, which elongated them slightly by thermal expansion, and were then attached at each tubesheet. When the tubes cooled, the metal would have tried to contract back to the original length at ambient temperature, but because the ends remained fixed, contraction was prevented. This produced a tensile load on the tubes. At TMI-1, some tubes with complete circumferential cracks were freed from the original joint which fixed the tube in the upper tubesheet. These tubes contracted a small fraction of an inch, relieving all or part of the pretension. When the kinetic expansion was performed on these tubes, the tubes were again fixed at each end, but with the absence of part or all of the original pretension. This "loss of pretension" resulted in a reduction of axial tube load of only several hundred pounds.

The kinetic process relies on horizontal forces to expand the tubes, while pretension is an axial load (i.e., vertical in

direction). Since these load components are perpendicular with respect to each other, the loss of pretension does not affect the ability to expand the tube and form the new joint. Thus, kineticly expanded joints formed in tubes with loss of pretension are as tight, and therefore are no more prone to leakage, than tubes with preload.

Even if there is leakage past the repair joint, it will be through the tight crevice between the tube and tubesheet. The loss of pretension does not affect the tightness of this joint and thus can not affect the potential leakage flow path once fixed. Monitoring of leakage through such a joint is thus unaffected by a loss of pretension.

Q14. Would loss of pretension cause IGSAC cracks to be masked due to decreased leakage?

A14. In theory, a tube without pretension would exhibit a lower leak rate than a tube with pretension for a circumferential through-wall crack of a given size. In practice, however, this phenomenon is unlikely to mask the detection of a critical size crack at TMI-1.

The rigorous testing already conducted on each tube--special eddy current testing, bubble testing and leak-testing--show that such cracks do not exist in the tube pressure boundary. While the conditions which caused the circumferential intergranular stress-assisted cracking in TMI-1 have been eliminated, if such a crack were to exist, it would propagate only during conditions when the tube was placed in axial tension; this will tend to offset the effect of pretension loss.

Tubes without a pretension load are placed in axial tension under some operating conditions, just as tubes with preload are sometimes in axial compression. During the steam generator hot testing program, transients placed axial tensile loads of at least several hundred pounds on every tube in the steam generators--even those which had lost preload. Measured leak rates were assumed to come entirely from one crack, and were compared with benchmark calculations of estimated leakage through cracks of a significant size under the transient load. Tubes both with and without preload were considered. These results confirmed the conclusion reached after eddy current, drip and bubble tests--that no large cracks remain undetected in tubing in the TMI-1 steam generators.

If future cracks are hypothetically assumed to be propagating due to IGSAC at normal operating conditions, the principal direction of propagation will be axial along the tube. IGSAC propagation is principally perpendicular to the direction of highest stress. The highest tube stress is in the hoop direction at these conditions. A loss of pretension will not cause reduced leakage from axial tube cracks because there are no forces associated with loss of pretension trying to keep the crack closed.

PROFESSIONAL QUALIFICATIONS

Richard F. Wilson
Vice President, Technical Functions
GPU Nuclear Corporation

GPU Experience:

Technical responsibility for the Engineering, Design, Licensing and Technical Support of all nuclear generating stations for the GPU System. The position manages the technical resources of GPU Nuclear including day-to-day support for plant operations.

Previously was Acting Director for TMI-2 from September, 1979, to about March, 1980, and before that was Director of the Engineering and Quality Assurance Departments within the GPU Service Corporation. Between 1975 and 1977, was Manager of Quality Assurance for the GPU Service Corporation with responsibilities for design and construction Quality Assurance.

Other Experience:

Prior work experience included two years (1973-1975) as Manager of Manufacturing Engineering for Offshore Power Systems, Jacksonville, Florida. Responsibilities included activities associated with manufacturing planning, tooling, industrial engineering, manufacturing engineering, and technical support to the planned manufacturing facility. Prior to joining Offshore Power Systems, held a number of positions at the Atomic International Division of Rockwell International, 1954 to 1973. Some of these positions included Engineering Supervisor, Department Manager, Chief Project Engineer, Program Manager, and Chief Program Engineer on a wide variety of Atomic International programs. The last position was Program Manager for the Atomic International work on the fast breeder program. Performed and supervised work in almost every facet of reactor engineering, physics, facility design, safety, reactor operations, etc.

Committee affiliations have included the EEI QA Task Force, the AIF Committee on Power Plant Design, Construction and Operation, B&W Plant Owners and BWR Owners Groups, EPRI Nuclear Divisional Committee, etc. Outside the utility industry has served on a number of company and company/government advisory groups as related to specific programs.

Education and training includes a B.S. degree in Mechanical Engineering, University of California at Berkeley, 1951; an M.S. degree in Mechanical Engineering, University of Michigan, 1953; and one year attendance at the former Oak Ridge School of Reactor Technology in 1954. Has attended a large number of management and other courses, including the University of Michigan Public Utility Executive Program.

