GPU Nuclear

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Writer's Direct Dial Number:

April 30, 1982
5211-82-104

Office of Nuclear Reactor Regulations
Attention: D. G. Eisenhut, Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

The purposes of this letter are:

- To provide an updated status of our work on the TMI-1 Steam Generator Failure Analysis.
- To outline the proposed repair and our plans to complete needed safety evaluations and fulfill regulatory requirements.
- To request early NRC action to confirm that you agree with our approach including the fact that hearings are not needed and will not be required by the NRC.

We recognize that the possibility of an unreviewed safety question must be considered in accomplishing the repair program itself, as well as in the return of the steam generators to service. We will, of course, carry out the required safety evaluation and submit it to the NRC whether or not an unreviewed safety question is identified.

The material enclosed is, in large part, the same as that presented to the NRC Staff and consultants, in the publicly noticed meeting on April 7, 1982.

While some confirmatory work remains to be completed, we are confident that the repair program will fully resolve safety concerns associated with the damaged tubes and result in return of the steam generators to a condition meeting all design and safety performance criteria.

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As identified in the course of our presentation and discussed with you and your Staff late that day, we believe that return of the TMI-1 steam generators to service following repair can be accomplished without any degradation in their ability to meet all design requirements, thusly no significant hazard exists. The basis for these conclusions is included in the attachments and can be summarized as follows:

- The failure mechanism is now known.
- The attack is highly localized within the upper tubesheet. It occurred when the plant was cold.
- This location allows for repair in such a way as to return the Unit to within design conditions.
- From a safety standpoint, the location of attack and the fact that it occurred when the plant was cold do not pose the significant increased risk of large abrupt increases in primary to secondary leakage typical of other steam generator tube problems.

In addition to our normal safety reviews and evaluations of this situation, we have established an independent safety review group of highly qualified experts for this matter. The group will provide a final written assessment which will be made available to the NRC. Additional information on the scope and composition of the group is provided in the attachments.

On April 7, 1982 we also discussed with the Staff whether a public hearing on the steam generator repairs would be required or recommended prior to resumption of operations. We noted that, even if, contrary to our expectations, a license amendment is necessary, we believe that we and the Staff would conclude that no significant hazards considerations were involved.

Return of TMI-1 to service is of great importance to the Company and its customers and to proceeding with the cleanup of TMI-2. If it were now known that a public hearing were to be held or offered, we would, of course, be urging prompt issuance of an appropriate notice in order that at least the hearing preliminaries could be accomplished in parallel with the Staff's review. However, we see no safety considerations which would warrant the time and expense of a public hearing; and, a hearing, even if noticed at this time, would, in all probability, delay acceptance of the program many months beyond completion of the Staff review.

In view of the information above, we request written advice as to whether the Staff would be prepared to take the position (1) that a public hearing, whether or not requested by others, would not be legally required to be held prior to issuance of a license amendment, if needed, and resumption of operation, and (2) that the staff would not on its own initiative propose such a hearing.

We recognize that the so-called Sholly Amendment to the NRC Authorization Bill may be enacted. Under the pending legislation, we understand the Commission would be required to issue implementing regulations concerning the criteria for determining whether an amendment involves no significant hazards consideration and other procedural matters. Therefore we would urge that the Staff reviews of this matter be made with the possible need for meeting such criteria in mind.

An early reply to this letter including the basis for any NRC disagreements with our position is requested.

Very truly yours,

H D Hukill

H. D. Hukill
Director, TMI-1

HDH/klk
Attachments

cc: R. Haynes
J. Stolz
R. Jacobs
T. Novak

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of METROPOLITAN EDISON COMPANY (Three Mile Island Nuclear Station, Unit No. 1)	))))))	Docket No. 50-289 (Restart)

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April 28, 1982

ATTACHMENT I

OTSG Repair Technical Summary

The following information provides a summary of the failure analysis program, repair plan/program and licensing aspects of the TMI-1 OTSG tube leak problem. The discussion refers to the various slides which were used on April 7, 1982, to present a status review to the NRC. These slides have been bound and are attached hereto.

I. Failure Analysis Program

A. What Happened?

Three Mile Island Unit 1 has remained in a shutdown condition for approximately 3 years as ordered by the Nuclear Regulatory Commission on July 2, 1979. Slide 4 shows the primary system pressure and temperature during this shutdown period. As shown, the primary coolant system was heated up and pressurized in late August/early September, 1981 for hot functional testing (HFT). This HFT was conducted to test equipment and plant modifications in anticipation of completion of the TMI-1 restart hearings. Subsequent to this heatup, the plant was again placed in a cold shutdown condition. In late November the primary system pressure was increased to 45 psi for further equipment testing. At this point in time, both activity measurements and boron contamination on the steam generator secondary side indicated the presence of primary to secondary leakage. The primary coolant system was depressurized and drained down to a level just above the steam generator upper tubesheets. The secondary side was pressurized with nitrogen to approximately 15 pounds and a visual inspection was conducted in order to identify the leaking steam generator tube or tubes. During this testing in early December, a total of over 130 tubes were identified as leaking in the two steam generators and a non-destructive examination of the OTSG tubes using eddy current techniques was commenced. The initial eddy current examinations indicated that there were probable defects in thousands of tubes and GPUN established an internal task force to evaluate the status of the TMI-1 steam generators and reactor coolant system.

B. What Resources Are Assisting GPUN?

As soon as GPUN realized the extent of damage to the steam generators, an extensive failure analysis and industry experience review was commenced. GPUN requested assistance from numerous experts in the nuclear power industry and research laboratories as detailed on Slides 2 and 24. To date, 19 failed tubes have been removed from the TMI-1 steam generators with independent failure analysis being conducted by B&W, Battelle Columbus and Westinghouse. There are plans to pull an additional six to ten tube samples to be used to confirm the status of steam generator tubes that have been accepted by eddy current evaluation and for corrosion testing.

C. What Do We Know?

1. Eddy Current Results

Eddy current examination of the steam generator tubes proceeded during December and early January utilizing the standard differential probe and multiple frequencies. This examination has proven to be an acceptable examination technique in the areas of the steam generator tubes between the upper and lower roll transitions. The adequacy of the standard differential probe has been demonstrated by both 100% correlation with defects in the removed tube samples and good correlation between the standard circumferential coil and pancake coil and the standard differential probe and absolute probes. By an empirical process, the eddy current inspection contractor (Con Am) identified a frequency mix using 400 kilohertz and 800 kilohertz signals which minimized the roll transition* signal and improved the sensitivity to detect defects at this location. We later confirmed by the inspection of tube samples that defects existed at the roll transition. However, it was soon recognized that the standard differential probe could not adequately define the status of the roll transition, tube exit, and heat affected tube area because of the signal interference from the transition and exit. Therefore, GPUN in conjunction with Con Am and Zetech developed an absolute probe with four coils in order to inspect these areas at the very top and bottom of the tubes. In addition, the eddy current manipulator was modified to allow rotation of the new 4x1 probe resulting in 360° coverage. Mockup testing has confirmed the adequacy of the 4x1 probe to detect defects 40% through wall or greater. The 100% correlation with metallurgical samples also confirms the adequacy of the 4x1 probe to detect the type of defects that exist in the TMI-1 OTSG.

Although eddy current examination of the steam generators is continuing, GPUN has projected the probable defect population as a function of both radial and axial position within the steam generator. These projections are shown on Slides 6, 7 and 8 and result in an estimate of between 8,000 and 10,000 tubes which will require repair due to defects at or near the roll transition. These projections specifically exclude the location in the tube immediately below the seal weld to the upper tubesheet. The new 4x1 probe is qualified to detect defects at this location but both the number of tubes that are defective at this location and the implications of defects at thi location have not yet been finally determined. However, it is expected that a majority of all the tubes will be shown to have defects in this location. The existence of cracks in the tubes at this location was first noticed by metallurgical examinations of the pulled tube samples.

*See Slide 22 for location of the roll transition.

ATTACHMENT I (cont'd)

2. Failure Analysis

The failure analysis activities have centered around the metallurgical analysis of tube samples, the chemical analysis of TMI-1 water samples, operating history review, OTSG fabrication history review and tube stress analysis. These activities have been aimed at identifying the combination of material condition, tube stress and aggressive environment that caused damage to the OTSG tubes at TMI-1. The failure mechanism should, based on evaluating the combination of these conditions, be able to explain the timing of cracking, the material failure mode, the contaminant source and the axial/radial crack distribution as identified by eddy current.

With regard to material conditions, both the fabrication history and the metallurgical analysis of the tube samples indicate that the tube material is sensitized. Based on the tube metallurgical analysis, the cracking has been caused by intergranular stress corrosion initiated from the inside diameter. In order to explain the circumferential orientation of the cracking, it is necessary to determine when the axial stress in the tubes exceeded the hoop stress. The fabrication of the steam generator involves installing the tubes with an axial tensile preload in order to minimize the potential for tube buckling during operation. In addition, cooldown transients apply axial tensile loads to the steam generator tubes. During periods of normal operating pressure and temperature, the hoop stress is tensile and the axial stress is compressive in the tubes. Thus, it has been concluded that the circumferential cracks identified in the TMI-1 OTSG tubes must have occurred during a period of cooldown or cold shutdown.

With regard to the environment, initial analysis of the contaminants on the cracked surface identified the presence of sulfur. An extensive evaluation of potential sulfur contamination sources and the existence of sulfur contamination in TMI-1 systems was conducted. The results of this evaluation are summarized on Slides 32 and 33. Although there are several potential sources for sulfur contamination, an evaluation of their concentrations, volumes and dilution in the reactor coolant system has led to the conclusion that the most probable source of sulfur contamination in sufficient quantities to result in aggressive concentrations in the reactor coolant system is the sodium thiosulfate. The scenario for the injection of sodium thiosulfate involves: 1) leakage from the sodium thiosulfate tank to the suction of the building spray pumps over the long period of shutdown, 2) surveillance testing of the building spray pumps in June, August and September of 1981 in anticipation of TMI-1 restart (resulting in addition of sodium thiosulfate contamination to the BWST), 3) subsequent

ATTACHMENT I (cont'd)

injection of BWST water into the reactor coolant system prior to and during the August/September, 1981, hot functional testing (HFT) and 4) reduced (aggressive) sulfur species formed during HFT.

The overall summary of the failure analysis status as of April 7, 1982, is detailed on Slide 26. It indicates that the TMI-1 OTSG tube cracking is caused by intergranular stress corrosion, initiated on the ID surface, and sulfur is identified as the contaminant which most probably caused the cracking. The cracking probably occurred subsequent to the RCS heatup and cooldown that was conducted in late August/early September, 1981, and was then detected the next time the reactor coolant system was pressurized in late November 1981. As noted on Slide 31, the primary system water level subsequent to the August/September heatup varied from about the 13th tube support plate to the upper tubesheet during the month following the heatup. It is believed that this water level is important in that it provided an opportunity for concentration of the contaminant on the tube surface in sufficient quantity to initiate and rapidly propagate the tube cracking.

3. Corrosion Testing

GPUN has initiated numerous corrosion tests in order to duplicate and, therefore, confirm the failure scenario, identify whether or not the water currently in the TMI-1 reactor coolant system is aggressive, and confirm that subsequent to the planned repairs the TMI-1 OTSG tubing will be acceptable for continued operation. In addition, a testing program is being implemented to identify what reactor coolant system clean up of sulfur contamination if any is required and how this clean up should be accomplished. This program is summarized on Slide 83 with hydrogen peroxide identified at this time as a likely candidate additive for RCS cleanup. The preliminary corrosion test results are summarized on Slide 35.

4. Reactor Internal Inspection

In order to confirm that other materials within the reactor coolant system have not been damaged by the environment which caused the steam generator tubes to crack, GPUN has initiated a comprehensive review and inspection of reactor coolant system components and materials. This program involves removal of the reactor vessel head and the inspection or test of approximately 1000 items. The objectives, program plan and inspection plan are summarized on Slides 44 through 47. The program involves classifying items by material condition, environmental exposure, applied stress, and safety significance. Candidates have been selected for inspection and testing that are representative of the worst metallurgical, chemical or stress conditions expected with an additional objective of minimizing man-rem exposure as much as practical. This inspection program is well underway and has shown no evidence of attack on any reactor internals to date.

D. What Future Work Is To Be Done?

Slides 9 and 88 provide a summary of the future work and/or decisions that still remain. This remaining work entails: 1) completion of the eddy current examination in the heat affected area and other work to complete determination of the status of OTSG tubes, 2) completion and evaluation of the inspection of other RCS components and materials, 3) establishing the need for and the approach to be used in cleaning up sulfur in the reactor coolant system, 4) selecting, removing and evaluating the final tube samples, 5) completion and evaluation of the corrosion test program, 6) the development, qualification testing, and implementation of a tube expansion process in order to repair the OTSG's.

II. Repair Plan/Program

A. What Is The Repair Criteria?

The repair criteria is summarized on Slides 48, 49 and 50. The criteria require adequate pullout load capability, primary to secondary leakage as low as reasonably achievable and within the reactor coolant leakage and radioactive effluent limits of the technical specifications, and confirmation that the repair will maintain the thermal and hydraulic performance within the acceptance criteria for both normal operating and design basis accident conditions.

B. How Will The OTSG Be Repaired?

The failure analysis information available to date continues to indicate that the large majority of the defects in the steam generator tubes are located within the first few inches below the upper surface of the upper tube sheet. The tube material below this area is essentially like new except in limited areas where surface intergranular attack and small pits several grains deep exist (as found by metallurgical examination) and where defects have been identified by eddy current. The local areas are considered typical of S.G. tubes after several years of service. In accordance with the technical specification requirements, GPUN plans to plug and remove from service all tubes with indications of defects outside the tube sheet area. For those tubes that are defective within the tube sheet, the repair plan is to expand the tube within the tube sheet below the defect indications so that the repaired tube has adequate strength and leak tightness.

ATTACHMENT I (cont'd)

Preliminary plans call for 1) cleaning the tube ID in the area of the repair, 2) heating the upper tubesheet tube crevice to drive moisture out of the crevice, 3) expanding the tube for a distance of approximately 8-10 inches from the top of the upper tube sheet in order to close the crevice, 4) cleaning the tube ID to remove residue from the expansion process (if required), 5) conducting a second expansion (if required for leak tightness or load carrying capability) approximately 1 inch long using a mechanical roll at a location near the bottom of the expansion and below any defect indications. Depending on the total number of tubes requiring repair GPUN may decide to conduct the expansion on all tubes. As shown on Slide 77, a 10 inch expansion leaves approximately 376 tubes with defects in the tube sheet that may either be plugged or expanded depending on current tooling and future sleeving considerations.

The repair program outlined above is possible because of the condition of the remaining tubing and the fact that the defects are located within the tube sheet area. Given these facts, it is possible to establish a new mechanical and essentially leak tight joint below the defect in a manner which is similar to the original configuration of the steam generator. The original configuration involved a 1 inch minimum roll at the top of the existing tubes. An extensive qualification program involving pullout load tests, thermal cycling tests and leak tests to be conducted over the next several months is expected to provide confirmation of the acceptability of this repair. Slide 60 and 61 provide a summary of the points to be addressed by the qualification program. In addition, GPUN's review of the industrial experience in both steam generator and other reactor components indicates that mechanical joints have been used successfully in numerous applications in order to form an essentially leak tight joint. GPUN intends that the repair process and program as currently identified with subsequent leak testing and leakage monitoring, will provide assurance that the probability of abnormal primary to secondary leakage during operation is very low and the generator will conform to its original design basis.

Slide 84 details the testing plans to be used to confirm the adequacy of the OTSG repair. These testing plans involve drip and bubble tests with a differential pressure of 150 psi and a leak test with a differential pressure of 1500 psi. The sensitivity during the bubble test is expected to be approximately .1 gal per day/tube and the sensitivity during the 1500 psi differential pressure test is about 10 gal. per day/steam generator. These sensitivities give us confidence that we can confirm adequate leak tightness of the steam generator prior to criticality, thus continuing to assure minimum risk to the health and safety of the public. Once the reactor is critical, leak rate will be monitored continuously with a sensitivity of approximately 10/GPD/OTSG. Corrective action will be taken to plug or repair steam generator tubes based on excessive total leak rate and/or the rate of change of leak rate.

ATTACHMENT 2

License/Technical Specification Conformance Summary

The TMI-1 Technical Specifications identify the key parameters that set the envelope within which the margins of safety originally reviewed and approved by the NRC will not be exceeded. The Code of Federal Regulation Title 10 Part 50.59 established conditions that if met will allow a licensed facility to be modified without obtaining prior NRC approval. In that section specific tests are outlined to guide licensees in properly making that determination. The tests are twofold. First the change must not require a change to the unit technical specifications and second, the change must not involve an unreviewed safety question.

An unreviewed safety question is defined in that part to exist "(i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margins of safety as defined in the basis for any technical specification is reduced."

At present we are evaluating the above criteria regarding the extent of damage, the cause of failure, and the repair method. This evaluation will be the subject of a future report to the staff. The completion of the repair activities as discussed in the Technical Summary and the attached slides is not expected to increase the possibilities or probabilities of events or consequences of events as compared to the current licensing basis.*

With regard to the TMI-1 Technical Specification, the key functions of the steam generator are retention of reactor coolant pressure boundary integrity, and the implied maintenance of adequate heat transfer capability and reactor coolant flow area. The following discussion highlight the portions of the Technical Specifications that focus on the maintenance of steam generator tubes as a portion of the reactor coolant pressure boundary. As noted in the attached presentation the material safety functions related to heat transfer and adequate RCS flow area should not be significantly impacted by the repair processes.

*See Northern States Power Company (Monticello Nuclear Generating Plant, Unit 1), DD-79-5, 9 N.R.C. 588, 591 (1979) for an example of the Staff's application of the definitions of "unreviewed safety question" found in 10 CFR 50.59. Note particularly the emphasis placed on the consequences of an accident or malfunction, rather than on the mechanisms of causation.

The TMI-1 Technical Specifications require in Section 3.1.1.2 that both steam generators be operable for temperatures above 250ºF. The basis of this specification is to insure system integrity against leakage under normal and transient conditions. One requirement in declaring the generators operable is execution of T.S. 4.19, which defines a periodic test program intended to demonstrate that the structural integrity of the tube portion of the OTSGs is maintained (Bases, T.S. 4.19).

T.S. 4.19 addresses specifically the tube area between the tubesheets as requiring inspection (T.S. 4.19.4.8) and defines accceptance criteria for that inspection area (T.S. 4.19.4). Further, the specification exempts from consideration leakage in the area of the tube-to-tubesheet joint (T.S. 4.19.3.C.1).

Not addressed is the area within the tubesheets excluding the tube-to-tubesheet joints, although clearly some portions of this area would be expected to affect the tube's structural integrity. This portion of the tube coincides with the location of the majority of cracks identified in the OTSG tubes. Therefore, GPUN has examined the regulatory and design bases for the OTSG tubes to determine for what portion of the tube area in the tubesheet the tube inspection acceptance criteria (T.S. 4.19.4) should be applied, and for what portion of the tube area in the tubesheet the acceptance criteria seem to be inappropriate.

This question was considered looking at a repaired steam generator tube, expanded to some point below our identified cracking. T.S. 4.19 and the Regulatory Guides 1.121 and 1.83, refer to the structural integrity, requirements applicability as the "tube portion of the reactor coolant pressure boundary" and the "heat transfer surface." The GDC in 10CFR50 use the same words in establishing design bases. It is the purpose of the expansion qualification program to demonstrate that the new tube to tubesheet joint will carry all normal and transient loads and remain essentially leaktight. The tube area above that joint would no longer be a part of the heat transfer surface or the reactor coolant pressure boundary. As such, it would be inappropriate to apply the acceptance criteria for the freestanding tube (T.S. 4.19.4) to the remainder of the tube, however, GPUN would apply the acceptance criteria of the balance of T.S. 4.19 and plug tubes, as required. After repair, the cracked areas are no longer part of the RCS pressure boundary or heat transfer surface.

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 concensus 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. The program will also include an evaluation of other portions of the reactor coolant system to determine if corrosion mechanism extended out of the steam generator boundaries.

B. Eddy Current Examination Program

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

C. OTSG Performance Evaluation

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

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.

Rev.3,4/27/82

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.

Rev. 3,4/27/82

ATTACHMENT 4

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 - Nuclear Analyst, 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 - Department Manager, Generic Safety Analysis Electric Power Research Institute

- PLANT OPERATIONS

  STAN HOLLAND - System Production Engineer - Duke Power Co.

