CONSOLIDATED EDISON COMPANY OF N. Y.

INDIAN POINT #2 NUCLEAR POWER PLANT

LICENSED OPERATOR REQUALIFICATION

TRAINING PROGRAM DESCRIPTION

APPROVALS:

Operations Training Administrator	2/27/90 Date
Curriculum Development Manager	3/1/90 Date
Training Manager	3/1/90 Date
Operations Manager	3/1/90 Date

PENDING NRC APPROVAL

Effective Date

9003120506 900301
PDR ADOCK 05000247
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TABLE OF CONTENTS

I. INTRODUCTION

II. PROGRAM REVIEW and APPROVAL

III. OBJECTIVES

IV. DEFINITIONS

V. RESPONSIBILITIES

VI. COURSE CONTENT

VII. PROGRAM IMPLEMENTATION

VIII. EVALUATION OF PERFORMANCE

IX. PROGRAM EVALUATION

X. RECORDS

I. INTRODUCTION

This program description establishes a continuing training program for NRC licensed personnel based on a systematic approach to performance-based training. The program meets or exceeds the requirements of Title 10 of the Code of Federal Regulations (10CFR), Part 55.59 and American National Standards Institute (ANSI) N 18.1-1971. In addition, this program was developed using the criteria established by the National Academy for Nuclear Training for program Accreditation and the Institute of Nuclear Power Operations (INPO) Guidelines for Continuing Training of Licensed Personnel (86-025).

The goal of this training program is to achieve a high level of performance and professionalism of all licensed personnel in meeting the high operational standards required to ensure nuclear plant safety and reliability.

The training program is intended to consist of approximately 200 contact hours of instruction per year. This does not include time for **General Employee Training (GET), Fire Brigade Training,** or other collateral duties. Typically, 50 hours of this time will be spent on the simulator.

The requalification program for all Reactor Operators (RO), Senior Reactor Operators (SRO), SRO Certified Instructors, and Watch Engineers/Shift Technical Advisors is intended to maintain operating competence, particularly for infrequent situations such as abnormal or emergency situations.

The lecture sets are presented to all individuals at the SRO level. Continuing training specific to the Watch Engineer will also be provided to all those participating in the Requalification Program. The simulator and in-plant performance items are evaluated at the level commensurate with the license/certification held by the individual.

Under the guidance provided in the May 26, 1987 revision to 10CFR55 for INPO accredited training programs, the selection process for topics will be based on the Systems Approach to Training and as such a strict topical prescriptive approach will not be utilized. The methods for selecting training topics will be discussed in detail in the body of this program description.

II. PROGRAM REVIEW and APPROVAL

1. The Code of Federal Regulations, Title 10 Part 55.59(c) states:

   "A facility licensee shall have a requalification program reviewed and approved by the commission".

   In addition, Part 50.54 (i-1) states:

   "Notwithstanding the provisions of Part 50.59, the licensee may not, except as specifically authorized by the Commission, decrease the scope of an approved operator requalification program".

2. To insure the above requirements are met, any revision made to this Program Description will require either of the following:

   a. A resubmittal to the Nuclear Regulatory Commission for approval

   b. A written justification by the Operations Training Administrator identifying why the revision results in no decrease in the scope of the Program. This justification will receive concurrence from the Manager Nuclear Safety and Licensing.

This Program Description and any revision to it will be reviewed at a minimum by the Administrator of Operations Training, and the Manager of Curriculum Development. Approval will be by the Manager of Nuclear Training and the Manager of Operations.

III. OBJECTIVES

This program is established to meet the following objectives:

1. Maintain and upgrade the skills and knowledge necessary to conduct routine and emergency duties.

2. Maintain an awareness of the responsibility of the licensed individual for the safe operation of the plant and the consequences of mis-operation.