PROFESSIONAL QUALIFICATIONS

DAVID G. SLEAR

WORK EXPERIENCE

Company:	GPU Nuclear Corporation
Title:	TMI-1 Manager Engineering Projects
Responsibilities:	Management of TMI-1 modification, which entails: Management of the $25 million annual budget allocated for plant modification; prioritization of the various phases of plant modification; oversight of the technical adequacy of plant modification and of the components involved in plant modification; consultation regarding problem resolution with respect to matters concerning plant modification; and direct supervision of 16 GPU employees. This position demands constant attention to long term and daily plant modification concerns and an extremely firm grasp of both the technical aspects of TMI-Unit 1 and of the various modes and components of modification available for implementation at TMI-Unit 1.
Dates:	1983 - Present

Company:	GPU Nuclear Corporation
Title:	OTSG Repair Project Manager
Responsibilities:	Management (in conjunction with individual task managers) of all aspects of the OTSG Recovery program at TMI-1 including failure analysis, eddy current testing, corrosion testing, RCS examination, RCS sulfer cleanups, and plant performance analysis. This position involved direct management of the OTSG repair process and personal involvement in the decision making process with respect to the repair program. This position also entailed the definition and implementation of the overall project, and required a broad overview and analysis of the OTSG Recovery program. In his capacity as OTSG Repair Project Manager, Mr. Slear was also called

upon to deliver numerous presentations concerning project details before the NRC, ACRS, TPR, and the GPU Nuclear Corp. management.

Dates: December 1981 - November 1983

Company: GPU Service Corporation

Title: TMI-1 Manager Engineering Projects

Responsibilities: Similar to those listed for Mr. Slear's present position including management of a $20 million budget and of project engineering for modifications.

Dates: 1979 - 1981

Company: GPU Service Corporation

Title: Preliminary Engineering Manager

Responsibilities: This position entailed: the analysis and preliminary design of 400 Megawatt combustion turbines and of a 600 Megawatt coal fired power plant; extensive analysis of the reliability and availability of the components to be installed in the prospective power plant; and the establishment of a baseline criteria document for the designated plants including the technical documentation and presentation of the plant design for management review.

Dates: 1978 - 1979

Company: GPU Service Corporation

Title: Component Engineer

Responsibilities: This position entailed: the review of design specifications and technical details of products going into TMI-2, including the steam generators, pressurizer, main

condensors, cooling towers, reactor vessel, and internals; technical consultation and analysis of problems; and review of the contractor's design work on new components going into a plant.

UNITED STATES NAVY NUCLEAR SUBMARINE FORCE OFFICER

Title: Engineer Officer

Responsibilities: This position entailed: essentially primary responsibility and control of the onboard nuclear power plant; control of all engineering sections, command of 4 divisions; and supervision of approximately 55 crewmen.

Dates: 1972 - 1974

Title: Machinery Division Officer

Responsibilities: As Machinery Division Officer, Mr. Slear was responsible for: all mechanical components of the primary and secondary systems of the power plant including the steam generator, reactor, and drive controls; chemistry control of the primary and secondary systems; and the supervision of 15 crewmen. Mr. Slear also served as an Auxiliary Division Officer in charge of non-nuclear life support systems, and as a Communications Division Officer.

Dates: 1968 - 1972

Mr. Slear also attended the Nuclear Power Submarine School from 1966 - 1968, during which time he obtained one year of nuclear power plant training (6 months classroom, 6 months actual plant training) in addition to the submarine qualification program.

EDUCATION

College: University of Oklahoma

Degree: B.S. Mechanical Engineering

Dates: 1961 - 1966

College: Stevens Institute of Technology

Degree: M.S. Mechanical Engineering

Dates: 1974 - 1978

PROFESSIONAL QUALIFICATIONS

Don K. Croneberger
Director - Engineering & Design
GPU Nuclear Corporation

GPU Experience:

Technical responsibility for the Mechanical, Electrical, Civil/Structural, Chemical, Radwaste and Materials Engineering support for all nuclear generating stations for the GPU Systems.

1978 to 1980 was Manager - Design and later Manager - Engineering & Design with GPU Service Corporation. Directed design engineering activities for all nuclear and fossil power generating facilities and modifications assigned to GPUSC.

Other Experience:

Prior work experience included a number of positions at Gilbert/Commonwealth during the period 1963 to 1978. The last position was Manager Structural Engineering. It included technical responsibility for structural engineering mechanics for all nuclear and fossil generating facilities. Some of the other positions included Project Manager for balance of plant studies for a liquid metal fast breeder reactor demonstration plant. Other positions as Project Structural Engineer included responsibility for technical supervision of structural engineering and engineering mechanics for a number of domestic nuclear power plants. Earlier experience with the U.S. Navy included engineering and construction of radio telescope and ancillary experience.

Industry affiliations have included the EPRI Steam Generator Owners Group, ASME Section 3 Division 2 (former Chairman) and other industry nuclear standards activities including Nuclear Structures and Plant Design Against Missiles.

Education and training includes a B.S. degree in Civil Engineering from Pennsylvania State University, 1959. Other technical training includes courses at U.C.L.A., M.I.T. and the University of Michigan.

I have been involved in the Steam Generator tube failure issue from the beginning. I provided technical management oversight of failure analysis and repair activities. Special emphasis was placed on understanding the mechanical design of the Steam Generators and applying that understanding to the repair program and the understanding of the impact of the repair on the response of the components.

My department provided engineering support in the areas of Materials Engineering/Failure Analysis, Chemical Engineering and Chemistry, Mechanical Engineering and Engineering Mechanics.