- NON-DESTRUCTIVE EXAMINATION

  STEPHEN BROWN - Principal Engineer - Inspection, Electric Power Research Institute

- SECRETARY

  ED WALLACE - Manager, PWR Licensing, GPU Nuclear

GPU Nuclear

TMI-1
OTSG Status Review

April 7, 1982

~~8204260221~~

TMI-1 OTSG STATUS REVIEW

I.	INTRODUCTION	- R. F. WILSON
II.	FAILURE ANALYSIS	- DR. R. L. LONG
III.	STEAM GENERATOR REPAIR	- J. PEARSON
IV.	OTSG REPAIR PROGRAM OVERVIEW	- D. G. SLEAR
V.	CONCLUSIONS/SUMMARY	- R. F. WILSON

BETHESDA, MARYLAND

4/7/82

Organizations Actively Working with GPUN on Steam Generator Program

B&W, Lynchburg and Alliance Research Labs

EPRI and consultants

Battelle Laboratories

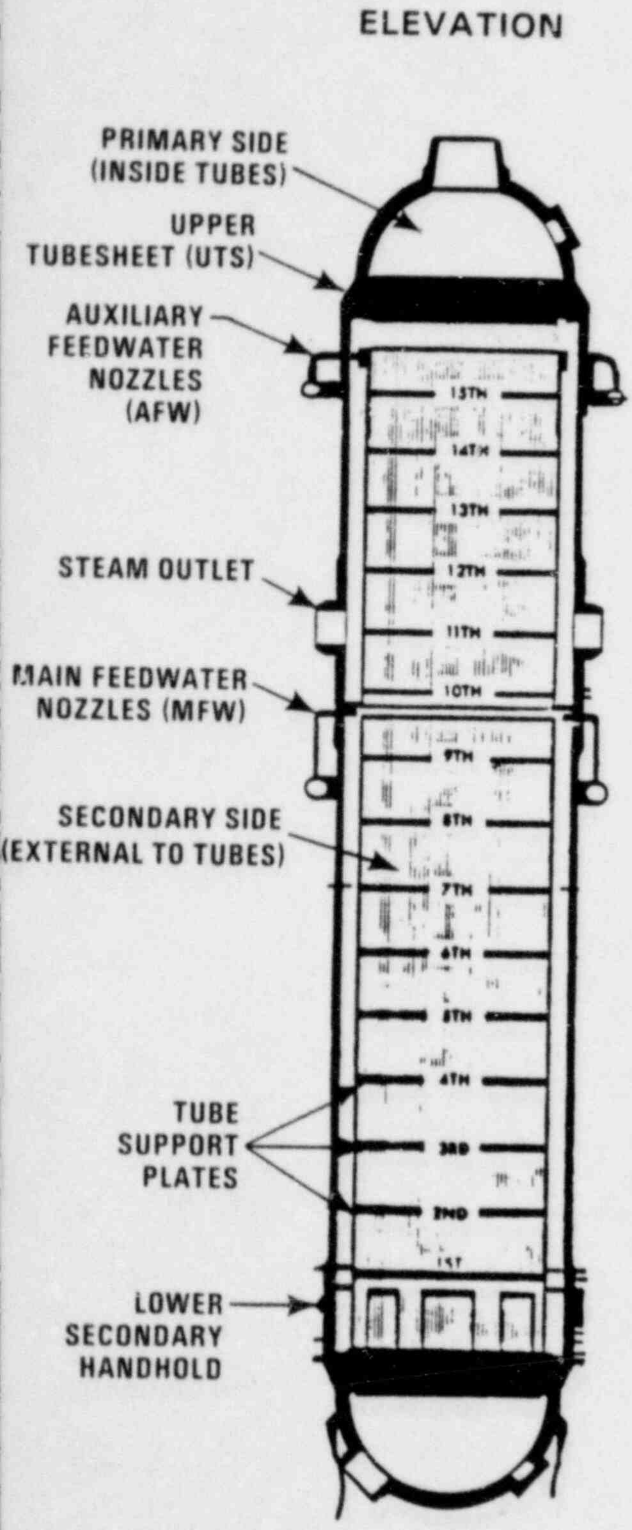
MIT

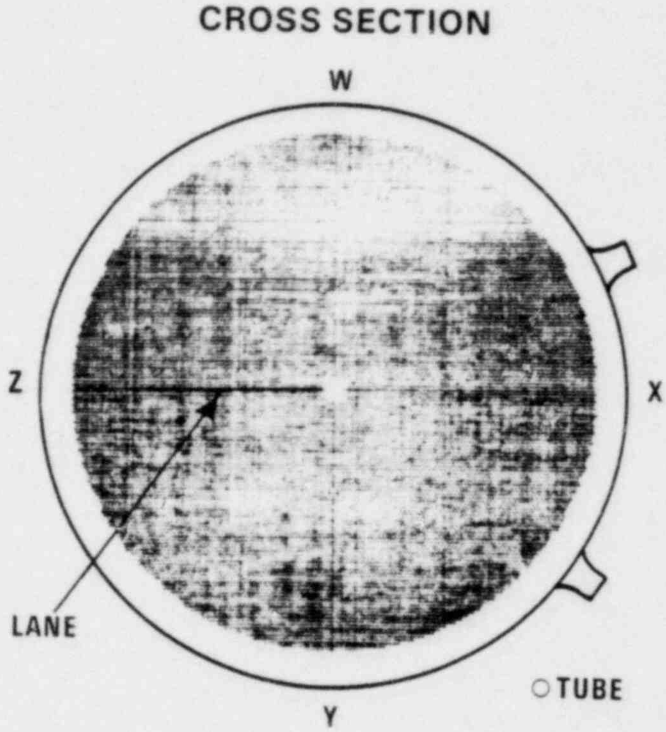
ORNL

CONAM *

* Inadvertently left out of slide used on 4-7

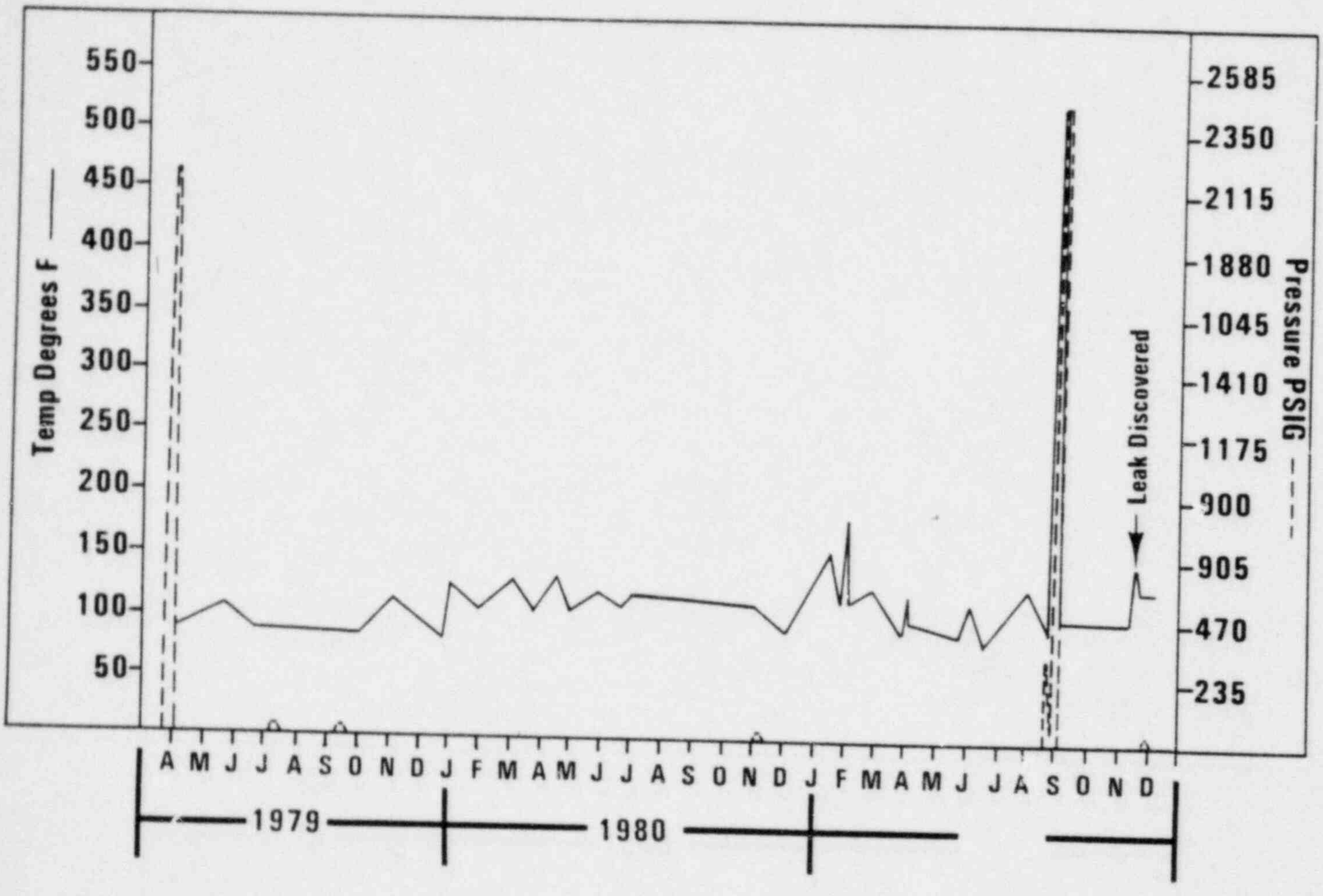
TMI-1 Steam Generator





Weight, operating	637 tons
Height	73 feet
Primary flow	$69X10^6$ #/hr.
Steam flow	$6.1X10^6$ #/hr.
Number tubes	15531
Tube size, material	0.625" od, .034 wall inconel 600
Manufacture date	5/69 to 11/70

Reactor Coolant System



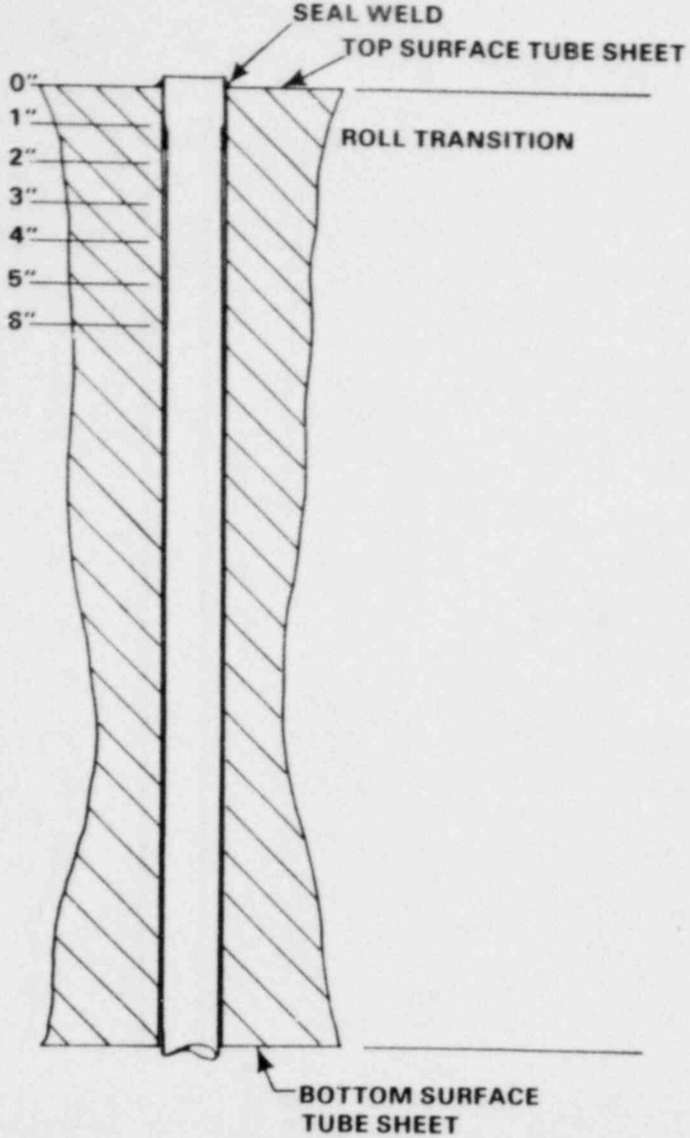
TMI-1 STEAM GENERATORS

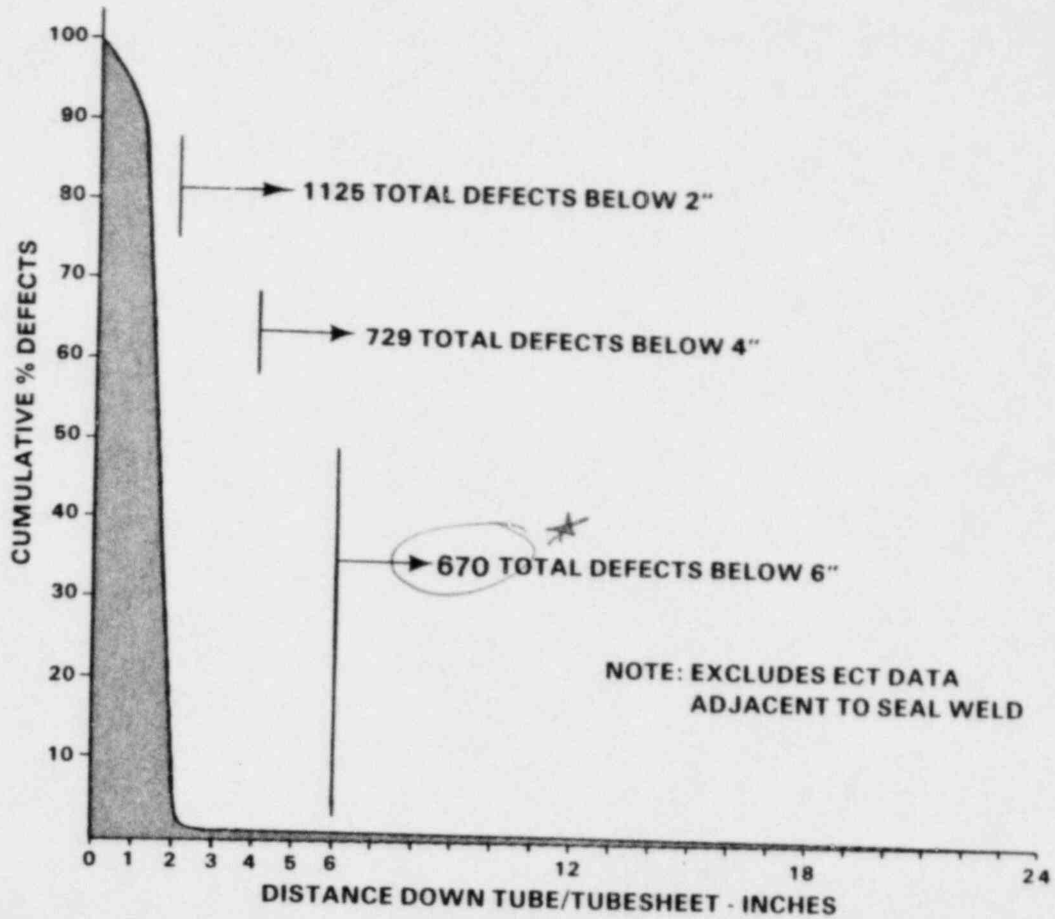
EXTENT OF ATTACK

- STEAM GENERATOR
  - NUMBER OF LEAKING TUBES — 200 TO 500
  - NUMBER OF TUBES WITH INDICATION OF SIGNIFICANT DEFECTS — 8000 TO 10,000
  - UNRESOLVED AREA AT SURFACE OF UPPER TUBESHEET — UNKNOWN
- ELSEWHERE
  - MATERIALS POTENTIALLY SUBJECT TO SIMILAR ATTACK USED ELSEWHERE IN REACTOR
    - EXAMINATION PROGRAM GETTING UNDERWAY
    - ATTACK, IF ANY, REQUIRES RIGHT COMBINATION OF MATERIAL CONDITION, STRESS, LOCAL ENVIRONMENT

TMI-1 Steam Generator

Tube Cracking

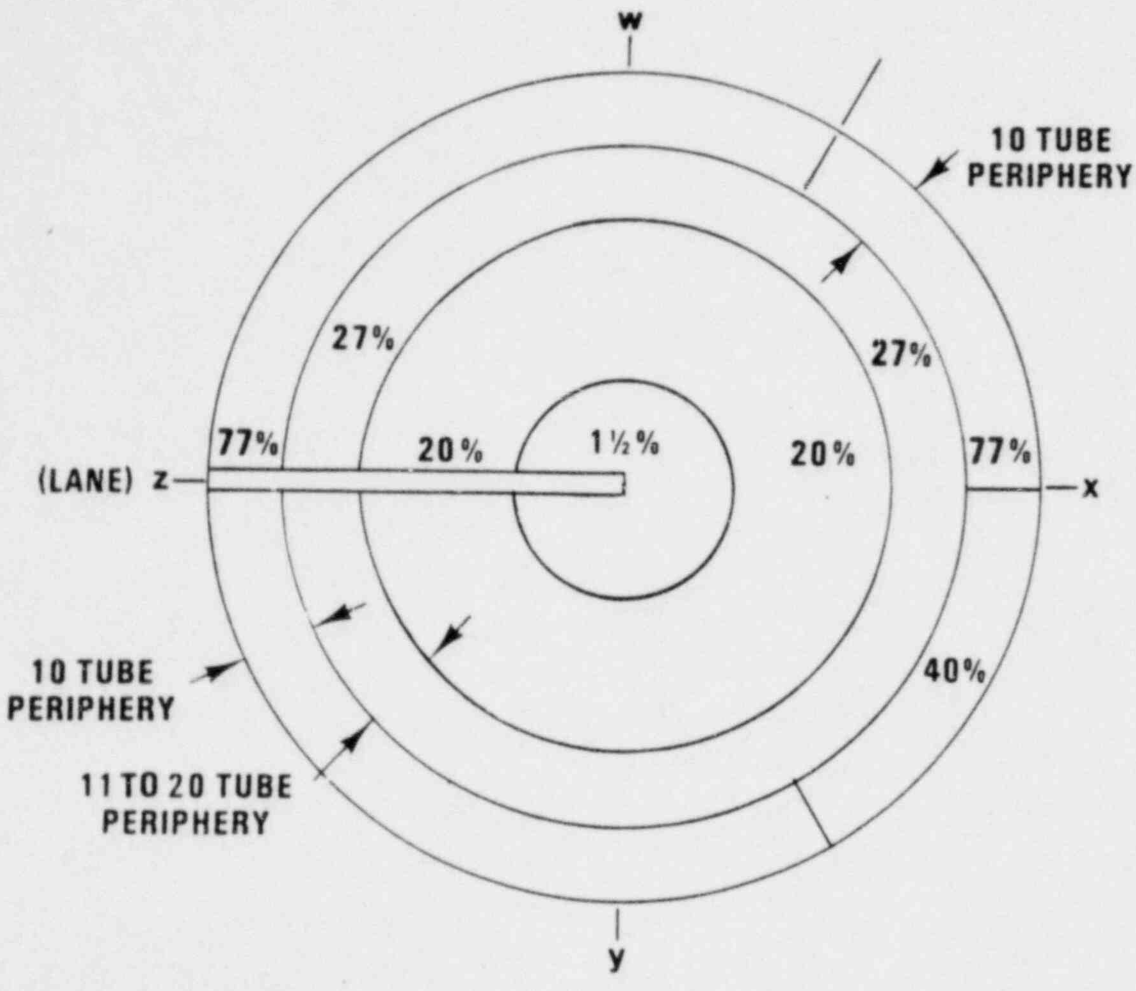




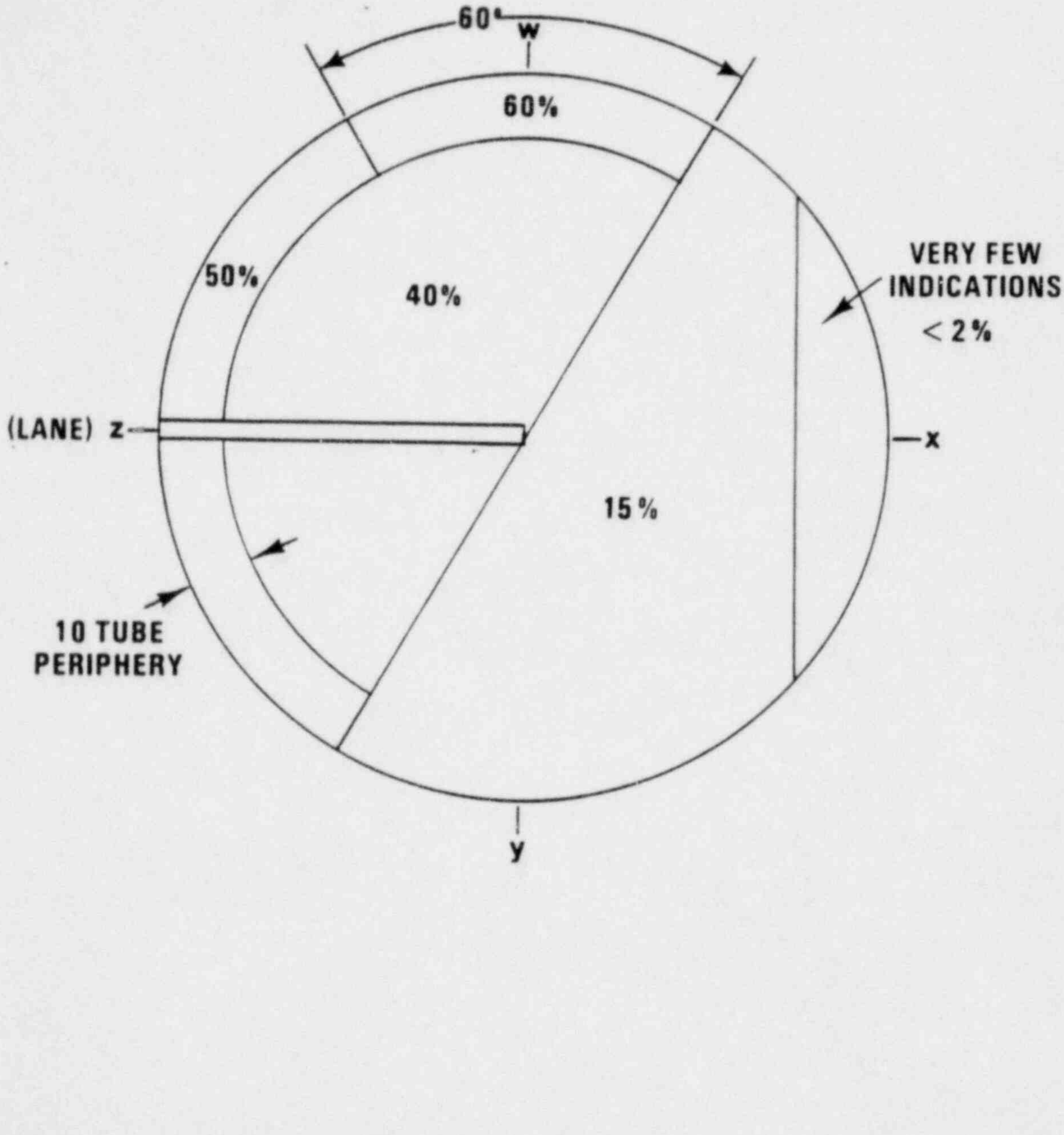
* Updated to reflect recent data as of 4-15-82.