3. Ensure Watch Engineers (STAs) maintain the skills necessary for accident assessment and mitigation.

4. Emphasize lessons learned from industry operating experience to prevent repetition of errors.

5. Correct performance deficiencies amenable to training solutions.

6. Evaluate systematically individual and team performance to identify areas for improvement

7. Ensure that individuals receive information on plant modifications and procedure changes in a timely manner.

8. Maintain and enhance teamwork and diagnostic skills

9. Increase the level of understanding of fundamental principles presented in initial training

10. Enhance the professionalism of licensed personnel

11. Provide input to station management for improvements in operating practices, procedures, and plant design.

IV. DEFINITIONS

1. ACTIVE LICENSE

The license held by an individual who is assigned on shift as a:

- Senior Watch Supervisor (SWS)
- Control Room Senior Reactor Operator, or
- Control Room Reactor Opertor

To maintain an active license, the individual must:

- Perform in the position for which he/she holds an active license (as identified above) by completing a minimum of seven 8-hour or five 12-hour shifts per calendar quarter, and

- Attending and successfully completing the requirements of the license requalification training program.

See Section VII.3 for requirements to resume active license status.

2. ACTIVE LICENSE FOR REFUELING

A Senior Licensed Operator assigned as Refueling SRO:

For an individual to activate his/her license solely for the purpose of refueling, see Section VII.3.

3. ANNUAL OPERATING TEST

An examination consisting of both a Simulator Performance Evaluation and a walkthrough using Job Performance Measures. The Annual Operating Test will be administered in accordance with 10 CFR 55.59(a)(ii).

4. COMPREHENSIVE WRITTEN EXAMINATION

A written examination covering those areas identified in 10 CFR 55.59(a)(i). It will be administered at the end of the Requalification Program.

5. **CONTACT HOUR OF INSTRUCTION**

A one-hour period in which the course instructor is present or immediately available for instructing or assisting students. Lectures, simulator sessions, seminars, discussions, problem-solving sessions, and examinations are considered contact periods under this definition.

6. **INACTIVE LICENSE**

The license held by an individual who attends and successfully completes the requirements of the requalification training program but is not assigned responsibility for performing the functions of a licensed operator or a senior licensed operator on shift in the position defined under Active License.

7. **JOB PERFORMANCE MEASURE (JPM)**

Test used to evaluate an operator's knowledge of, and proficiency on, a specific task. A JPM may also be used as an instructional tool when training on a task.

8. **LICENSED PERSONNEL**

Any person who possesses an active or inactive Nuclear Regulatory Commission (NRC) reactor operator license or senior reactor operator license.

9. **REACTIVITY MANIPULATION**

Any change in core reactivity resulting in a noticeable change in RCS temperature (>1°F) or in core power (>10%)

10. **REMEDIAL TRAINING**

Training provided to an individual or crew as a result of inadequate performance on walkthrough, written, or simulator performance examinations. The intent of this training is to refresh and upgrade specific knowledge and skills related to licensed operator responsibilities.

11. **REQUALIFICATION PROGRAM**

The program which establishes the requirements set forth in the Code of Federal Regulations, Title 10, Part I, Chapter 55.59. It establishes the standards and criteria by which licensed operators maintain proficiency in the safe and reliable operation of the Indian Point Unit 2 Facility. The program will not exceed 24 months in length.

12. **REQUALIFICATION TRAINING COURSE**

The Training Courses that make up the Requalification Training Program include:

- o classroom training
- o simulator training
- o on-the-job training

13. **REQUALIFICATION TRAINING CYCLE**

The amount of time required to present one increment of training to all licensed individuals.

14. **STATIC SIMULATOR**

A condition where the simulator is placed in freeze following a plant transient and examination questions are asked based on the static plant conditions.

15. **TRAINING ADMINISTRATIVE DIRECTIVE (TRAD)**

A document which establishes the standards and criteria by which the Nuclear Training department conducts its business.

V. RESPONSIBILITIES

Manager Nuclear Training

Has overall responsibility for the development and implementation of all nuclear power training programs, and ensuring all station training commitments are met.

Operations Training Administrator

Has overall responsibility for supervision of the Operations Training Section to ensure the effective coordination and implementation of the Licensed Operator Requalification Program. He is responsible for the development of the material to be presented during Operator Requalification Training. He will approve all lesson plans and simulator scenarios. He will also approve all requalification examinations. He will meet regularly with the Operations Manager to discuss program content and improvement. He will ensure that the training presented meets the needs identified by the Operations Manager, and fulfills the Training requirements specified by this Program Description.

Operations Manager

Responsible for the content of Operator training Programs. As such, he will approve the content of each requalification training course. He will establish the policy for the Conduct of Operations to be reinforced during training. In addition, he will actively participate in the annual evaluation of licensed operator simulator performance. He will conduct periodic observations and evaluations of licensed operators performing watchstanding duties.

Assistant Operations Manager

Perform functions in place of the Operations Manager as directed.

VI. COURSE CONTENTt

1. The license requalification program will run for a continuous period not to exceed 24 months in length. The development of each period will be based on a "Systems Approach to Training" and utilize the following sources to establish the course content:

   a. Annual Requalification Exam Results.

   b. Infrequent and important tasks identified to require continuing training

   c. Feedback from licensed operators, supervisors, and instructors.

   d. Plant Operating Experience/Events.

   e. Modifications to plant systems, procedures or administrative controls.

   f. Significant Industry Operating Experience.

   g. Regulatory Requirements.

   h. Fundamental knowledges that are seldom used but important to safe and reliable plant operation.

2. These sources are to be utilized as follows:

   a. ANNUAL REQUALIFICATION EXAM RESULTS

      Following each Annual Requalification Exam, an evaluation of the overall results will be conducted identifying the generic weaknesses observed. This list of weaknesses will be evaluated for incorporation into subsequent training cycles. The areas identified as weaknesses will be covered in subsequent examinations to insure they are resolved.

b. TASKS IDENTIFIED FOR CONTINUING TRAINING

- These tasks are identified in the licensed operator task lists. These are tasks that are important to plant safety and require operator proficiency, but are infrequently performed during normal operations. These tasks will be utilized to identify lecture topics, simulator scenarios, and will be part of the Requalification On-The-Job-Training (OJT) Program.

- Attachment I is a listing of these tasks current as of the writing of this Program Description. The systematic approach to the training may result in a revision to this list. Such a revision will not require a revision of this Program Description.

c. FEEDBACK

Feedback may be received in the following formats:

- from students immediately following training

- from supervisors/managers observing on-the-job performance

- from instructors conducting in-plant observation

- results of Quality Assurance (QA) audits, NRC inspections or other program evaluations.

This feedback will be evaluated for incorporation into subsequent training cycles.

d. PLANT OPERATING EXPERIENCE/EVENTS

Plant specific events as they occur will be evaluated for incorporation into the training program. The source of this information will be through **Licensee Event Reports** and **SAO-132 Event Reports.**

e. MODIFICATIONS TO PLANT SYSTEMS, PROCEDURES AND ADMINISTRATIVE CONTROLS

i. System Modifications: Modifications will be received from the Records Management Center and evaluated for training.

ii. Procedure Modifications: Procedure modifications are received from the Nuclear Power Generation Support Section. Significant procedure modifications are included in the Operations Required Reading Program. Certain procedure changes will require training prior to implementation (e.g. EOPs). These will be identified by Operations and included in a subsequent training cycle.

iii. Administrative Controls: Changes to the Station Administrative Orders (SAOs), Operations Administrative Directives (OADs), and Technical Specifications are covered by the Operations Required Reading Program.

Administrative Procedure changes requiring training prior to implementation will be identified by Operations.

f. SIGNIFICANT INDUSTRY OPERATING EXPERIENCE

The source of this information will be received through the SAO-420 system. These will consist principally of INPO SOERs and NRC IE Notices and Generic Letters.