Potentially Defective Tubes (Projection of Eddy Current Data)

OTSG-A



Potentially Defective Tubes
(Projection of Eddy Current Data)
OTSG-B



FUTURE WORK/DECISIONS REMAINING

- FINAL ECT AT ROLL TRANSITION AND UPPER END

- RCS INSPECTION

- FINAL RCS/SG CLEANUP METHODS/APPROACH

- REPAIR, DEVELOPMENT/QUAL. TESTING OF S.G. REPAIR

- FINAL TUBE SAMPLES/LABORATORY SIMULATION TESTS

TMI-1 TECHNICAL BASIS FOR REPAIR LICENSING

- THE S.G. REPAIR APPROACH IS

  INITIAL - ROLL SEAL EXISTING TUBE TO TUBESHEET TO ISOLATE LEAKS/TUBE DEFECTS

  LONG RANGE - SLEEVE OVER DEFECTIVE TUBES (INCLUDING ROLL SEAL) TO ISOLATE LEAKS/TUBE DEFECTS/ INITIAL REPAIR, IF AND AS REQUIRED

- S.G. DAMAGE IS UNIQUE IN INDUSTRY IN TWO IMPORTANT WAYS

  - LOCATED WITHIN THE UTS
  - DAMAGE MECHANISM OPERATES COLD/REACTOR SHUT DOWN

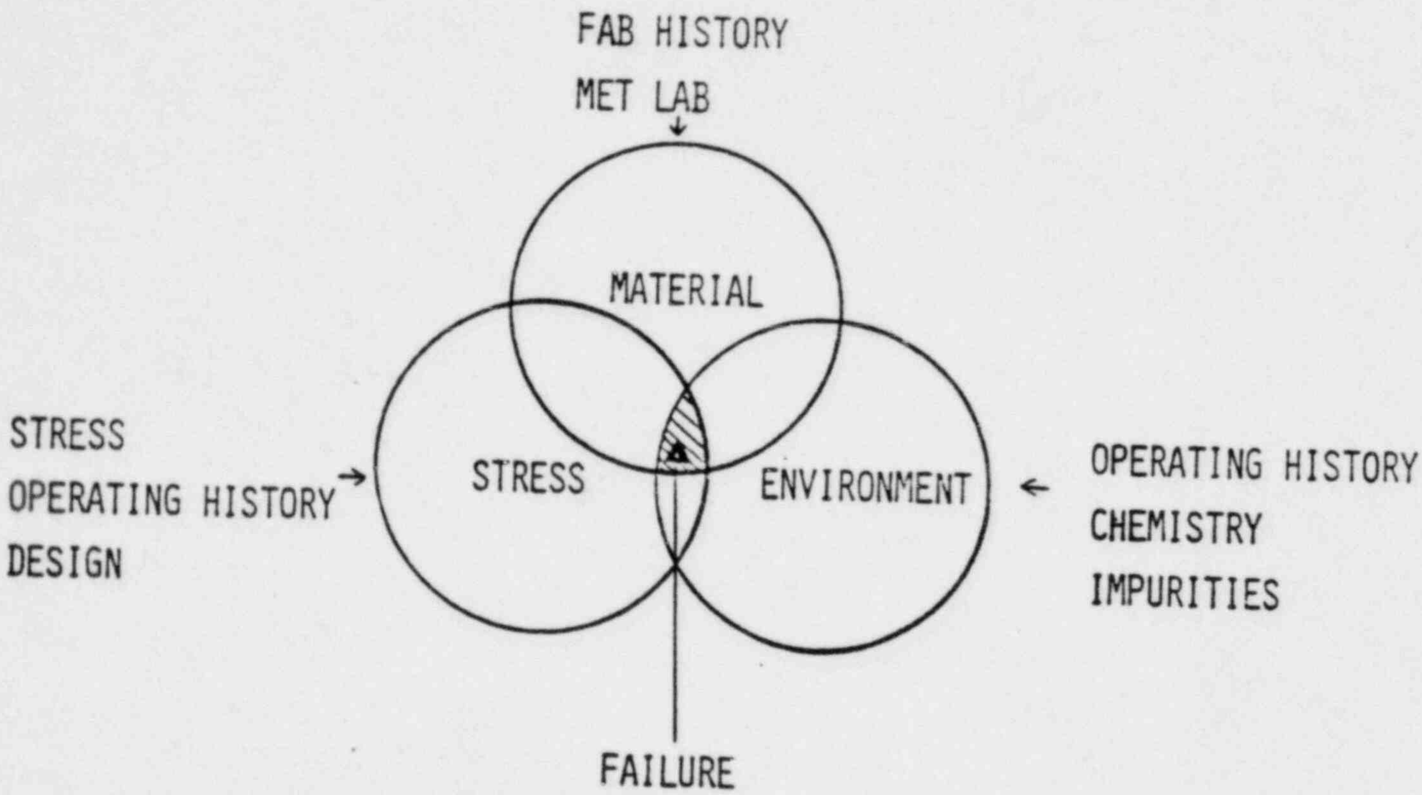
- THE REPAIR APPROACH ISOLATES THE FAILURES AND RESTORES THE S.G. TUBE TO ITS ORIGINAL FUNCTIONAL/DESIGN BASIS CONDITION

- BELIEVE NO INCREASED PROBABILITY FOR LARGE PRIMARY/SECONDARY TUBE RUPTURE OR ACCELERATED DEGRADATION OF TUBES IN SERVICE

- THERE ARE INSPECTION/SURVEILLANCE/TESTS TO MONITOR CONTINUED SATISFACTORY PERFORMANCE OF THE S.G.

TMI-1 STEAM GENERATORS

GENERAL INTERGRANULAR STRESS CORROSION CRACKING



MUST EXPLAIN - TIMING OF CRACKING
- MATERIAL FAILURE MODE, I.E., INTERGRANULAR
- CONTAMINANT SOURCE FORM
- AXIAL/RADIAL CRACK DISTRIBUTION

4/7/82

TMI-1 OTSG TUBE MAKING PROCESS

- o ALL TUBES MANUFACTURED BY PATCO
  - NO FORMAL PATCO RECORDS AVAILABLE
  - GPUN/B&W REPS VISIT TO PATCO (1982)
  - MPR TRIP REPORT TO PATCO (1968)

- o BASE MATERIAL SUPPLIED BY B&W TUBULAR PRODUCTS

- o GENERAL PROCESS
  - BASE MATERIAL - ROUND HOLLOW BARS ~2" OD, ~0.088" WALL
  - ONE COLD DRAW THRU ROCKER TYPE REDUCER DIE TO ~1¼" OD, ~0.080" WALL
  - FOUR COLD DRAWS OVER FLOATING MANDRELS THRU A DIE TO ~0.625" OD, ~0.034" WALL
  - TUBES CLEANED, ANNEALED IN HYDROGEN ENVIRONMENT AT 1650°F ± 25°F
  - TUBES STRAIGHTENED AND CENTERLESS GROUND - MINIMUM WALL IS 0.034"

- o OTHER DATA
  - EXTREME CARE TO PREVENT CONTAMINANT CONTACT WITH TUBE
  - NDE TESTS INCLUDED UT, PT, EC, HYDRO, METAL COMPARATOR CHECK
  - INTERMEDIATE CLEANING, ANNEALING AFTER EACH DRAWING OPERATION

4/7/82

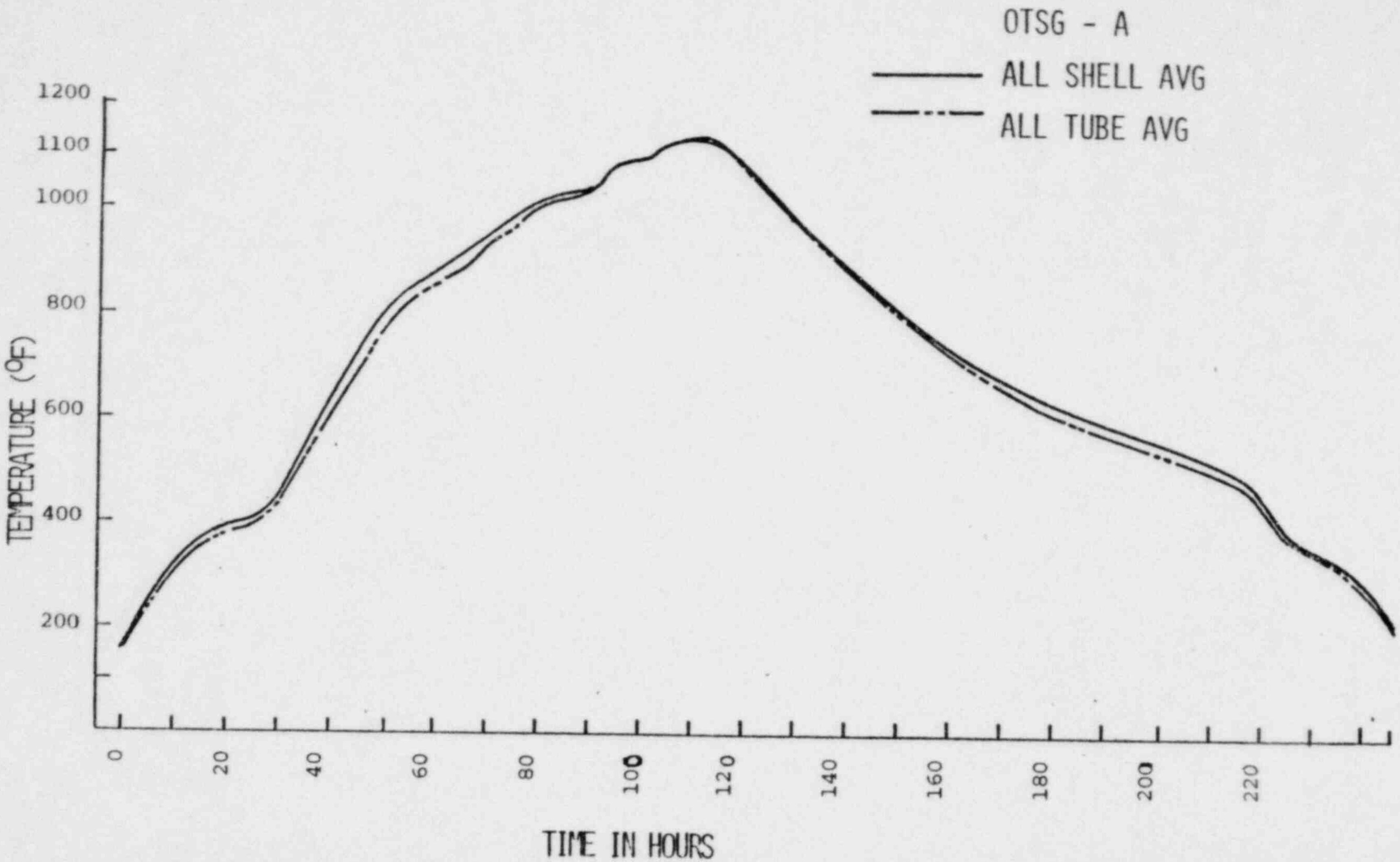
OTSG POST WELD HEAT TREATMENT

- o BASIC CYCLE TO PERFORM ASME CODE HT
  - HEAT TO 1100-1150°F
  - HOLD FOR WELDS, 1 HR. PER 1 INCH OF THICKNESS
  - FURNACE COOL TO BELOW 600°F
  - MAX. HEATING/COOLING RATES $\sim$100°F/HR
  - WELD THICKNESSES, 9½ IN. AND 7 IN.

- o ACTUAL RATES
  - HEATING $<20^\circ$F/HR FOR $T>600^\circ$F
  - COOLING $<15^\circ$F/HR FOR $T>600^\circ$F

- o FURNACE
  - 85′ x 18′ x 18′
  - ELECTRIC HEATING ELEMENTS - CAR FLOOR, ROOF, EACH WALL
  - ARGON GAS CIRCULATED

FINAL FULL VESSEL PWHT IN OTSG ELECTRIC FURNACE IN BARBERTON



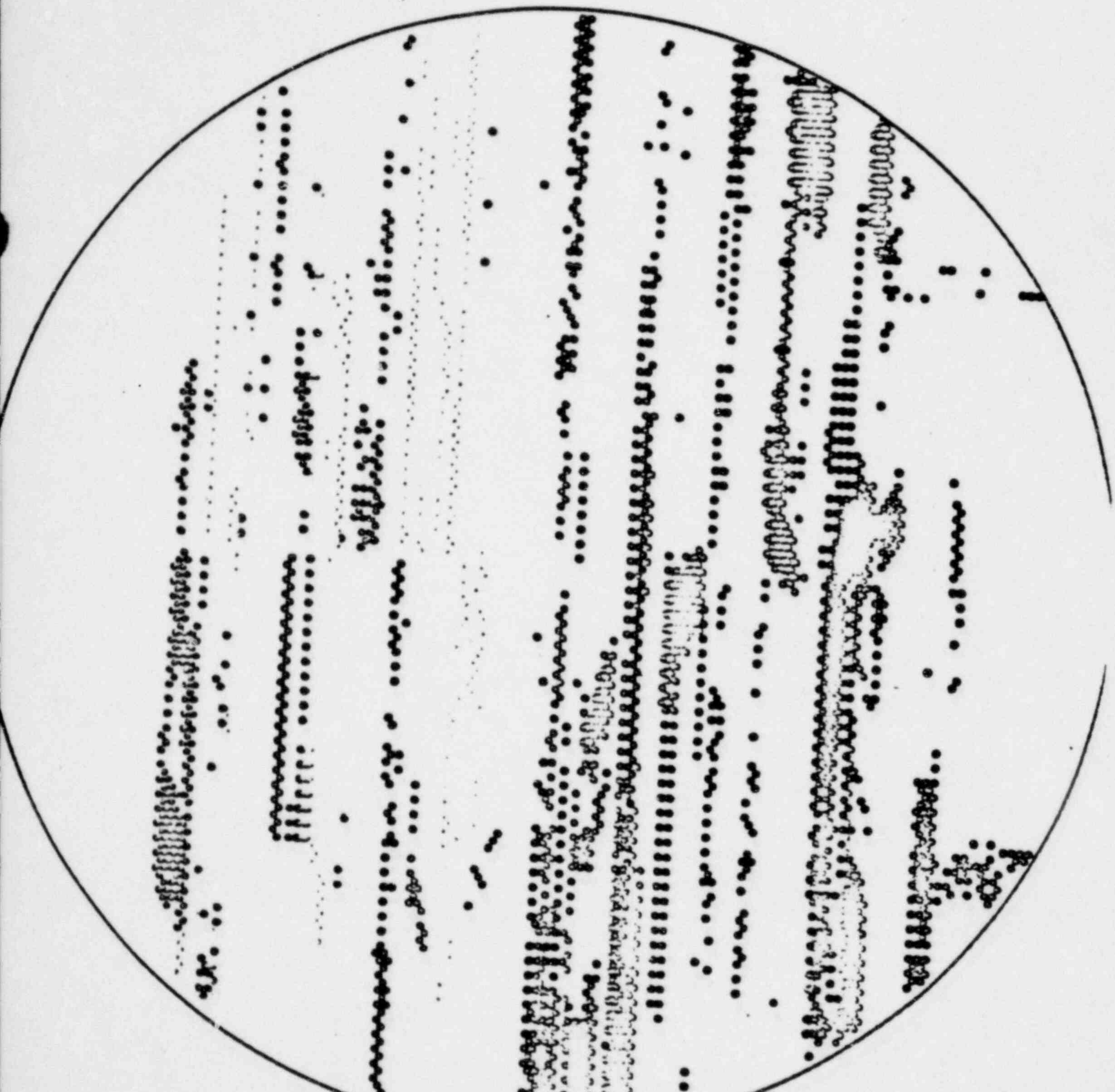
4/7/82

OTSG POST WELD HEAT TREATMENT

OTSG	HEATUP TIME 200°F TO 1100°F	TIME AT G.T. 1100°F/850°F	COOLDOWN TIME 1100°F TO 200°F
A	~100 HRS	~18 HRS/~87 HRS	~128 HRS
B	~68 HRS	~13 HRS/~73 HRS	~129 HRS