These items will be evaluated for incorporation into the training program.

g. REGULATORY REQUIREMENTS

Items as identified by commitments made to industry regulators will be identified as such and included in the training program.

3. Any of the above discussed sources for program content are subject to the approval of the Operations Manager and Operations Training Administrator. In addition, any of these items may result in a modification of the list of tasks for continuing training. Each of the sources are to be evaluated on a continuing basis and are to be included as part of the Evaluation of the Requalification Program effectiveness.

VII. PROGRAM IMPLEMENTATION

1. Once the training proram content has been identified it will be implemented using the following methods:

   - Classroom lectures
   - Simulator training
   - On-The-Job Training

   a. **CLASSROOM LECTURES**

   Using the sources identified in Section VI, a lecture series shall be systematically selected for each cycle of training which fulfills the identified needs of licensed personnel. The lecture series shall be designed to reinforce the other program implementing processes. Scheduled plant activities will be considered in planning the lectures so that those pertaining to scheduled plant activities, such as refueling or draindown operations, are conducted during the training cycle immediately preceding such activities.

   b. **Simulator Training**

   i. Simulator Training will be structured to reflect the actual plant operating environment and to apply the theory and fundamentals presented in lecture sessions to actual plant situations. Simulator Training will be implemented using any of the following:

      - Simulator demonstrations
      - Simulator exercise scenarios
      - Job Performance Measures
      - Static Simulator

ii. The simulator will be used both in the training of individuals in position specific tasks as well as in the training of crews in team-work and diagnostic skills. In addition, the simulator will be used in the evaluation of licensed operators.

iii. Simulator training will be conducted in accordance with the Training Administrative Directive 8, on **"Conduct of Simulator Training"**. Simulator Training content will be determined using the sources identified in Section VI.

iv. The Simulator will be used to accomplish the following:

- Conduct Reactivity Manipulations
- Exercise the Emergency Operations Procedures
- Exercise the Abnormal Operating Instructions
- Conduct Infrequent Evolutions
- Exercise and further develop teamwork and diagnostic skills.
- Conduct individual and crew evaluations

v. The Requalification Program will include a minimum of 40 hours of simulator training annually.

c. **On-The-Job Training**

On-the-Job Training (OJT) will be structured to take advantage of the actual plant environment to reinforce the performance of tasks which are identified as significant and which require continuing training. In addition, the simulator will be used extensively to exercise individuals on tasks not conducive to normal plant operations.

On-The-Job Training is conducted using the following methods:

i. **Annual Requalification Document:**

Each license holder will be required to complete a Requalification OJT Document annually. This document is comprised of significant field tasks important to plant safety, but which cannot be performed on the Simulator.

ii. **Required Reading Program:**

The Required Reading Program will be conducted consistent with Operations Administrative Directive (OAD) 21 **"Operations Required Reading Program".** Required reading requirements will be determined by the Operations Manager or his delegate.

The Required Reading contents will be coordinated with the overall Requalification Program content to insure reinforcement during training if necessary.

iii. **Control Room Tasks**

The simulator will be utilized to train on significant control room tasks important to plant safety. These tasks will be on the list of tasks identified for continuing training.

2. **Requalification Training Attendance**

a. Requalification training will be attended by all licensed Indian Point Unit 2 Reactor Operators and Senior Reactor Operators, and by all Watch Engineers (Shift Technical Advisors)

b. Absences will be handled as follows:

- Individual tutoring by licensed operator instructors
- Self-study. (No more than 20% of missed lecture hours shall be completed in this manner).
- In all cases, missed exams will be made up.

c. Missed Requalification Training will be made up prior to the end of the subsequent training cycle.

3. **ACTIVE LICENSE REQUIREMENTS**

a. An active license, as defined in 10CFR 55.53e, is maintained by accomplishing the following:

Actively perform the functions of an operator or senior operator on a minimum of seven (7) eight hour, or five (5) twelve hour shifts per calendar quarter.

b. If the above requirements for an Active License are not met, before resumption of functions authorized by the license, the Operations Manager shall certify the following:

i. that the qualifications and status of the license are current and valid (verified by enrollment and participation in the Requalification Program).

ii. that the licensee has completed a minimum of forty (40) hours of shift functions under the direction of an operator or senior operator as appropriate and in the position to which the individual will be assigned. The forty hours will include a complete tour of the plant and all required shift turnover procedures.

iii. the Operations Manager may add additional requirements at his own discretion.

c. For a licensed Senior Reactor Operator to activate his license solely for the purpose of fuel handling, one shift (12 hours) of fuel handling operations will be conducted under the direction of a Senior Reactor Operator.

d. The requirements identified above for activating a license are to be documented on Attachment II and forwarded to Operations Training.

VIII. EVALUATION OF PERFORMANCE

1. Each licensed individual's performance will be evaluated during the implementation of the Requalification Program. The following minimum criteria are established for licensed operator evaluation.

   a. An annual operating test

   b. A comprehensive written examination to be given at the end of each Requalification Course. (Not to exceed 24 months between exam cycles).

   c. Each licensed individual will be satisfactorily evaluated during the course of his six (6) year license at least once by the Nuclear Regulatory Commission during a Requalification Program Evaluation as defined in Examiner Standard 601.

2. The annual operating test and comprehensive written examination will meet the requirements stated in 10 CFR 55.59 (a)(2).

3 Additional examinations will normally be administered at the end of each Requalification Training cycle. The need for such exams will be determined by the Operations Training Administrator. No more than two (2) consecutive training cycles will be conducted without examination.

4. Results of evaluations are used to identify licensee needs, retraining requirements, and validity of the training process. If generic weaknesses are identified by the evaluations, the results shall form the basis for subsequent retraining emphasis and/or special training programs.

5. The following evaluation methods will be utilized:

   a. written examination in a classroom setting (both open and closed reference)

   b. written examination using a static simulator (open reference only)

   c. Job Performance Measures (JPMs) to be performed either in the simulator, field or CCR

   d. Simulator Performance Evaluation

6. The following criteria are established for satisfactory evaluation:

   a. Written examinations - 80%

   b. Job Performance Measures - 80% of JPMs performed properly. 70% on JPM questions

   c. Simulator Performance- overall Satisfactory. Two or more Unsatisfactory can be cause for a failure. Individual Simulator Critical Tasks (ISCTs)

7. The following criteria are established for unsatisfactory performance:

   a. Unsatisfactory performance on any evaluation will require remediation. The level of remediation will be determined on a case by case basis; established by the Operations training Administrator and concurred with by the Operations Manager.

      Remediation of unsatisfactory performance will take place by the end of the subsequent Training Cycle.

b. Unsatisfactory performance on an annual operating test or comprehensive written examination will be reviewed by the Operations Training Administrator and will be submitted to the Operations Manager with a recommendation to either continue or not continue Active License Status.

c. All remediation will be administered in accordance with Training Administrative Directive 12 "Remediation" (TRAD 12).

IX. PROGRAM EVALUATION

1. The License Requalification Training Program will be evaluated for its effectiveness at least every two years. This program effectiveness evaluation will be conducted in accordance with Training Administrative Directive (TRAD) 16.

2. The evaluation, at a minimum, will look at the following:

   a. Results of Requalification Program Evaluation conducted in accordance with ES 601

   b. Results of Licensed Operator Written Examinations and Operating Tests

   c. Results of Quality Assurance Audits

   d. Plant LERs

   e. Results of outside organization inspections and evaluations (i.e., NRC, INPO)

   f. Feedback received from trainees and supervisors

The program evaluation is the responsibility of the Operations Training Administrator.