4/7/82

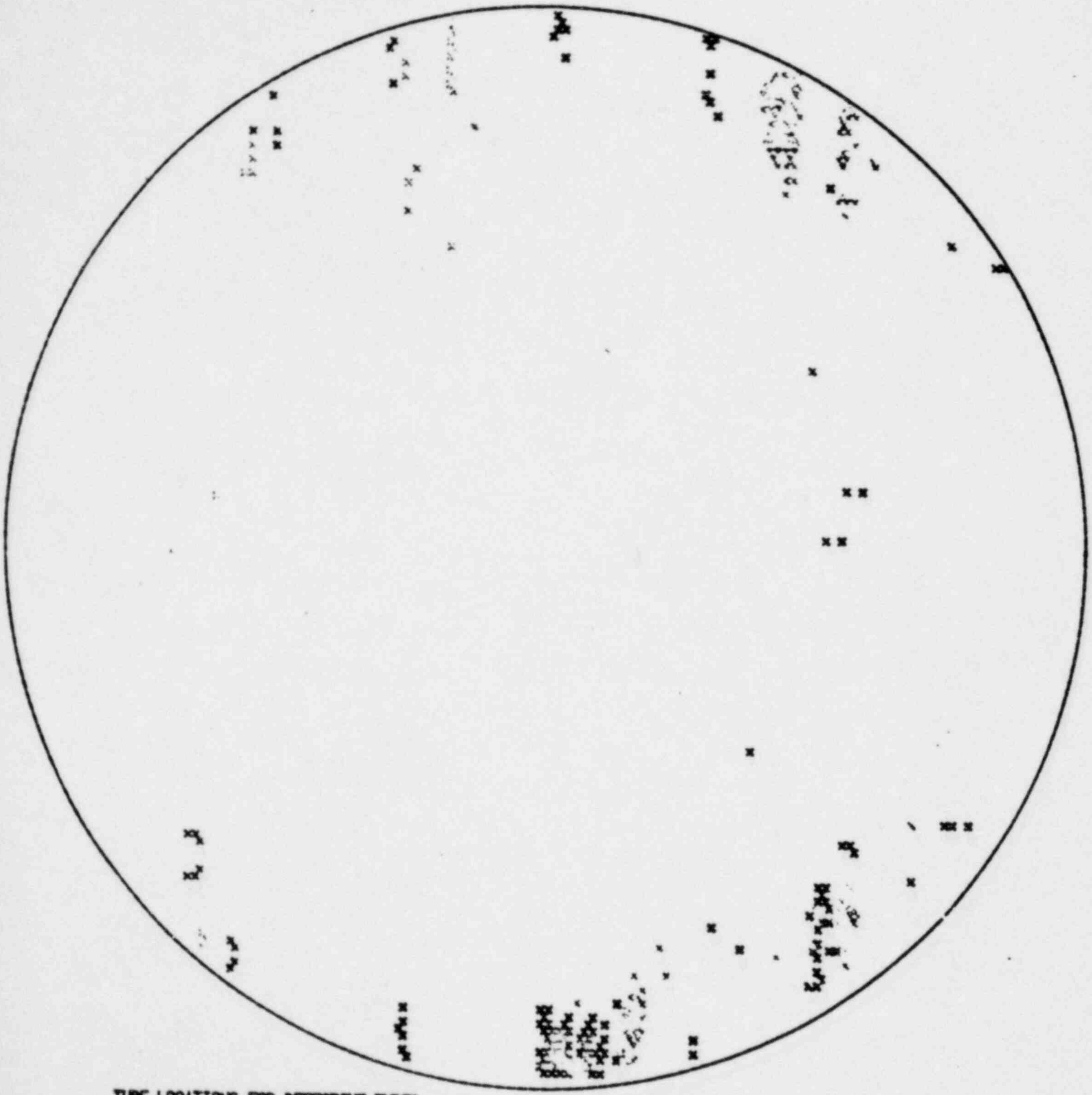
THREE MILE ISLAND NUCLEAR
GENERATING STATION
UNIT 1 STEAM GENERATOR A



TUBE LOCATIONS OF ALL TUBES
FOR TUBE HEAT NUMBERS
M2320
M2346

M2582

THREE MILE ISLAND NUCLEAR
GENERATING STATION
UNIT 1 STEAM GENERATOR A

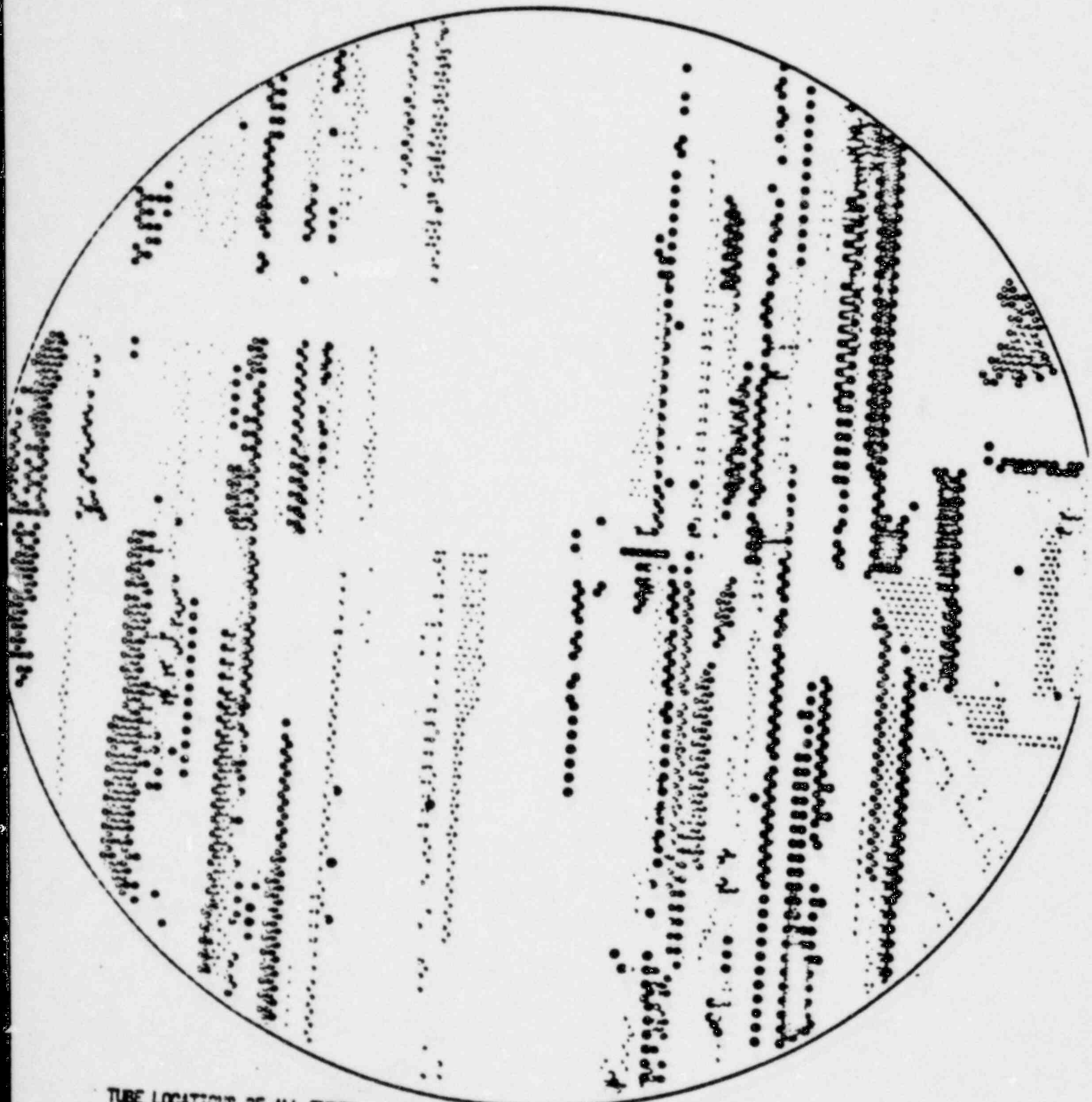


TUBE LOCATIONS FOR DEFECTIVE TUBES
FOR TUBE HEAT NUMBERS
M2328

M2582

4/7/82

THREE MILE ISLAND NUCLEAR
GENERATING STATION
UNIT 1 STEAM GENERATOR B

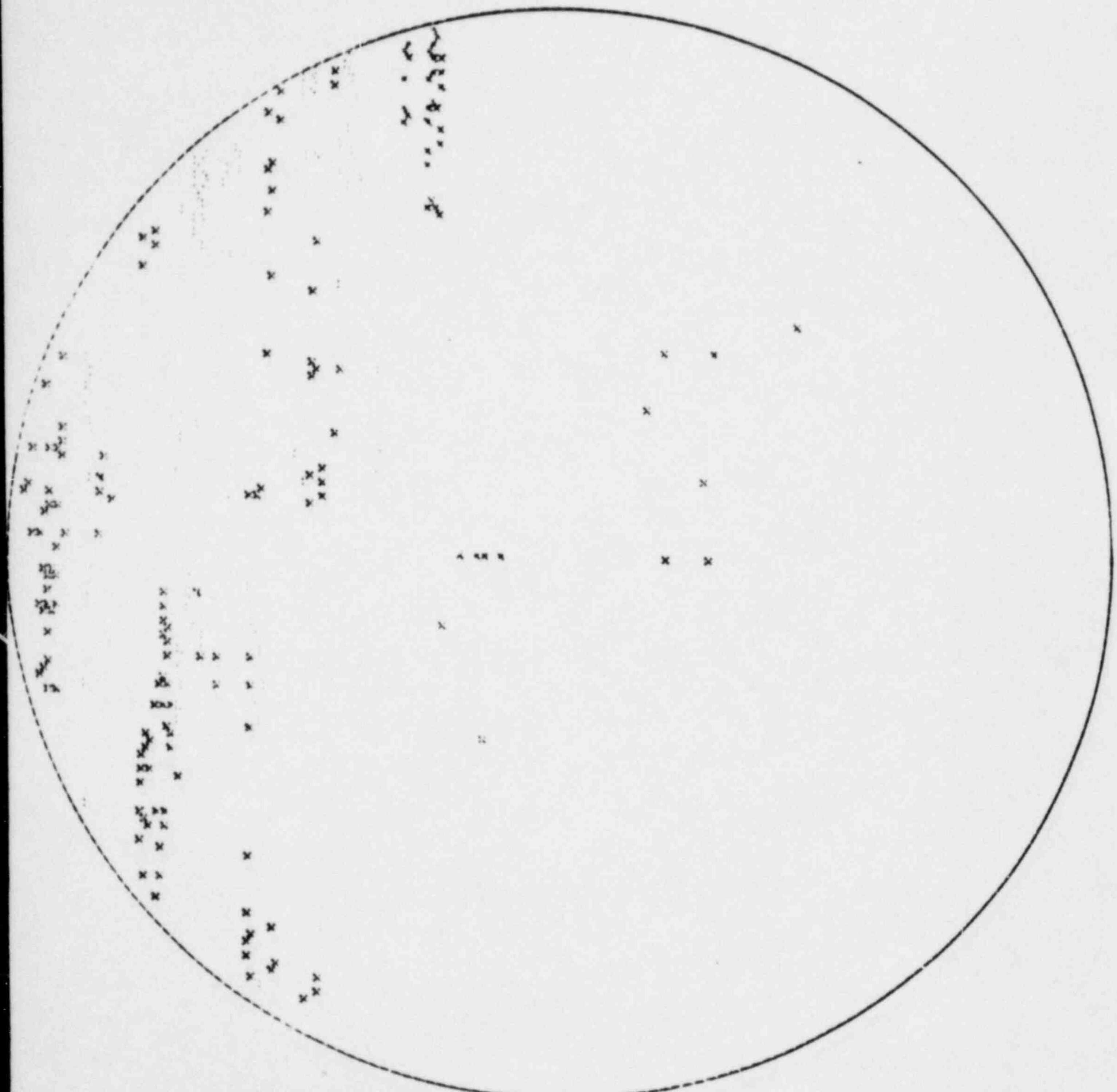


TUBE LOCATIONS OF ALL TUBES
FOR TUBE HEAT NUMBERS
M2708

M2889

11/7/82

THREE MILE ISLAND NUCLEAR
GENERATING STATION
UNIT 1 STEAM GENERATOR B



TUBE LOCATIONS FOR DEFECTIVE TUBES
FOR TUBE HEAT NUMBERS
M2708

M2808

HEAT VS DEFECT CORRELATION

APRIL 2, 1982

SUMMARY

- o TUBE FAILURES ARE ASSOCIATED WITH SPECIFIC LOCATIONS IN THE GENERATOR NOT HEAT RELATIONSHIPS.
- o THE DEFECT PATTERNS IN THE TWO GENERATORS ARE DIFFERENT AND THIS WILL NEED TO BE EXPLAINED BY A PARAMETER OTHER THAN HEAT NUMBER.
- o HEATS OF MATERIAL EXIST WHICH HAVE HIGH DEFECT FREQUENCIES IN BAD AREAS AND THE SAME HEATS WILL HAVE LOW DEFECT FREQUENCIES IN GOOD AREAS.

4/7/82

STRESS RELIEF DATA REVIEW, "B" OTSG UTS

- o CENTER OF BUNDLE IN UTS IS 10 - 20^{o}F HIGHER IN TEMPERATURE DURING HEAT UP AND HOLD BUT IS 5 - 10^{o}F LOWER IN TEMPERATURE DURING COOLDOWN

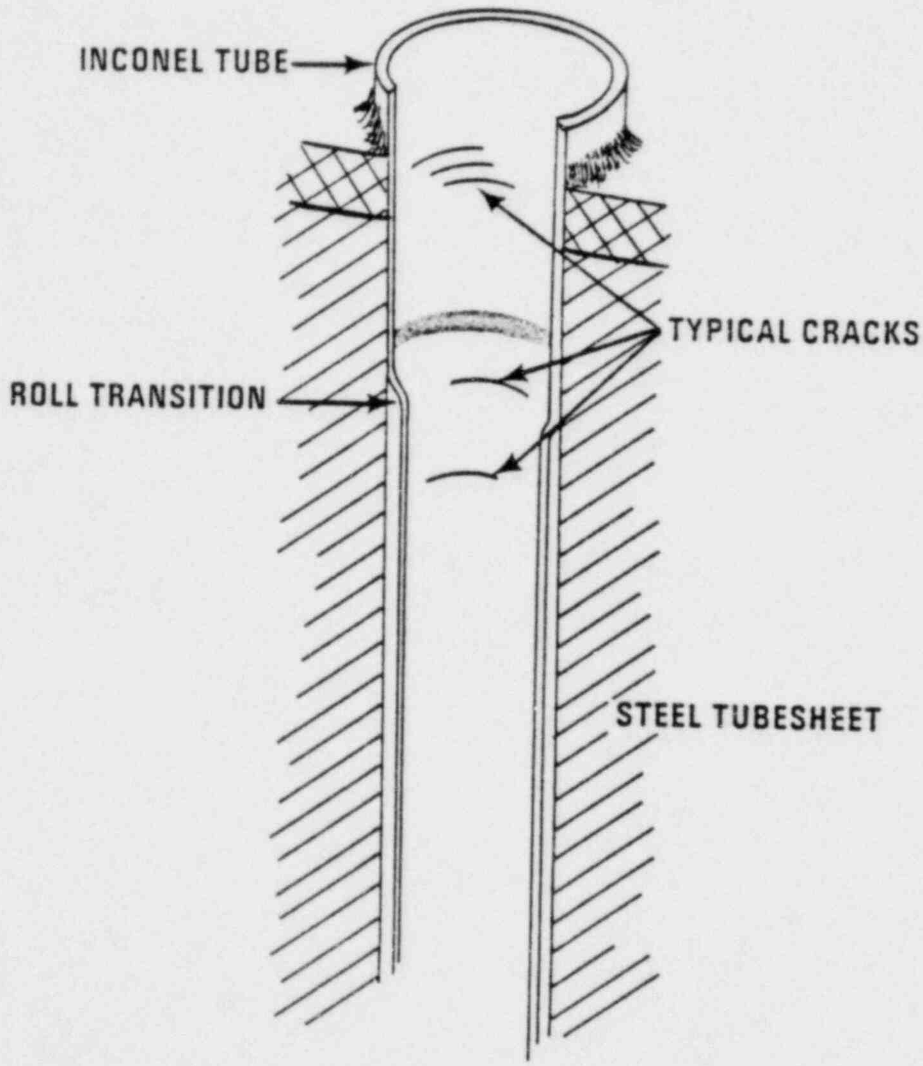
- o NO SIGNIFICANT TEMPERATURE VARIATIONS EXIST AROUND BUNDLE PERIPHERY

- o NO SIGNIFICANT DIFFERENCES IN TIMES AT TEMPERATURE EXIST AROUND THE PERIPHERY

- o MAXIMUM TUBE TEMPERATURE ACHIEVED DURING STRESS RELIEF WAS 1140^{o}F

- o OVERALL THE TIMES AT TEMPERATURES INDICATE THE TUBES WERE HELD IN TEMPERATURE REGIONS WHERE SENSITIZATION WOULD BE EXPECTED TO BE SEVERE

TMI-1 Steam Generator
Typical Cracks



CRACK CHARACTERISTICS: CIRCUMFERENTIAL
NOT FULL ARC
GENERALLY VERY TIGHT
INSIDE INITIATED

4/7/82

TUBE ANALYSIS SUMMARY

ANALYSIS	NO. OF TUBES	NO. OF SAMPLES
METALLOGRAPHIC	8	38
BEND TEST	15	19
SCANNING ELECTRON MICROSCOPY (SEM)	15	15
ENERGY DISPURSIVE X-RAY ANALYSIS (EDAX)	15	15
AUGER ELECTRON SPECTROSCOPY (AES)	5	7
ELECTRON SPECTROSCOPY FOR CHEMICAL ANLYSIS (ESCA)	5	6
SCANNING TRANSMISSION ELECTRON MICROSCOPY (STEM)	5	7
ELECTROCHEMICAL POTENTIOKENITIC REACTIVATION (EPR)	4	5
HUEY TEST	1	3
SECONDARY ION MASS SPECTROSCOPY (SIMS)	2	3
ELECTRON DIFFRACTION	1	1
TRANSMISSION ELECTRON MICROSCOPY (TEM)	2	2
TENSILE TEST	3	3
RESIDUAL STRESS	1	1
SODIUM AZIDE SPOT TEST	3	5

4/7/82

GPUN Failure Analysis Investigation Team

Babcock & Wilcox Lynchburg Research Center	Tube failure analysis
Babcock & Wilcox Alliance Research Center	Corrosion testing
Battelle Columbus Laboratories	Tube failure analysis
Oak Ridge National Laboratories Metals & Ceramics Division	Corrosion testing
Massachusetts Institute of Technology	Tube analysis for sensitization
*	
Electric Power Research Institute	Failure analysis review

ADDITIONAL LABORATORIES

Westinghouse Electric Research & Development Laboratories	Independent tube failure analysis

* Deleted Ohio State since there is no contract between GPUN and Ohio State.

TUBE UTILIZATION SUMMARY
APRIL 2, 1982

TOTAL TUBING AVAILABLE:		37.8 FT.
TOTAL TUBING EXAMINED:		13.1 FT.
TUBING ALLOCATED FOR TESTING:		
o WESTINGHOUSE FAILURE ANALYSIS	-	.35 FT.
o TENSILE TEST OF DEFECT TUBE	-	.67 FT.
o CORROSION TESTING	-	7.10 FT.
o ROLLING/SLEEVING TESTS	-	7.10 FT.
	TOTAL	15.22 FT.
UNALLOCATED TUBING		
o PIECES W/O DEFECTS		7.9 FT.
o PIECES WITH DEFECTS		1.6 FT.
	TOTAL	9.5 FT.

4/7/82

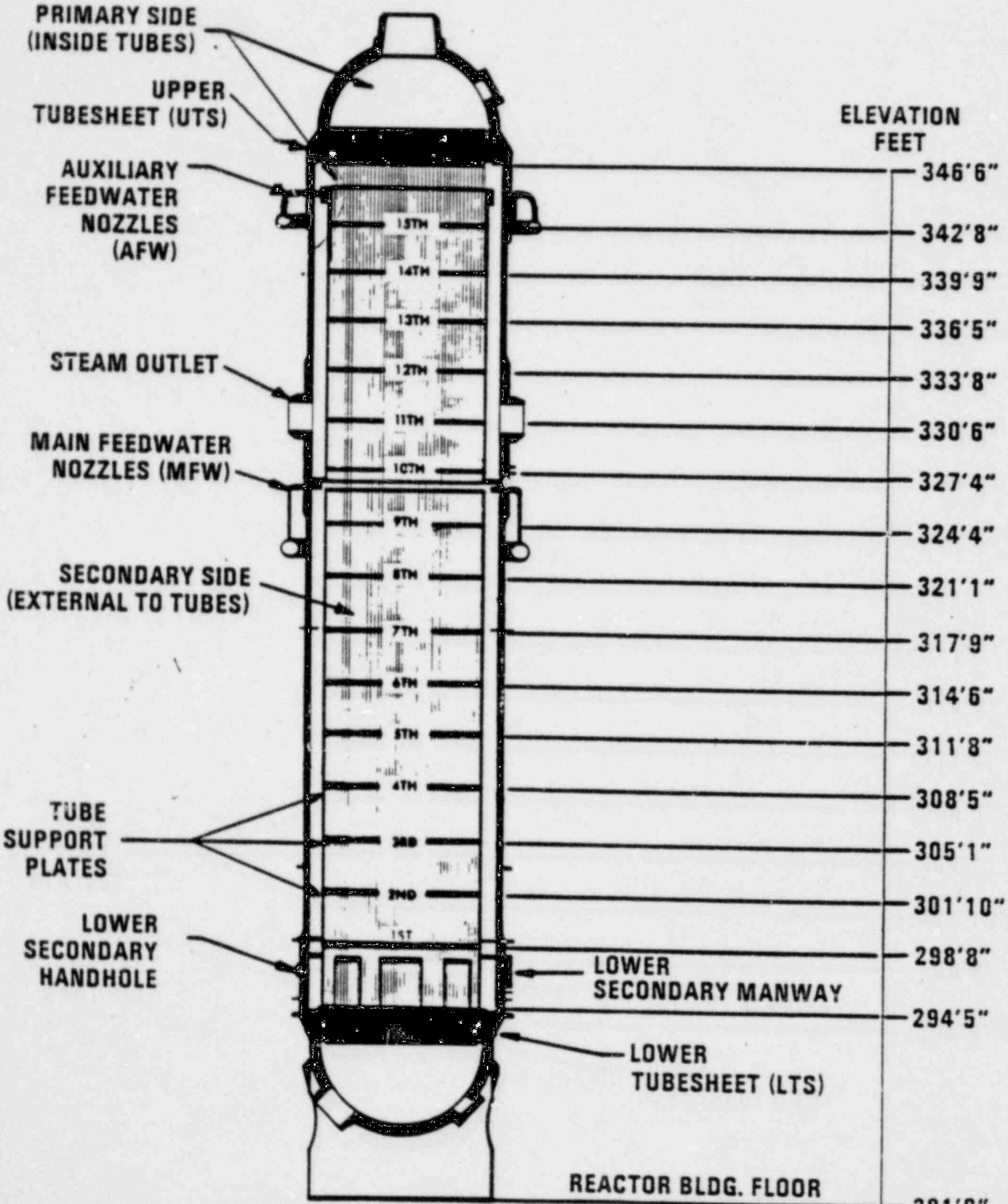
SUMMARY OF FAILURE ANALYSIS

APRIL 7, 1982

- ALL CRACKS ARE STRESS ASSISTED INTERGRANULAR CORROSION WITH INITIATION ON THE ID SURFACE
- EDDY CURRENT EXAMINATION HAS BEEN A RELIABLE INDICATOR OF CRACK LOCATION
- INCIPIENT CRACKS HAVE NOT BEEN DETECTED IN CLEAN SECTIONS (NO E.C. INDICATIONS) OF TUBING BY VISUAL AND DESTRUCTIVE EXAMINATION
- CARBON IN THE FORM OF A HYDROCARBON APPEARS AS THE MAJOR CONTAMINANT ON FRACTURE SURFACES. SULFUR AND CHLORINE ARE PRESENT AS SECONDARY CONTAMINANTS
- RESIDUAL STRESS MEASUREMENTS IN ROLL AND ROLL TRANSITION REGION SHOW NO STRESS PEAKS BUT RATHER A UNIFORM DISTRIBUTION
- CHROMIUM LEVELS IN THE GRAIN BOUNDARIES VARY FROM 8 WT. % TO 20 WT. %
- THE INCONEL MICROSTRUCTURE APPEARS TYPICAL FOR STEAM GENERATOR TUBING WITH DISCRETE CHROMIUM CARBIDE PARTICLES IN THE GRAIN BOUNDARIES
- SMALL AREAS OF INTERGRANULAR CORROSION SEVERAL GRAINS DEEP HAVE BEEN OBSERVED ON THE ID AND OD SURFACES AT RANDOM LOCATIONS
- NO RELATIONSHIP HAS BEEN ESTABLISHED BETWEEN MATERIAL HEATS AND DEFECTIVE TUBING
- MECHANICAL TESTING OF UNCRACKED TUBES SHOW THAT THE MATERIAL EXCEEDS MINIMUM SPECIFICATION REQUIREMENTS

F I G U R E II - 4

OTSG Longitudinal Section Elevations (Typ.)



4/7/82

TMI-1 STEAM GENERATORS - STRESSES

- STRESS MAXIMUM IN ROLL TRANSITION AREA, ~34 KSI±
- STRESS MAXIMUM ON OUTER EDGE OF GENERATORS
- STRESS EXPECTED TO BE QUITE VARIABLE IN ROLL TRANSITION AREA
- MAXIMUM STRESSES ARE AXIAL
- STRESSES SAME IN UPPER AND LOWER TUBESHEET
- STRESSES MAXIMUM DURING COOLDOWN, COLD
- AT OPERATING TEMPERATURE - HOOP STRESS >AXIAL STRESS

4/7/82

ATTACHMENT #2

PLANT COOLDOWN
6/22/78

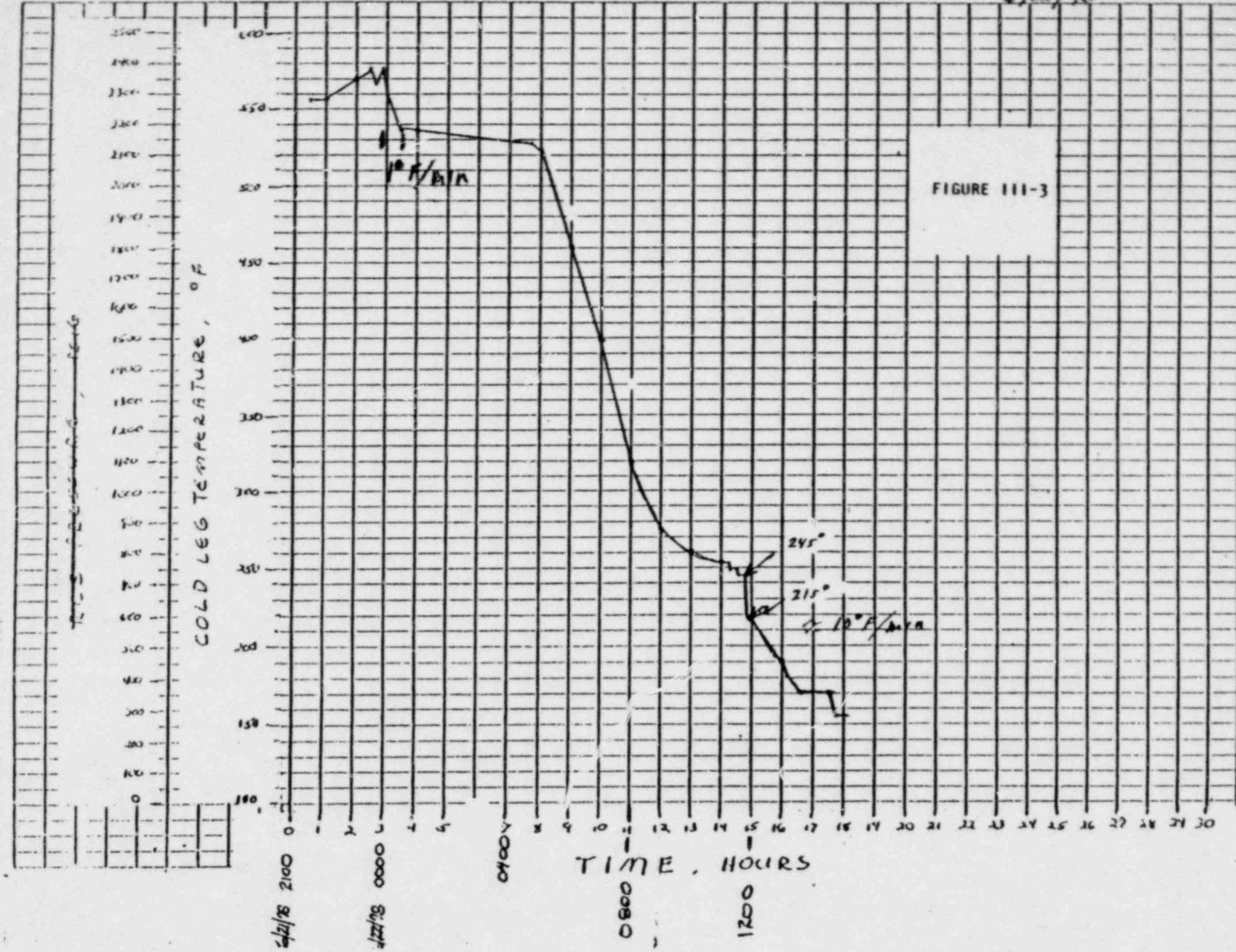


FIGURE III-3

4/7/82

OPERATING HISTORY OBSERVATIONS-1

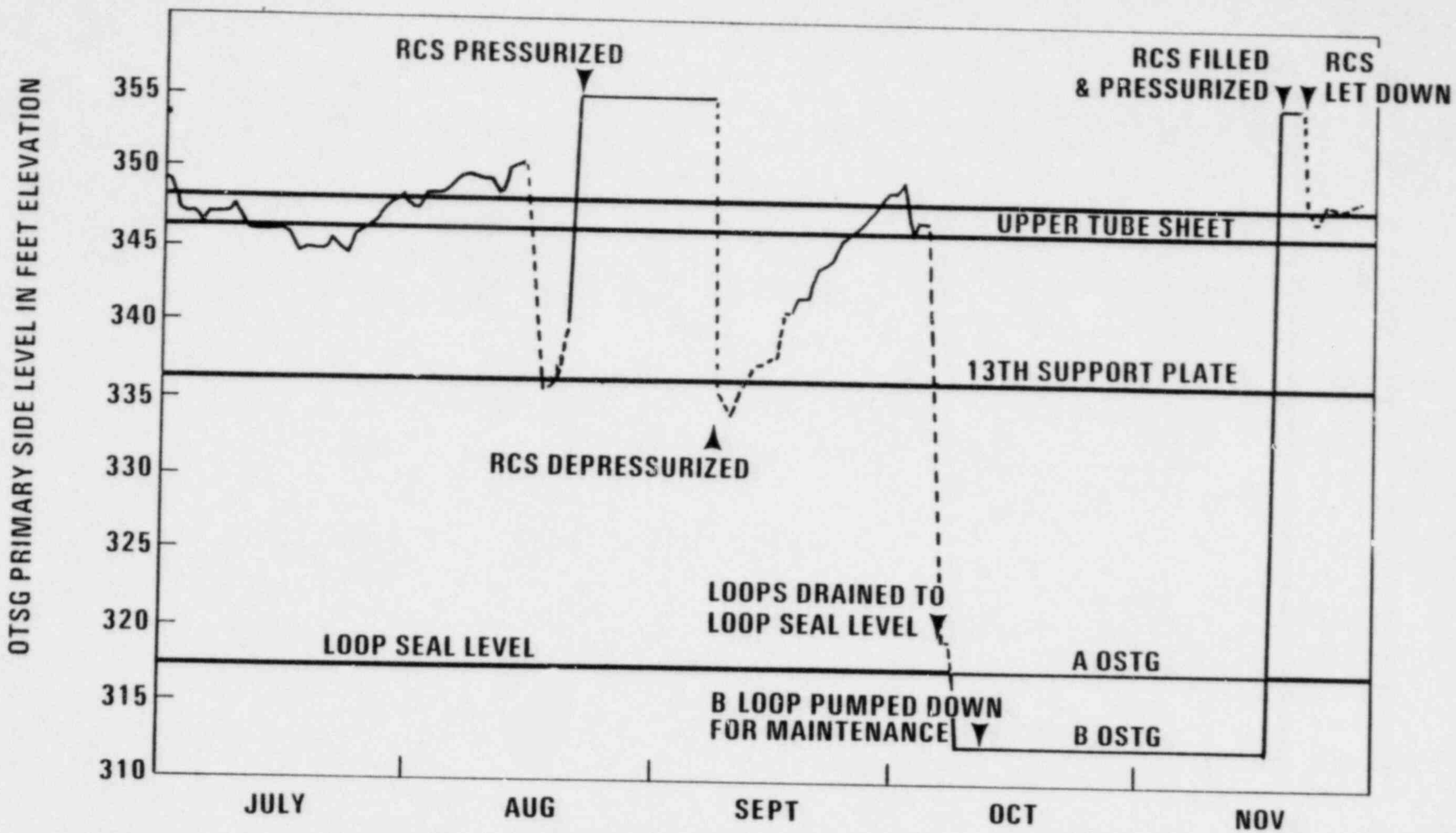
- o LOWER END GENERATOR ALWAYS SUBMERGED (WETTED), UPPER END ALTERNATE WET AND DRY WITH AIR (OXYGEN) INTERFACE

- o WATER LEVEL IN THE PRIMARY SIDE OF OTSG WAS IN UTS FOR BETWEEN 31 AND 243 DAYS

- o SOME DIFFERENCES IN AMOUNT OF FLOW SINCE FEB '79
  - TOTAL PUMP HOURS OTSG - A = 681 HRS
  - TOTAL PUMP HOURS OTSG - B = 393 HRS
  - BACK FLOW IN OTSG - B FOR 10 HRS

  DURING SEPTEMBER '81 COOLDOWN

OTSG Level
July 1981 — November 1981



DOTTED LINE MEANS ESTIMATED LEVEL
VENTING ARRANGEMENT CAUSES UP TO 27" ERROR

OPERATING HISTORY OBSERVATIONS - 2

- o POTENTIAL SULFUR SOURCES PRESENT
  - SOME OIL INTRODUCED INTO RCS IN MAR '79
  - SULFURIC ACID ADDED TO RCS IN OCT '79
  - SODIUM THIOSULFATE ADDED TO RCS AT VARIOUS TIMES OVER LIFE OF PLANT
- o SODIUM THIOSULFATE THOUGHT TO BE PRIMARY CONTRIBUTOR
  - ACCUMULATED IN BUILDING SPRAY PIPING - 1979-81 - AS A RESULT OF VALVE LEAKAGE
  - JUN, AUG, SEP '81 OPERATION OF SPRAY PUMPS ADDED SOLUTION TO BWST
  - INJECTION INTO RCS OCCURRED DURING SEP '81 COOLDOWN

TMI-1 STEAM GENERATORS - SULFUR SAMPLES

SYSTEM	DATE	SULFATE (PPB)	TOTAL SULFUR (PPB AS SO_4)
REACTOR COOLANT DECAY HEAT	7/31/79	-	1,500
	8/02/79	-	<600
	11/01/79	-	<660
	12/04/81	-	730
	1/18/82	-	400
	2/04/82	-	100
BORATED WATER STORAGE TANK	1/20/82	-	<100
REACTOR BUILDING SPRAY PUMP OUTLET	1/20/82	-	15,000
	3/17/82	2,876	-
	3/20/82	764	-
INTERCONNECT BETWEEN BUILDING SPRAY AND DECAY HEAT	1/20/82	-	176,000
	3/17/82	2,465	-
	3/20/82	752	-
SPENT FUEL POOL	1/18/82	-	400
	3/17/82	246	-
	3/20/82	149	-

4/7/82

CORROSION TESTING
SUMMARY

- o TEST WITH ACTUAL DECAY HEAT COOLANT ON SENSITIZED INCONEL AND STAINLESS STEEL BENT STRIPS. RESULTS - NO CRACKING IN TWO WEEKS

- o TEST WITH ACTUAL DECAY HEAT COOLANT ON AN ACTUAL TUBE SAMPLE REMOVED WITH AN INCIPIENT DEFECT. RESULT - NO CRACK GROWTH

- o 34 ELECTROCHEMICAL CORROSION TESTS WITH VARIOUS CONTAMINATED PRIMARY COOLANT ENVIRONMENTS AND VARIOUS SPECIMENS

  - BORIC ACID (PPM) - 13,000, 5,000
  - THIOSULFATE (PPM) _ 100, 10, 1, 0
  - HYDRAZINE (PPM) - 200
  - MATERIALS - M5442, M2320 - ACTUAL
M2320 - ARCHIVE
  - TEMPERATURE - 550, 100^{o}F
  - ATMOSPHERE - AIR, HYDROGEN

PRELIMINARY CORROSION TEST RESULTS

- CORROSION TESTS IN ACTUAL PRIMARY COOLANT INDICATE IT IS CURRENTLY INNOCUOUS
- REDUCED SULFUR SPECIES CAN REPRODUCE THE TYPE OF CRACKING OBSERVED IN STEAM GENERATOR TUBES
- THE DEGREE OF SENSITIZATION (I.E., PRIOR HEAT TREATMENT) IS A KEY PARAMETER IN DEFINING THE MATERIALS SUSCEPTABILITY TO IGSCC
- THE PROPENSITY FOR A SULFUR CONTAMINATED PRIMARY COOLANT ENVIRONMENT TO INITIATE CRACKING VARIES INVERSELY WITH THE BORIC ACID AND LITHIUM HYDROXIDE CONCENTRATIONS
- CRACK INITIATION APPEARS TO BE THE RATE CONTROLLING PAR-AMATER
- CRACK GROWTH RATE IS VERY RAPID ON THE ORDER OF 1MM/DAY
- CRACKING APPEARS TO BE A LOW TEMPERATURE OCCURRENCE
- CRACKING TENDENCY IS REDUCED BY RAISING THE pH

4/7/82

KEY ELEMENTS IN EXP REVIEW

- o IGSCC OF I-600 OBSERVED AT 575°F IN SULPHATE CONTAINING WATER; UNLIKELY TO OCCUR UNDER PWR PRIMARY SYSTEN REDUCING ENIVRONMENT
  - NOT ASSOCIATED WITH DEGREE OF SENSITIZATION
- o IGSCC OF I-600 OBSERVED AT 75- 225°F IN SULPHUR OXYANION (E.G. THIOSULPHATES) CONTAINING WATER; MORE LIKELY TO OCCUR IN PWR PRIMARY SYSTEM
  - CRACKING IS RAPID
  - SUSCEPTIBILITY DEPENDS ON SENSITIZATION, pH, TEMPERATURE, AND ELECTROCHEMICAL POTENTIAL
- o PLANT AND MODEL BOILER EXPERIENCE IS ENTIRELY RELATED TO SECONDARY SIDE PROBLEMS
- o NONE OF PRIMARY SIDE INDUSTRY EXPERIENCE IGSCC OF I-600 ATTRIBUTED TO ATTACK BY BULPHUR SPECIES

4/7/82

HIGHLIGHTS OF STRESS ANALYSIS

- TUBING AXIAL TENSILE STRESSES LARGEST DURING COOLDOWN; MAY APPROACH YIELD STRESS
- SIGNIFICANT AXIAL TENSILE STRESSES ALSO EXIST DURING COLD SHUTDOWN
- LOCALLY HI AXIAL TENSILE STRESSES POSSIBLE IN SEAL WELD HAZ AND NEAR ROLL TRANSITION
- AXIAL STRESSES GENERALLY LARGER AT PERIPHERY THAN IN CENTER OF TUBE BUNDLE

4/7/82

SUSCEPTIBLE MATERIAL MICROSTRUCTURE

- FAB HISTORY SHOWS TUBING TO BE MILL ANNEALED PLUS STRESS RELIEVED
  - ∴ HIGHLY SENSITIZED
- MET EXAMS CONFIRM EXPECTED MICROSTRUCTURE
- CORROSION TESTS SHOW PULLED TUBES SUSCEPTIBLE TO CRACKING IN THIOSULFATE/BORIC ACID SOLUTIONS

4/7/82

AGGRESSIVE ENVIRONMENT

- SO_4^{--} AND $S_2O_3^{--}$ CONTAMINATION PROBABLY PRESENT

- CHANGES IN S-SPECIES EXPECTED DURING HOT FUNCTIONAL -- DIFFICULT TO PREDICT SPECIES PRESENT AFTERWARDS

4/7/82

AQUEOUS SULFUR SPECIES

Formula	Structure	Sulfur Oxidation Number*	Name
H_2S or $S^{=}$		-2	sulfide
H_2S_2, $S_2^{=}$	$\left[S-S\right]^{=}$	-1	
H_2S_3, $S_3^{=}$	$\left[S-S-S\right]^{=}$	-2/3	polysulfides
H_2S_x,	$\left[S-S--\right]^{=}$	-2/x	
S S_8 rings		0	sulfur
$S_2O_3^{=}$	$\left[O-\overset{O}{\underset{O}{S}}-S\right]^{=}$	+2	thiosulfate
$S_4O_6^{=}$	$\left[O-\overset{O}{\underset{O}{S}}-S-S-\overset{O}{\underset{O}{S}}-O\right]^{=}$	+2.5	tetrathionate
$SO_3^{=}$	$\left[O-\overset{O}{S}-O\right]^{=}$	+4	sulfite (sulfurous acid)
SO_2		+4	sulfur dioxide
$S_2O_6^{=}$	$\left[O-\overset{O}{\underset{O}{S}}-\overset{O}{\underset{O}{S}}-O\right]^{=}$	+5	dithionate
$SO_4^{=}$	$\left[O-\overset{O}{\underset{O}{S}}-O\right]^{=}$	+6	sulfate

---

* Oxidation number is the formal electrical charge assigned to the sulfur on the assumption that H is +1 and O is -2 in these compounds.

4/7/82

PROPOSED FAILURE SCENARIO

1. SO_4^{--} AND $S_2O_3^{--}$ (POSSIBLY OTHERS) ADDED DURING LAYUP
2. REDUCED S-SPECIES FORMED DURING HOT FUNCTIONAL
3. WATER LEVEL DROPPED. HIGH CONCENTRATION OF AGGRESSIVE S-SPECIES FORMED IN DRY-OUT REGION
4. CRACKING OCCURS IN DRY-OUT ZONE
5. CRACKING TERMINATES DUE TO REDUCTION OF CONCENTRATION
6. CRACKING IS DISCOVERED WHEN OTSGs ARE PRESSURIZED

4/7/82

FEATURES COVERED BY SCENARIO

- TIME OF CRACKING
- MODE OF CRACKING
- AXIAL DISTRIBUTION OF CRACKING
- RADIAL DISTRIBUTION OF CRACKING (OTSG-A)
- CORROSION TEST RESULTS

4/7/82

IMPLICATIONS OF SCENARIO

- SULPHUR REDUCTION NECESSARY TO PREVENT RECURRENCE
  - -- OXIDATION TO SOLUBLE FORM
  - -- REMOVAL VIA DEMINERALIZER

- ATTACH OF OTHER PRIMARY SYSTEM COMPONENTS, IF ANY, MOST PROBABLE IN VICINITY OF WATER LINE LOCATION FOLLOWING HOT FUNCTIONAL
  - -- INCONEL X-750
  - -- SENSITIZED TYPE 304 STAINLESS STEEL

4/7/82

REACTOR COOLANT SYSTEM REVIEW

OBJECTIVES

- o REVIEW REACTOR COOLANT SYSTEM COMPONENTS FOR CONTINUED SAFE OPERATION
- o CLASSIFY ITEMS FOR MATERIAL CONDITION, ENVIRONMENT EXPOSURE AND APPLIED STRESS
- o SELECT CANDIDATES FOR INSPECTION AND TESTING THAT ARE REPRESENTATIVE OF WORST CONDITIONS
- o MINIMIZE EXPOSURES
- o EMPLOY STANDARD ACCEPTANCE TESTING BUT SELECT SUSCEPTIBLE MATERIALS FOR DESTRUCTIVE METALLURGICAL EXAMINATION

4/7/82

REACTOR COOLANT SYSTEM REVIEW

PROGRAM PLAN

- CLASSIFY ALL MATERIAL TYPES USING FABRICATION HISTORY AND LOCATION IN RCS
- IDENTIFY ASSOCIATED STRESS LEVELS AND SAFETY CONSIDERATIONS FOR APPLICATION
- EVALUATE RCS MATERIAL CORROSION SUSCEPTIBILITY
- IDENTIFY POTENTIAL PROBLEM AREAS FOR RECERTIFICATION INSPECTION AND TEST
- DEVELOP INSPECTION PLAN
- PERFORM INSPECTIONS AND EVALUATE RESULTS PROVIDING AS NECESSARY ANY CONTINGENCY TESTING
- DOCUMENT ACCEPTABILITY OF PRIMARY SYSTEM FOR SAFE RESTART

4/7/82

REACTOR COOLANT SYSTEM REVIEW

INSPECTION PLAN

O DESTRUCTIVE METALLURGICAL ANALYSIS	- INCONEL 600 - INCONEL X-750 - SS 304 - INCONEL 718
O EDDY CURRENT	- I-600 NOZZLE TO SS FLANGE - I-600 (NOT AXIALLY LOADED)
O ULTRASONIC TESTING	- SS 304 - BOLTS - INCONEL X-750 - SS 304 TUBING - I-600 SAFE ENDS - SS 304 WELDMENTS
O RADIOGRAPH TESTING	- I-600 SAFE ENDS - SS 304 WELDMENTS
O PENETRANT INSPECTION	- SS 304 CLAD - I-600 CLAD
O FUNCTIONAL TESTS	- IN-CORE DETECTORS - VENT VALVES

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INSPECTION PLAN (CONT'D)

O VISUAL EXAMINATION

- CORE COMPONENTS
- PLENUM
- HOLD DOWN SPRINGS
- END FITTINGS
- FUEL RODS
- SPACER ASSEMBLIES
- CONTROL RODS
- SHELLS AND BOLTING RINGS
- BAFFLE PLATE REGION
- LOWER BOLTING RINGS
- LOWER VESSEL HEAD

O OVERALL

- INSPECT OR TEST APPROXIMATELY 1000 ITEMS

Repair Criteria

(1) The maximum allowable primary-to-secondary leakage rate for normal operation shall be as low as reasonably achievable and allow plant operation within the radioactive effluent limits of the technical specifications.

Repair Criteria

(2) Repaired tube shall sustain, with adequate margins, the design basis loads

Loads	Generic 177FA	TMI-1
LOCA	+ 2641 lb	+ 2641 lb
MSLB	+ 3140 lb	+ 3140 lb (being reanalyzed)
FWLB	- 620 lb	- 620 lb
Normal cooldown:	+ 1107 lb	+ 1107 lb

+ = tension

\- = compression

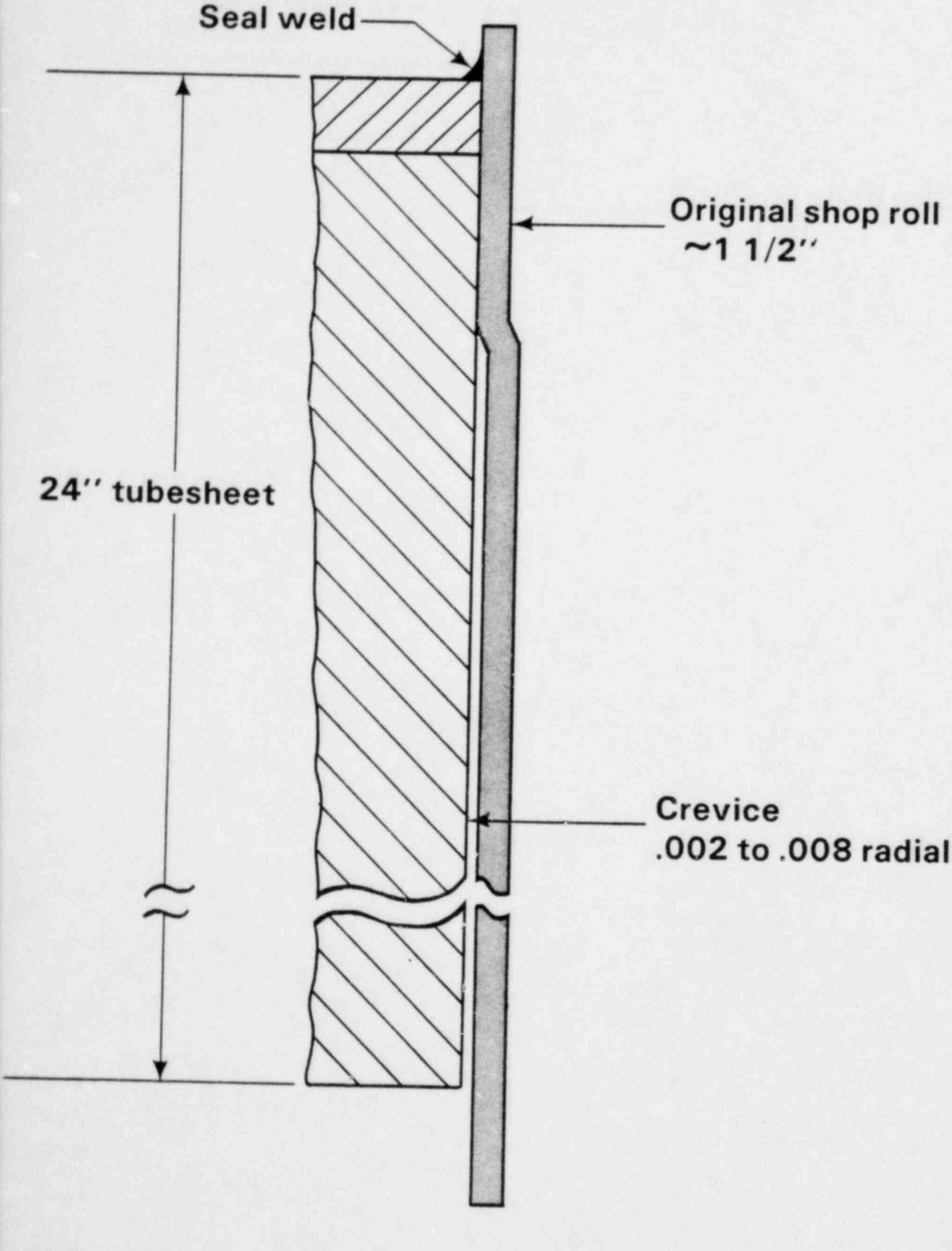
Repair Criteria

(3) The effects of both repaired and plugged tubes on the thermal and hydraulic performance of the plant and on the structural and vibrational adequacy of the steam generator shall be evaluated and shall be within the acceptance criteria for both normal operating and design basis accident conditions as specified in the licensing basis documents.

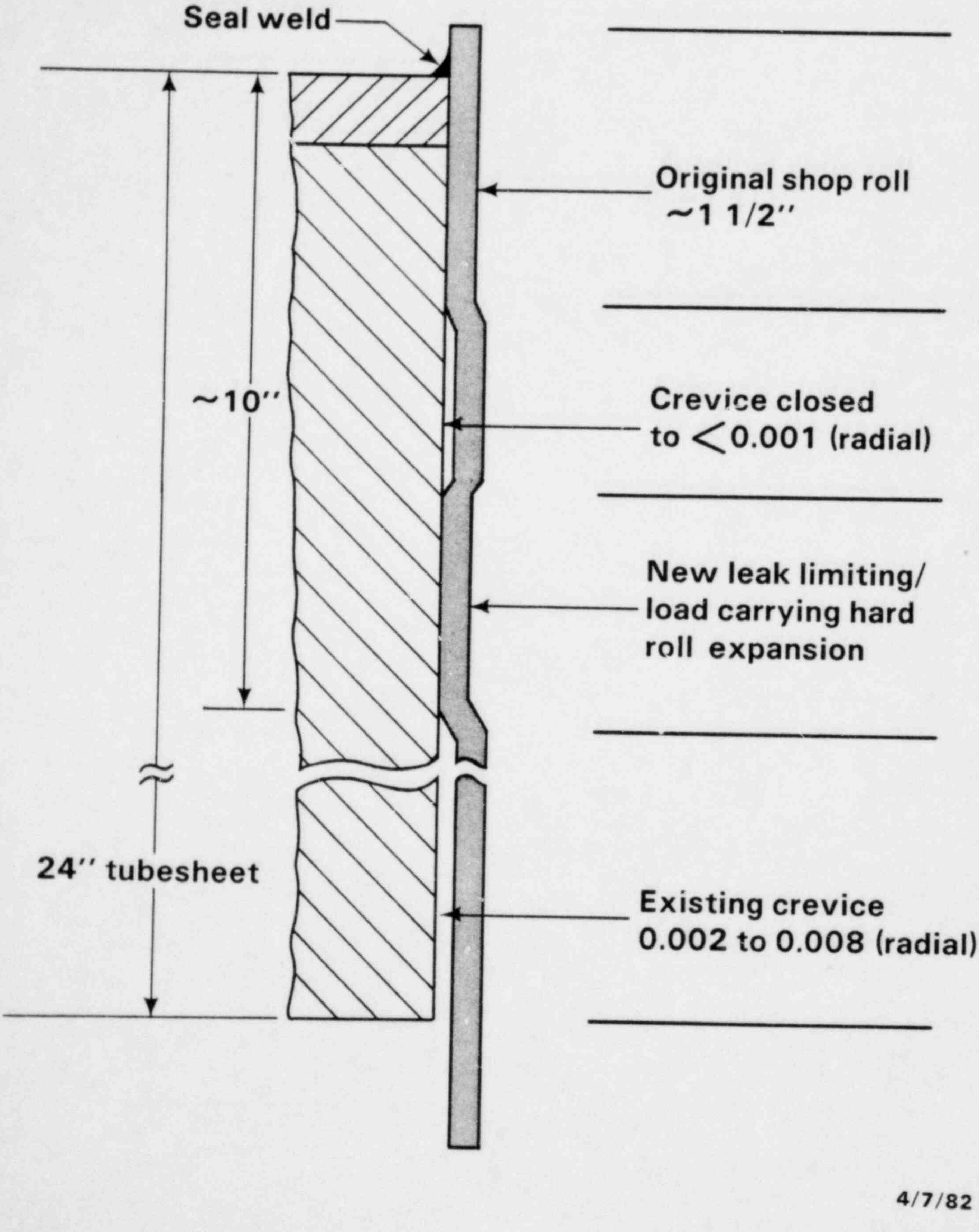
Preliminary Repair Process Qualification Criteria

- Result in a process capable of providing a leak-tight joint
- Produce a joint capable of carrying the design basis loads
- Maintain the tensile preload in the free standing portion of the tubes within allowable limits
- Result in minimal tensile stresses
- Produce an expansion capable of being non-destructively examined
- Be adaptable tc remotely operated tooling
- Permit future sleeving

Original Configuration of OTSG Tubes



Repair Configuration of OTSG Tube



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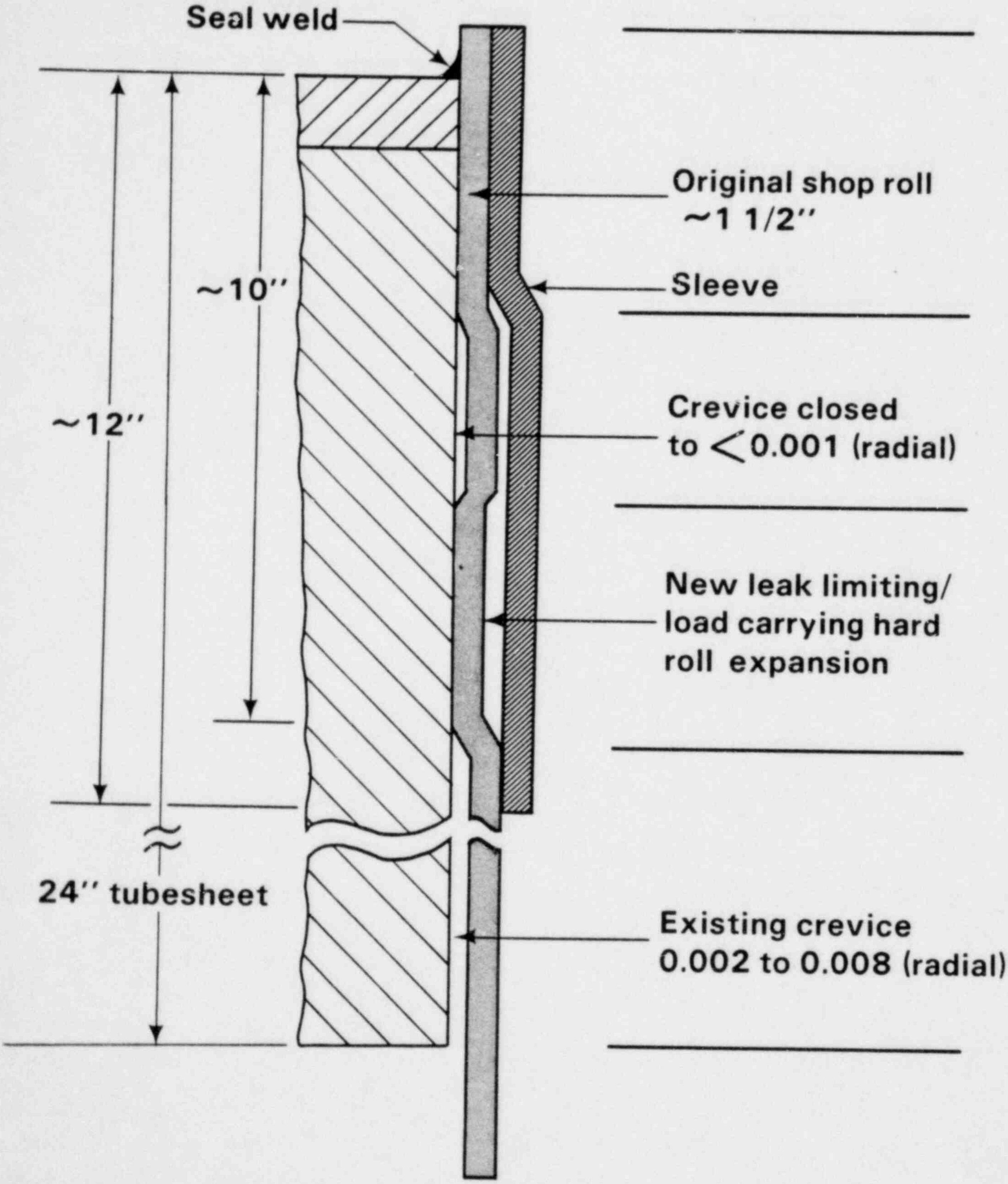
Sleeved Configuration of OTSG Tube



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Scope of Process Qualification Design Variables

- Hone ID surface in area to be expanded
  - —avoids inclusion of contaminants
- Depth of roll approximately 10″ max
  - —allows later sleeving
  - —leaves approximately 500 tubes or less to be plugged or otherwise repaired
- Crevice closure by low-torque roll, explosives, or hydraulics
  - —proven techniques
  - —minimizes residual stresses
  - —inspectable
- Mechanical roll with 4-10% wall thinning
  - —carries axial load
  - —retains preload
  - —leak tight
  - —proven technique
- Sound tube material below repair

Facts on Mechanical (Roller) Expansion

- Residual stresses in roll transition zones can be reduced by increasing the end radius on the rollers

- The optimum roller geometry has been determined to be:

  - 1 1/2″ long rollers with largest standard available diameter

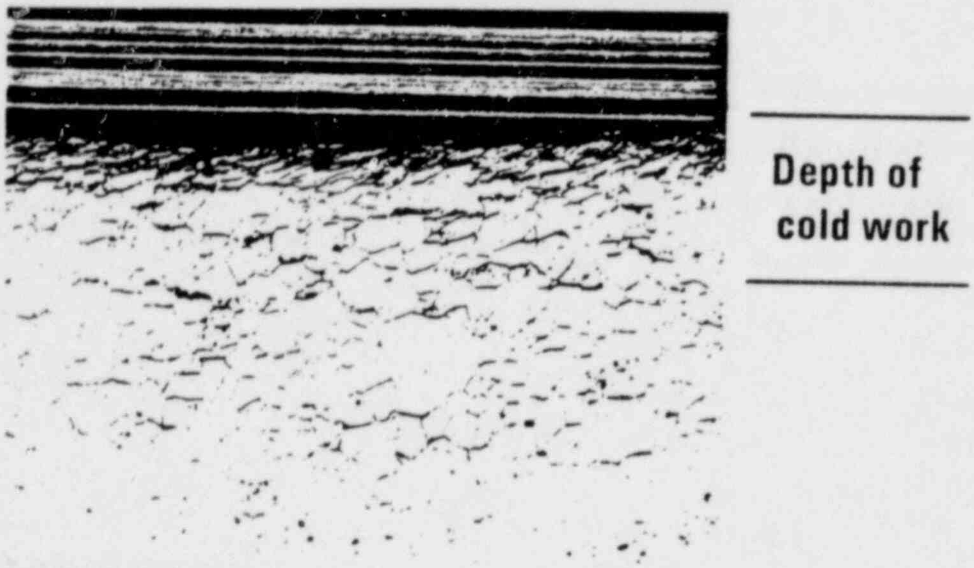
  - 1″ effective length of roller with 2 1/2″ end radii

- Axial residual stresses are greater than those produced by hydraulic or explosive expansions

- Residual hoop stresses are less than those produced by hydraulic or explosive expansions

- Roll expansions produce thinning of the tube wall in the expanded area; industry standards (based on allowable metal strain) are 4 to 10%

- Roll expansions produce a net elongation of the tube due to the extrusion of the tube walls



10% Chromic 100X

— Figure 8 —

Rolled portion of a tube showing the amount of cold work present in the overlap area using standard rolls.

10% Chromic 100X

— Figure 9 —

Depth of cold work produced by increasing the leading radius to 2-1/2 inches.

Experience with Mechanical Joints

- Industry SG experience with mechanical joints

- Doel-2	Tube/tubesheet rolls (repair)	About 100 rolled in 1980 & 1981
- Point Beach-1	Rolled sleeves (repair)	About 12 sleeves in 1982
- San Onofre-1	Rolled sleeves (repair)	About 7000 sleeves in 1981
- Obrigheim	Tube/tubesheet rolls (original)	12 years service
- Palisades	Hydraulically expanded sleeves (repair)	Installed commencing 1976

- Other industry experience with repair hard roll

- Big Rock Point	RTR vessel/CRD housing	4" tube in 1979. No leakage
- Oyster Creek	RTR vessel/in core flux monitor tube	2" tube in 1975. No leakage
- Gargliano	- Ditto -	2" tube in 1966. No leakage

- Standard heat exchanger manufacturing process

Preliminary Tube Expansion Process Comparison

	Mechanical Roll	Hydraulic Expansion	Explosive Expansion
Residual stresses			
ID	Greater	Base	Equal
OD	Less	Base	Equal
Effect on tube Preload *	Decrease	Increase	Little change
Load carrying capability	Greater than	Base	Greater than
Leak tightness	Greater than	Base	Greater than

Supporting data based on:

- B&W Canada and B&W USA R&D and production work accomplished on both once-through and u-tube steam generators

* Inadvertently left off of slide used on 4-7.

Points to Be Addressed by Qualification Program

Adequacy of repair process

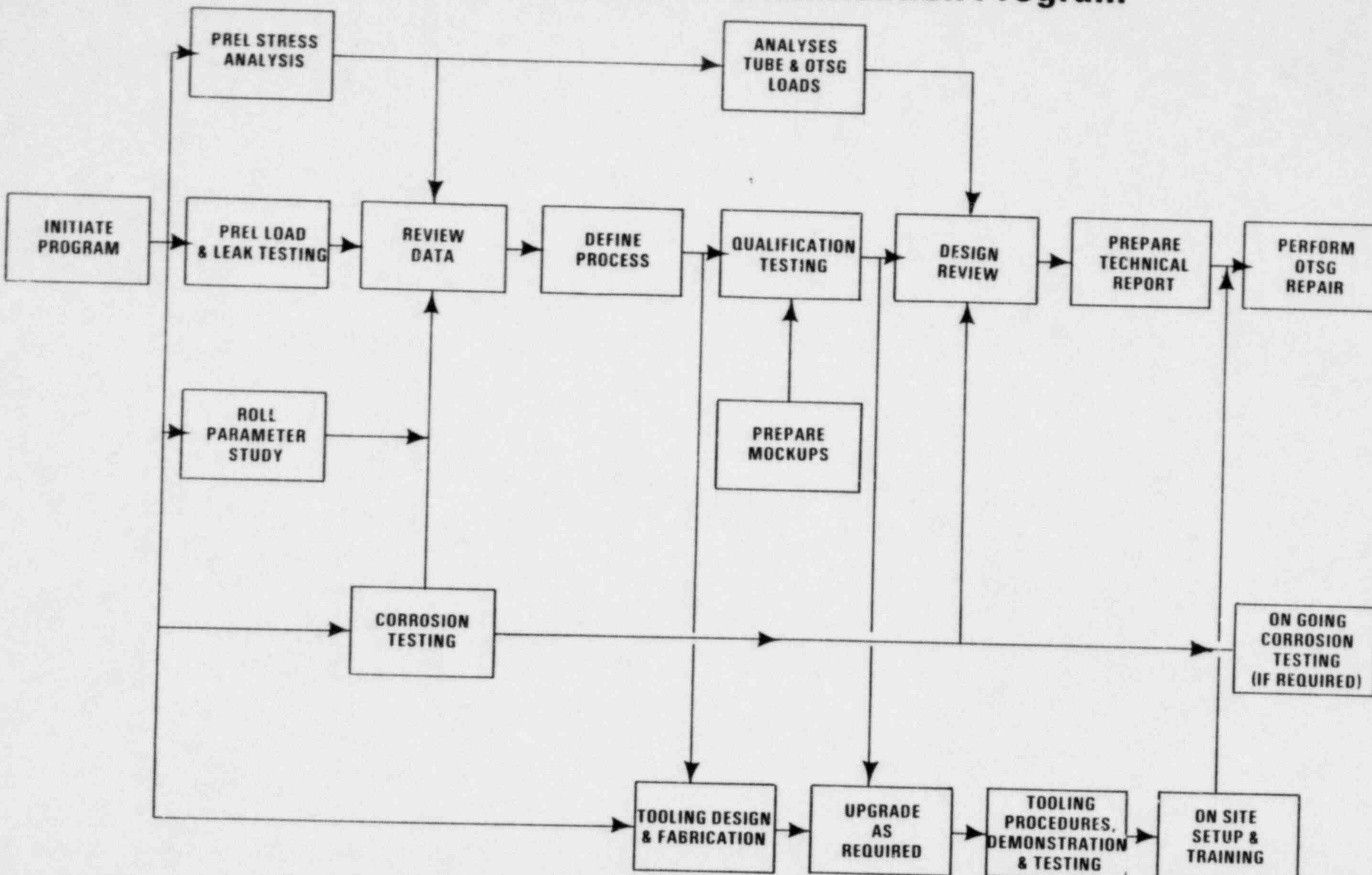
- Leak tightness following thermal cycling
- Load carrying capability following thermal cycling
- Tubesheet hole ovality
- Water or moisture in crevice
- Statistical leak tightness margin determination
- Roll torque/length vs leak tightness
- Roll torque/length vs load carrying capability
- Inspectability

Points to Be Addressed by Qualification Program

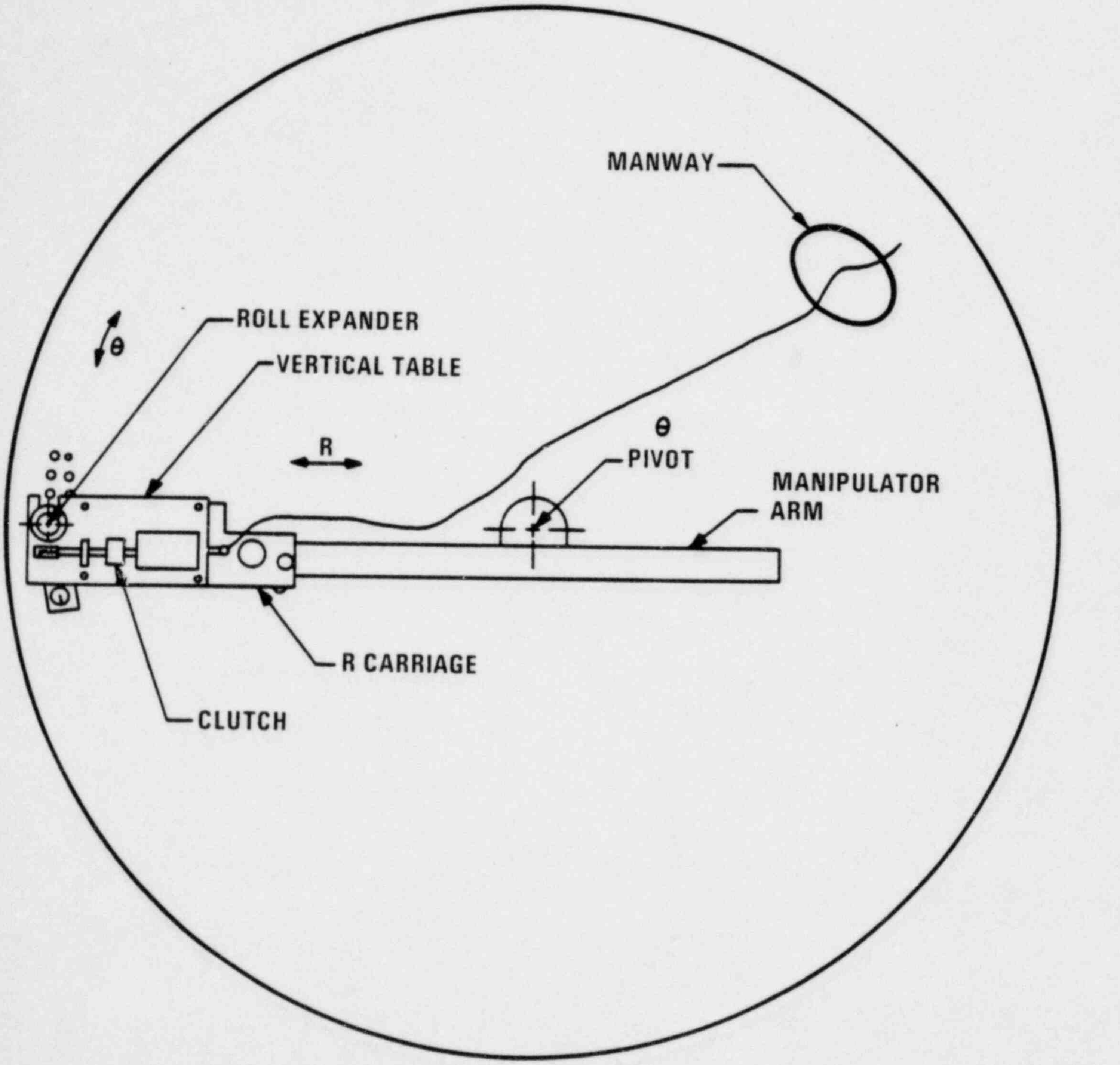
Effect of repair on total OTSG performance

- Primary water in crevice and tubesheet corrosion
- Change of tube preload
- Residual stresses in tube
- Effect of trapped contaminants
- OTSG performance with specific tubes plugged
- Confirm adequacy of existing operating and accident analyses

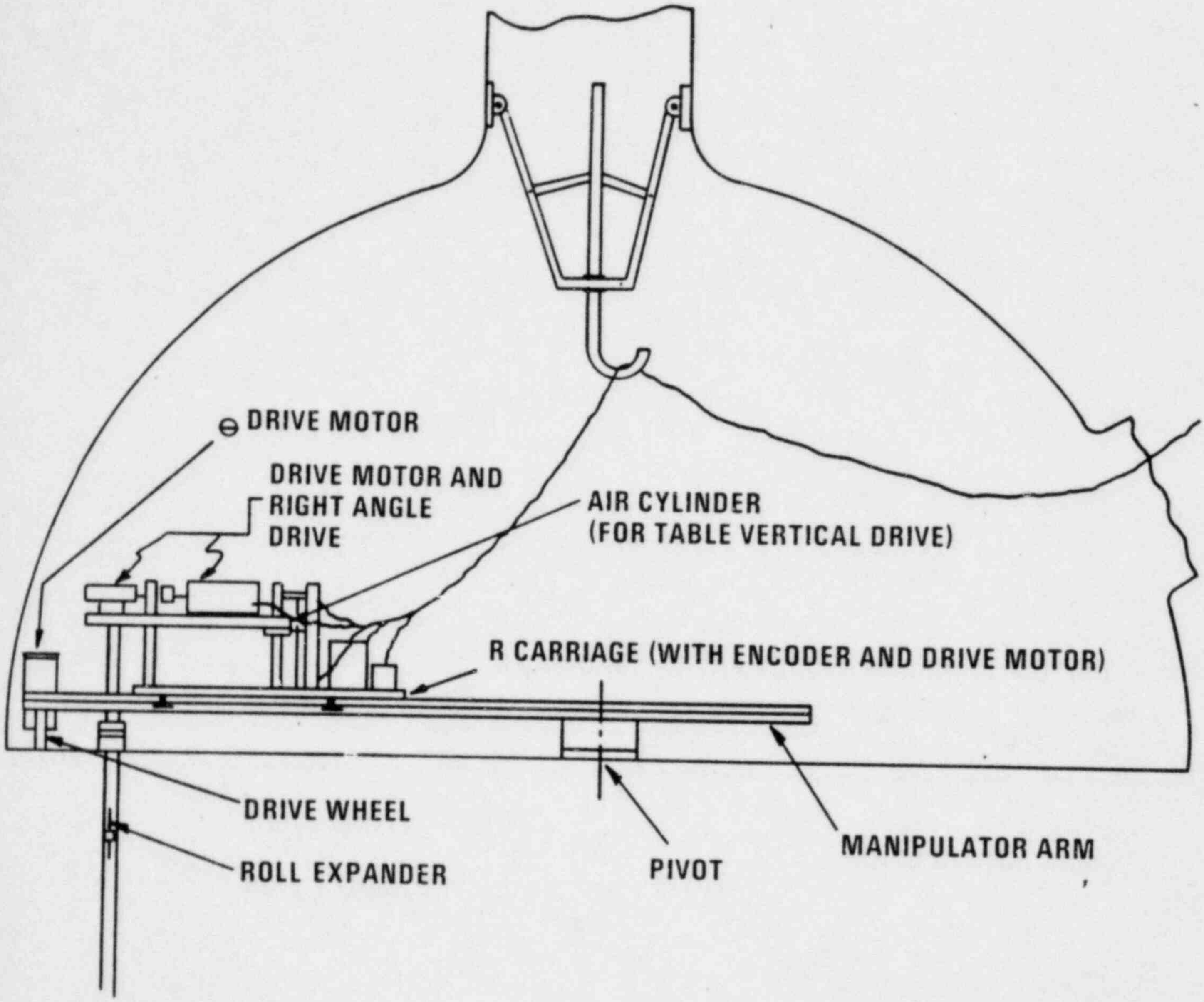
TMI-1
Tube Expansion Qualification Program



OTSG Tube Rolling
Top View-Manipulator



OTSG Tube Rolling — Elevation



Currently Planned
Process Monitoring and Inspection

Tube identification	• E/C manipulator record • Video record
Depth of rolls	• Automated insertion tool • Tool location feedback • Video
Torque	• Transducer feedback • Air pressure alarm • Periodic calibration • Frequent equipment inspections and cleaning

OTSG Repair Program Overview

I. Tube expansion/testing complete Oct. '82

II. Projected total repair exposure $<$ 500 man rem

III. 100% ECT examination in affected area

IV. Plug or repair all tubes with inside diameter ECT indications in the roll, roll transition or tubesheet crevice

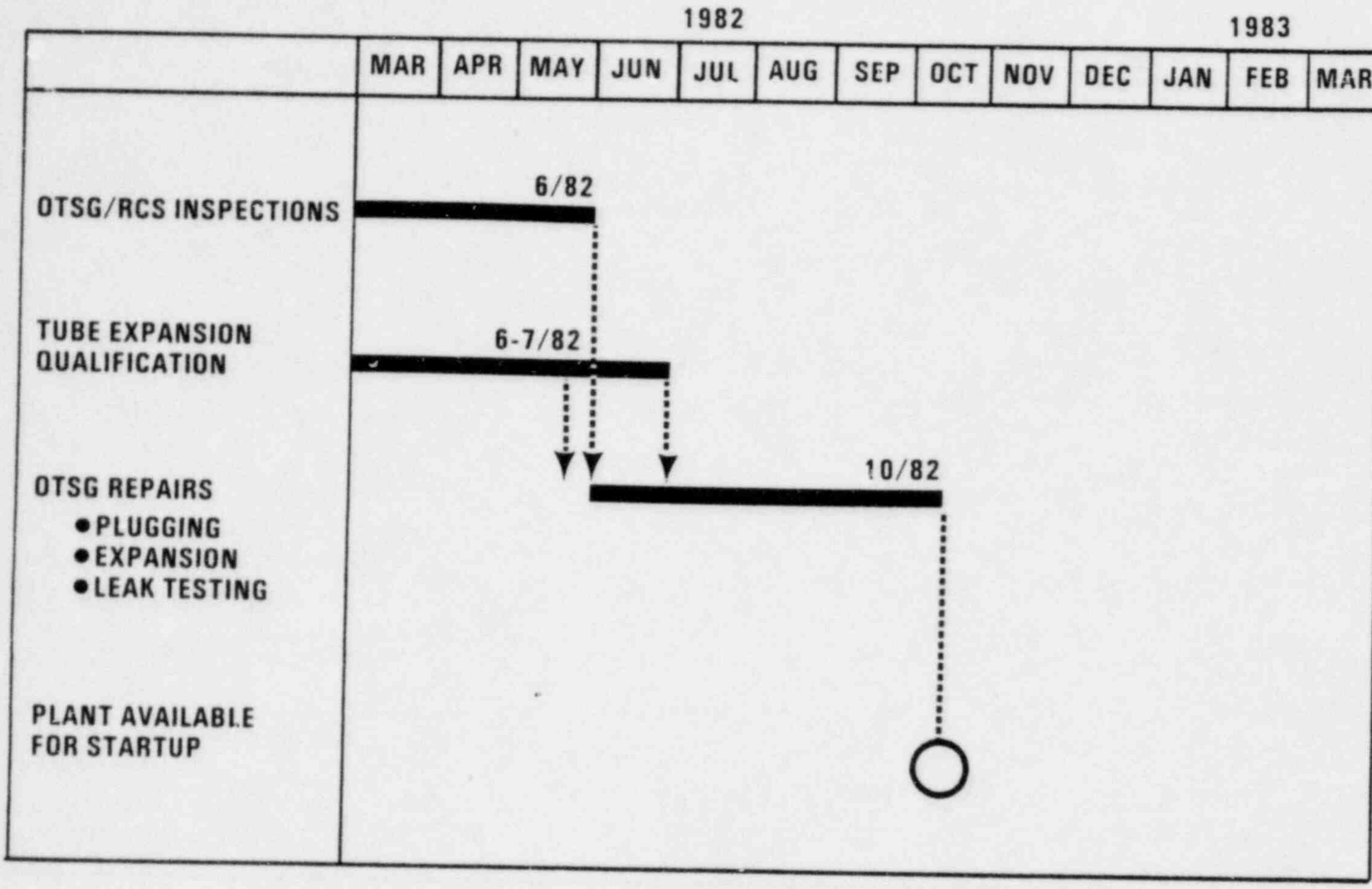
V. Plug/stabilize all tubes with inside diameter ECT indications that are not within the tubesheet

VI. RCS cleanup to reduce the amount of sulfur on surfaces

VII. Sensitive leak tests following repair

VIII. Sensitive and continuous leak rate monitoring during operation

OTSG Repair Program Plan/Schedule



Preliminary Cumulative Man Rem Exposure

I. Actual OTSG exposure to date ~70 man rem

II. Estimated additional exposure ~230 man rem

A. RCS inspection	~	60R
B. Eddy current testing	~	10R
C. Tube samples	~	10R
D. Tube plugging	~	100R
E. Tube expansion	~	50R

III. Projected total OTSG repair exposure with 200 man rem contingency < 500 man rem

OTSG Tubing Eddy Current Inspection Program

I. Objective

- Identify scope and extent of tubing damage and soundness of tubing areas accepted for service
- Repetitive inspections to detect new defects and/or defect growth

II. Scope

- 100% of affected areas
- Statistical sample below affected areas

III. Techniques

- Standard differential probe (multi-frequency system — increased sensitivity) tubing areas between lower and upper tubesheet roll transitions

  * "Gain Settings" used incorrectly on 4-7 slide.
- Absolute probe upper tubesheet roll transitions and rolled areas (4 coils — 2 orientations — 360 degrees coverage)

Standard Differential Eddy Current Technique Qualification

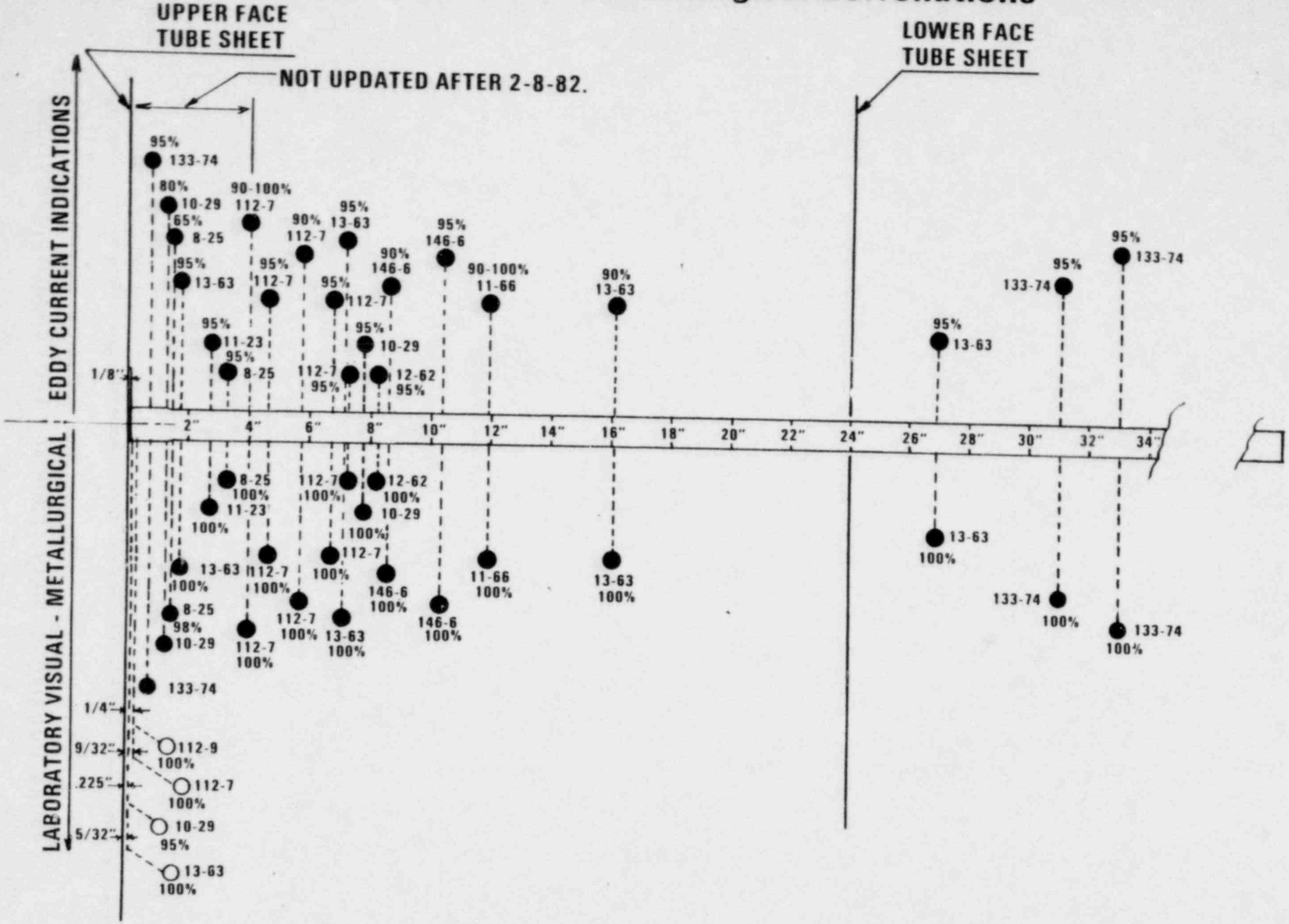
I. Metallurgical Analysis
- 100% correlation on 29 E/C defects located at or below the roll transition

II. Other destruct Testing
- Laboratory testing of ~ 13 feet of tubing verifies soundness of portions of removed tubes accepted by eddy current examination

III. Correlations among E/C designs/ techniques
- Circumferential coils vs pancake coils
- Standard differential vs absolute techniques

IV. Production E/C data interpretation
- Evaluation of all intelligible signals irrespective of amplitude to account for crack orientation and geometry

Composite ECT/Metallurgical Correllations



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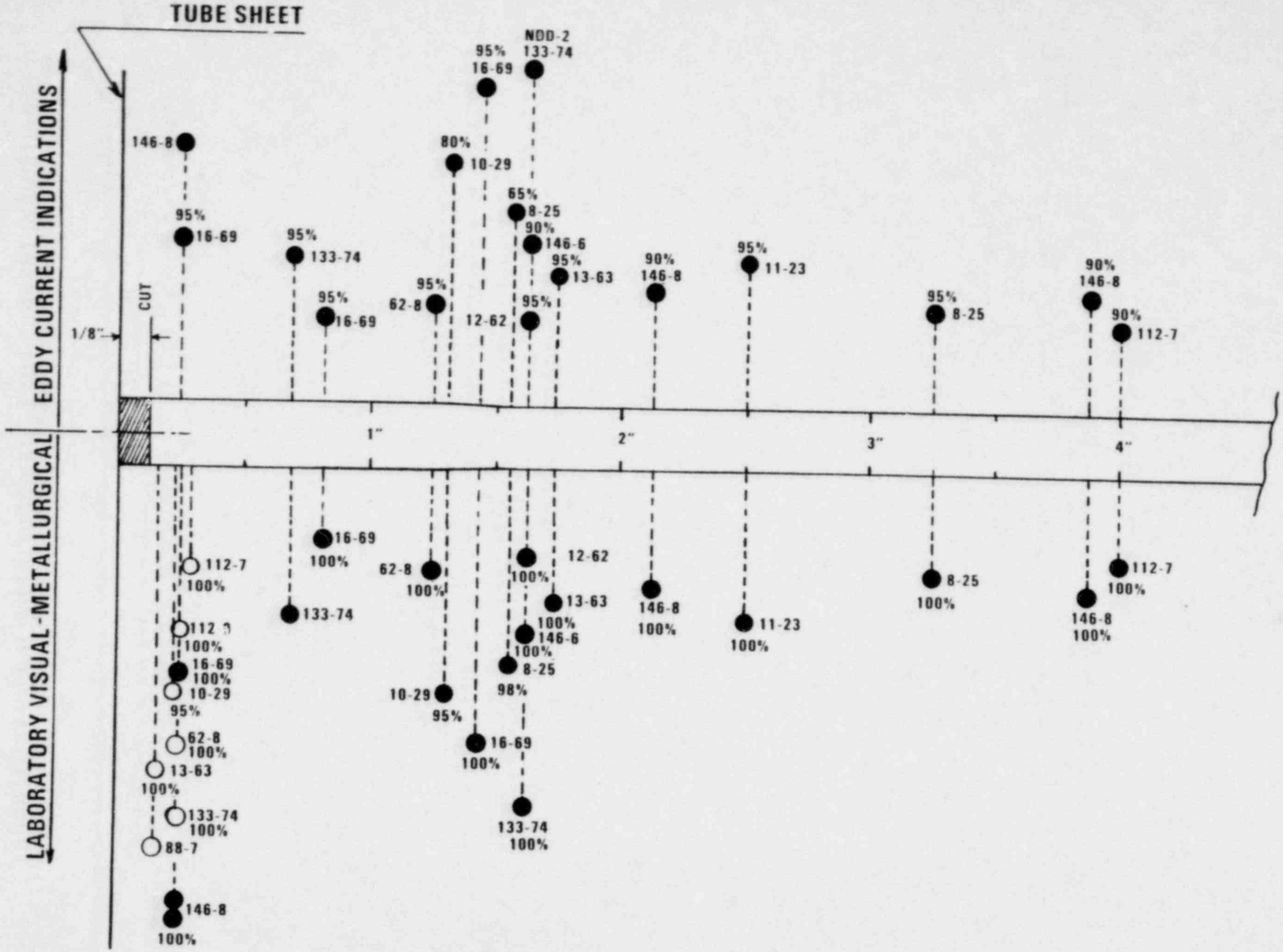
Composite ECT/Metallurgical Correllations



Correlations Among Coil Designs

I. Scope

- Standard differential vs 4X absolute

  ~435 tubes full length
  ~4500 tubes partial length

- Standard differential vs 3X pancake differential

  ~100 tubes full length

II. Conclusions

- In all cases there was good correlation

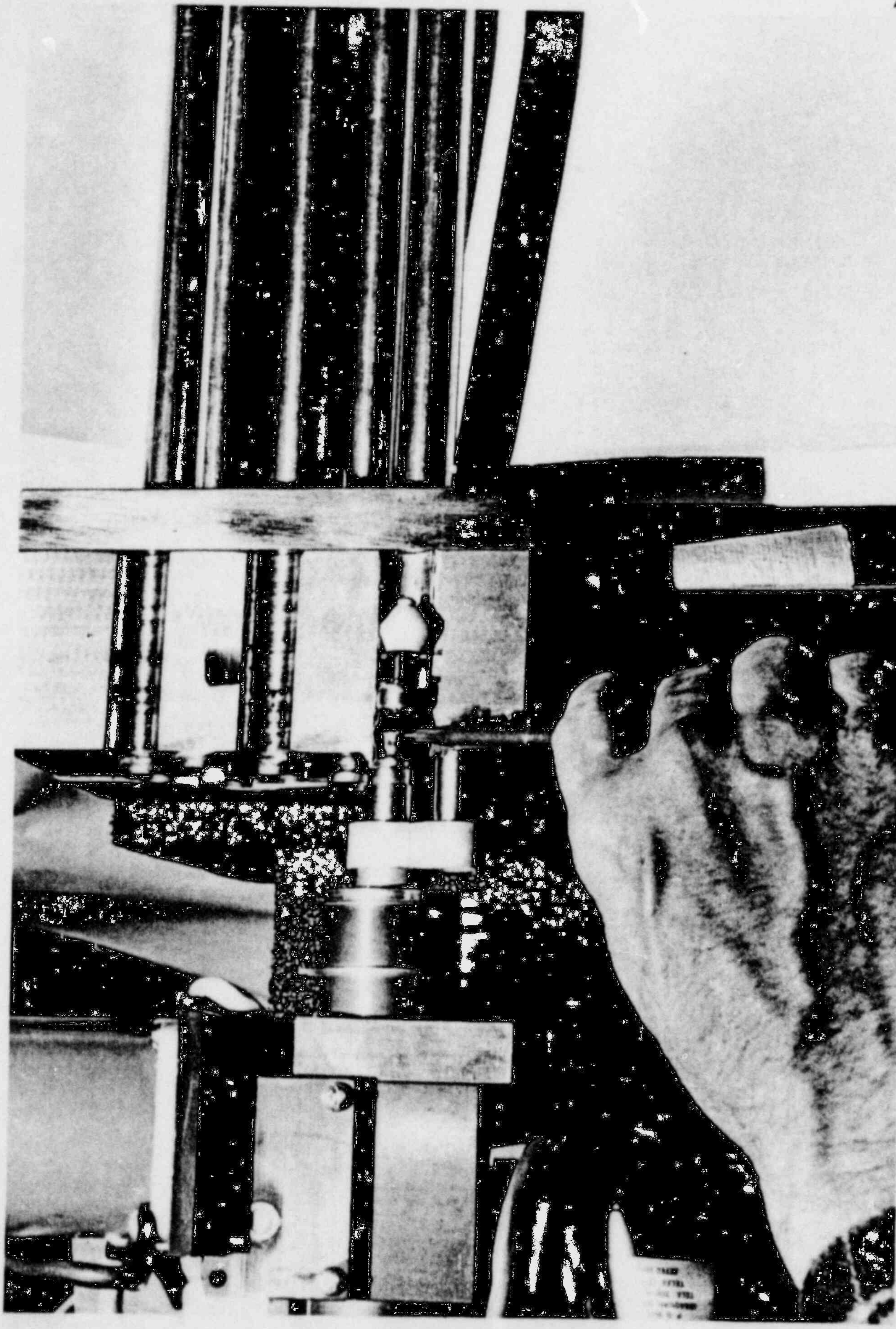
- Inconsistencies can be explained by:

  — low level signals ($<$1 volt) drop in and out by both techniques

  — resolution of multiple defects that are close together

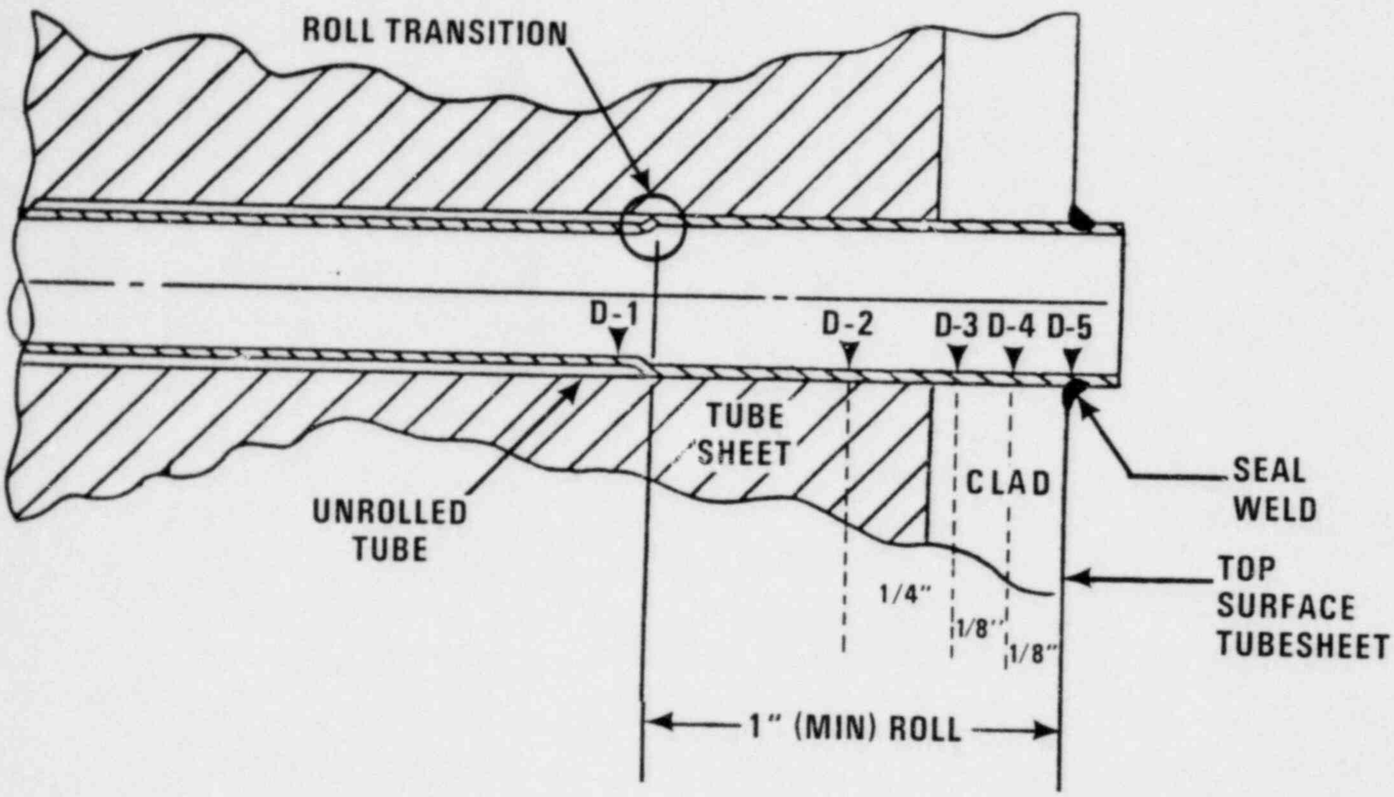
TMI-1 OTSG Absolute Eddy Current Technique Qualification

I. Metallographic analysis	—100% correlation on 30 E/C defects (top 0.25 inch excluded due to alignment problems which we are correcting)
II. Tube/tubesheet mockup testing	—demonstrates detection of simulated cracks located at the primary tube seal weld and below



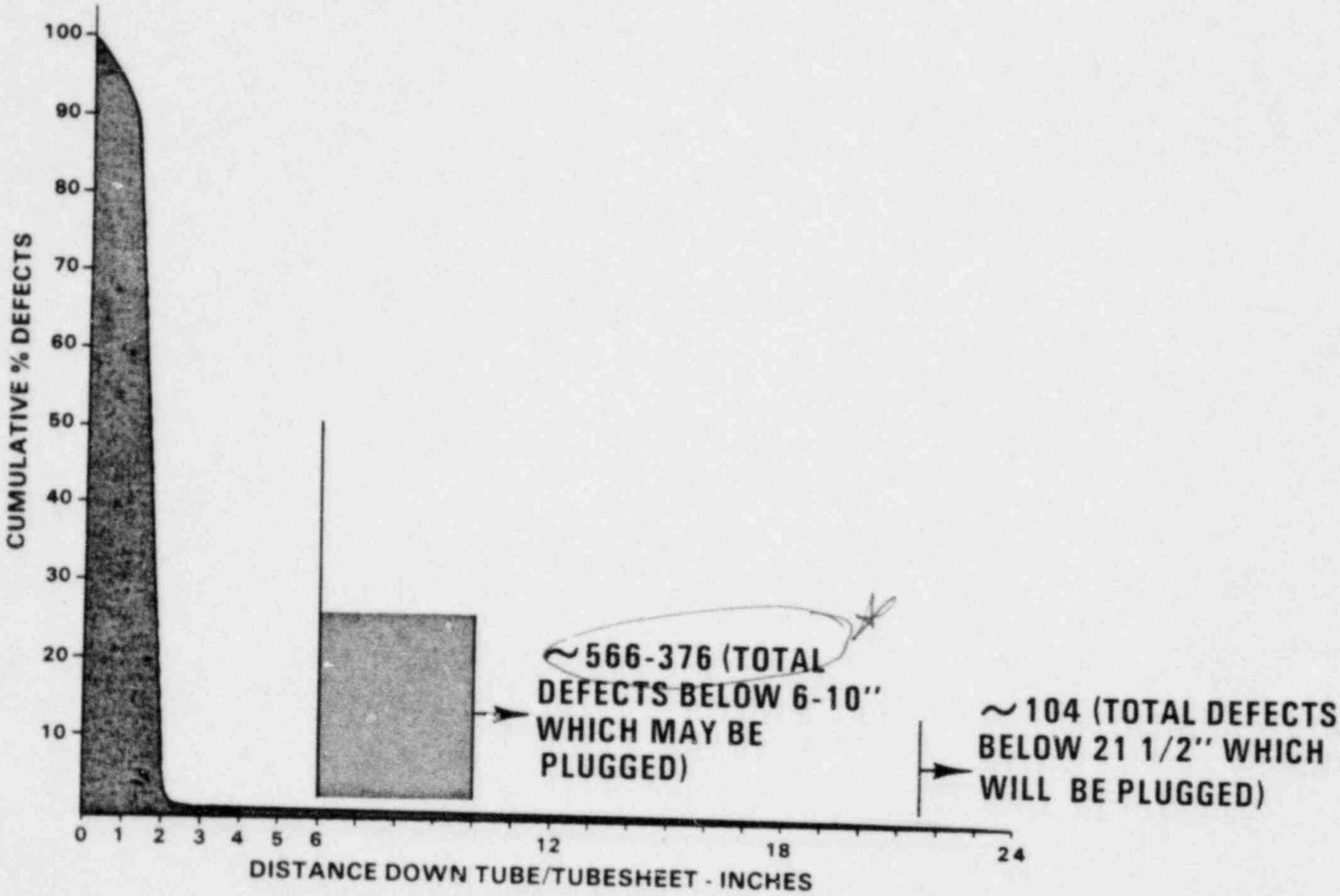
Absolute ECT Probe and Mockup

TMI-1
Eddy Current Defect Mockup Absolute Technique Qualification



- Defects #1 thru 5 (D1—D5) are ID defects located as shown
- Extent of defects varied from 20% thru 100% of tubing wall thickness — defects are 3/16 inch long (EDM notches)
- 40% and greater defect depths were detected

OTSG Tube Plugging Plans



* Updated to reflect more recent data as of 4-15-82.

Analysis of Tube Plugging Affects on OTSG Performance

- Reactor coolant system flow rate
- Safety analysis for LOCA
- OTSG exit steam quality

RC Flow Results

- Calc RC flow w/o plugging: 109.86%
- Error in calc: 1.50%
- Resulting flow: 108.36%
- Tubes plugged: 500/OTSG
- RC flow reduction: 0.25%
- Resulting flow: 108.11%
- Tech spec limit: −106.50%
- Margin: $\boxed{1.61\%}$

Conclusion

The reduction in RCS flow is acceptable

LOCA Results

Considerations

- Boiler — condenser mode heat transfer
- Initial RCS liquid inventory
- EFW spray cooling
- RC flow rate
- Core cooling

Conclusion

No effect on licensed power level of 2568 MW_T for up to 500 plugged tubes per SG

OTSG Exit Steam Superheat (100% Power Results)

- Normal superheat = 54°F
- 300 tubes plugged in one SG (uniform Dist'n)
  - —average exit superheat = 49°F
- 300 tubes plugged (25% of tubes plugged in a peripheral region)
  - —central region superheat = 54°F
  - —peripheral region superheat = 11°F
  - —average exit superheat = 49° F

Conclusion

The reduction in OTSG exit steam superheat is acceptable

Removable Plug Development

I. Objective

- Install removeable plugs in tubes which may be returned to service by sleeving

II. Type

- Roll plugs similar to those used at San Onofre

III. Qualification

- 100 thermal cycles (120°F to 650°F)
- Leak tests at $\Delta P = 2250$ psig
- Rapid cooldown from 650°F
- Simulated circumferential crack in roll
- Ejection/pull-out tests

IV. Results

* Updated to reflect leakrate at operating ΔP vice the qual. Test ΔP

- Average leak rate .03 drops/minute at operating ΔP *
- 6200-12,000 psi plug ejection pressure
- 3510 lb average pull-out load

V. Conclusion

- Roll plug qualified for intended use at TMI-1

Primary System Cleanup

I. Sulfur in RCS water has been reduced from 750ppb to 100ppb

II. If analysis shows it is required, we plan to reduce the amount of sulfur on the surfaces of primary system components and OTSG tubes

III. Cleanup method identification will consider:

- H_2O_2 concentrations of 0, 10, 100, and 1000ppm
- pH of 7.0, 8.0, and 9.0 with LiOH or NH_4OH additive
- Normal RCS chemistry

Preliminary OTSG Pre-Service Testing Plans

- **ECT** — Statistical baseline examination of the new expansion and transition
- **Drip test** — 150 psi on OTSG secondary side (H_2O)
- **Bubble test** — 150 psi on OTSG secondary side (N_2)
  — Sensitivity ~.1 gpd/tube
- **Leak test** — 2155 psi on primary side $\Delta P \geq$ 125% of normal ($\geq$1500 psi)
  — Sensitivity ~ 10 gpd (after 5 hours, current RCS activity level)
- **Power escalation testing** — Natural circulation cooldown
  Main feed pump trip (40% power)
  Turbine trip (100% power)

OTSG In-Service Monitoring

- Continuous leak rate monitoring using activity, mass balance and/or chemical concentrations
  - —sensitivity ~10 gpd
    - •after 5 hours of leakage
    - •.03% failed fuel
    - •condenser vacuum pump discharge activity
- Basis for corrective action
  - —total leak rate
  - —rate of change of leak rate

OTSG Repair Program Overview

We expect that the overall OTSG repair program, including inspections, repair process qualification, primary system cleanup, leak testing and differential pressure testing, will provide assurance that the probability of abnormal primary to secondary leakage during operation is very low.

SUMMARY

- THE REPAIR METHOD IS EXPECTED TO SHOW CONFORMANCE TO EXISTING LICENSE AND REGULATORY REQUIREMENTS

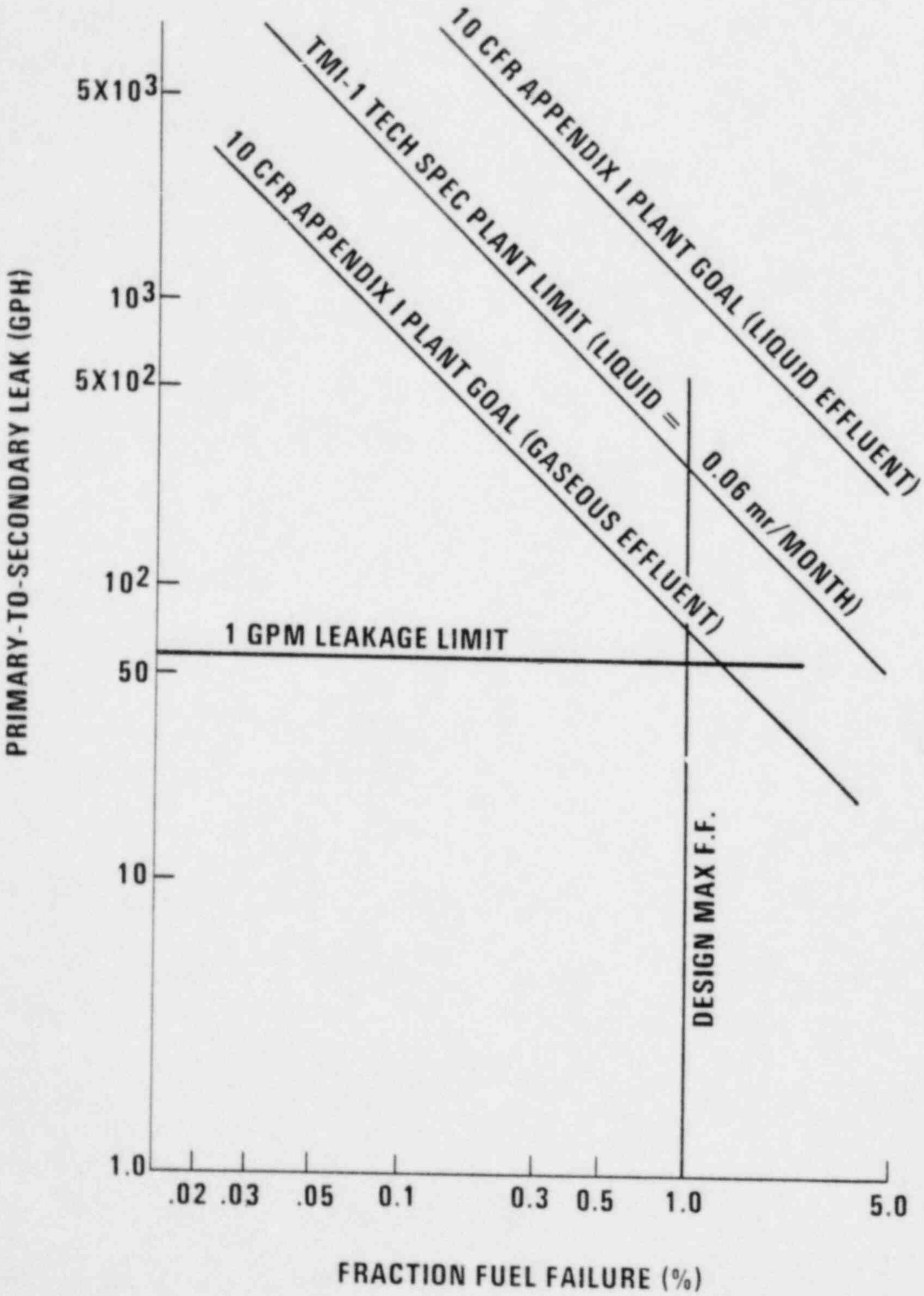
- TECH SPECS FOR APPENDIX I MAINTAIN NORMAL OPERATIONAL CONDITIONS WELL WITHIN ACCIDENT ASSUMPTIONS

- THE REPAIRED STEAM GENERATOR IS EXPECTED TO PRESENT NO SIGNIFICANT HAZARD TO STATION OR PUBLIC

REMAINING WORK

- DEVELOP, TEST, QUALIFY THE TUBE - TUBESHEET REPAIR METHOD AND PROCESS DETAILS
- RESTORE ADEQUATE STATE OF CLEANLINESS
- INSPECT OTHER PRIMARY SYSTEM INTERNALS
- COMPLETE DETAILS OF FAILURE ANALYSIS AND TECHNICAL AND SAFETY ANALYSIS OF REPAIRED STEAM GENERATORS

Current TMI-1 Operating and Effluent Limits



SAFETY EVALUATION PARAMETERS - STEAM GENERATOR REPAIR

- TUBE PLUGGING - AFFECTS PRIMARY SYSTEM FLOW
  - AFFECTS LOCA ANALYSIS (HEAT TRANSFER)

  ≤ 1000 ± TUBES MAY BE PLUGGED WITHIN BOUNDS OF EXISTING SAFETY ANALYSIS/TECH SPEC'S

- TECHNICAL SPECIFICATION LEAKAGE - UNIDENTIFIED 1 GPM
  - STEAM GENERATORS 1 GPM

- TECHNICAL SPECIFICATION RELEASE - ≤ 0.06 MR/MO WITHOUT TREATMENT

- DESIGN BASIS STEAM GENERATOR RUPTURE - 435 GPM

- APPENDIX I - LIQUIDS ≤ 3 MR/YR
  - GAS ≤ 5 MR/YR

- ALARA CONSIDERATIONS FOR SECONDARY SYSTEMS ACTIVITY, IF ANY ACTIVITY PRESENT

SAFETY REVIEW

- WILL PERFORM A COMPLETE INTERNAL REVIEW UNDER THE PROVISIONS OF 50.59

- WILL HAVE THE INTERNAL SAFETY ASSESSMENT/REVIEW FURTHER EXAMINED BY

  - THE GPUN GENERAL OFFICE REVIEW BOARD (GORB)

  - AN EXTERNAL (TO GPUN) FURTHER INDEPENDENT REVIEW GROUP

- THE REVIEW WILL BE BASED ON MEETING ESTABLISHED NRC REGULATORY REQUIREMENTS AND EXISTING TMI-1 TECH SPEC'S

CONCLUSIONS - STEAM GENERATORS

- FAILURE MECHANISM THEORY IDENTIFIED
- BASIC STEAM GENERATOR TUBE MATERIAL REMAINS GOOD
- REPAIR WILL NOT DEGRADE THE ORIGINAL DESIGN MARGINS
- CAUSATIVE CHEMICAL SPECIES DEPLETED BEFORE RESTART
- EVEN WITH LARGE NUMBERS OF TUBE FAILURES IN UTS - DESIGN BASIS TUBE RUPTURE ACCIDENT IS NOT APPROACHED OR EXCEEDED
- OPERATIONAL EXPERIENCE SUGGESTS TIGHT/VERY SMALL LEAK PATHS CLOSE DURING OPERATION
- METHODS EXIST TO CONFIRM CONTINUED SERVICEABILITY OF THE STEAM GENERATORS AFTER REPAIRS ACCOMPLISHED
    - ECT
    - ABSOLUTE LEAKAGE AND LEAKAGE TREND MONITORING
    - SECONDARY SAMPLING/RADIATION MONITORING
    - LIMITED PLANT THERMAL CYCLE TESTING