X. RECORDS

1. Requalification Program Records consist of the following:

   a. Course files as described in Training Administrative Directive 5

   b. Operator License Applications (Personal Qualification Statement) NRC FORM 398

   c. Operator License Medical NRC FORM 396

   d. Operator Examinations

LICENSED OPERATOR REQUALIFICATION PROGRAM
TASKS SELECTED FOR CONTINUING TRAINING
SIMULATOR/CLASSROOM TRAINING

ITEM NO.	TASK DESCRIPTION	TASK ID NO.
01	REACTOR STARTUP TO THE POAH	0000010301
02	PLANT SHUTDOWN	0000020301
03	MANUAL CONTROL OF FEEDWATER DURING PLANT STARTUP	0000030301
03	MANUAL CONTROL OF FEEDWATER DURING PLANT SHUTDOWN	0000040301
04	PERFORM REQUIRED ACTIONS FOR A STEAM GENERATOR TUBE RUPTURE	0000040501
05	PERFORM ACTIONS FOR LOSS OF COOLANT ACCIDENT (SMALL LEAK)	0000010501
05	PERFORM REQUIRED ACTIONS FOR LOCA INSIDE CONTAINMENT	0000100501
06	VERIFY COMPLIANCE WITH RCS LEAKAGE SPECIFICATIONS	0020060201
07	FORM REQUIRED ACTIONS FOR SATURATED RX COOLANT RESPONSE (LARGE BREAK LOCA)	0000090501
08	PERFORM REQUIRED ACTIONS FOR LOCA OUTSIDE CONTAINMENT	0000110501
09	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF INSTRUMENT AIR	0780010401
10	PERFORM REQUIRED ACTIONS FOR LOSS OF ALL AC	0000070501
10	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF A 6.9 KV BUS	0800020401
10	RECOVER FROM A LOSS OF OFFSITE POWER	0800030401
10	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF A 480 VAC BUS	0810010401
11	PERFORM REQUIRED ACTIONS FOR LOSS OF FORCED FLOW/NATURAL CIRCULATION	0000010401
12	PERFORM ACTION FOR LOSS OF ALL FEEDWATER	0000080501
13	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF SERVICE WATER	0760010401
14	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF SHUTDOWN COOLING	0050010401
14	PERFORM REQUIRED ACTIONS FOR LOSS OF RHR	0050030401
15	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF CCW	0080010401
15	FORM REQ ACTIONS/RECOVER FROM CCW INLEAKAGE	0080020401

LICENSED OPERATOR REQUALIFICATION PROGRAM
TASKS SELECTED FOR CONTINUING TRAINING
SIMULATOR/CLASSROOM TRAINING

ITEM NO.	TASK DESCRIPTION	TASK ID NO.
16	PERFORM REQUIRED ACTIONS FOR STEAM FLOW CHANNEL FAILS HIGH	0390070401
16	PERFORM REQUIRED ACTIONS FOR STEAM FLOW CHANNEL FAILS LOW	0390080401
16	PERFORM REQ ACTIONS/RECOVER FROM MBFP SPEED CONTROLLER MALFUNCTIONS	0590010401
16	PERFORM REQ ACTIONS/RECOVER FROM A MBFP TRIP	0590020401
16	OPERATE WITH A FEED FLOW TRANSMITTER OUT OF SERVICE	0590040401
16	PERFORM REQUIRED ACTIONS FOR LOSS OF FEEDWATER	0590070401
16	PERFORM REQUIRED ACTIONS FOR FEEDWATER FLOW CHANNEL FAILS HIGH	0590080401
16	PERFORM REQUIRED ACTIONS FOR FEEDWATER FLOW CHANNEL FAILS LOW	0590090401
17	PERFORM REQ ACTIONS/RECOVER FROM A LOSS OF CONDENSER VACUUM	0550020401
18	PERFORM REQ ACTIONS/RECOVER FROM A LEVEL CHANNEL FAILURE (LOW)	0110010401
18	PERFORM REQ ACTIONS/RECOVER FROM A LEVEL CHANNEL FAILURE (HIGH)	0110020401
18	RECOVER FROM A S/G LEVEL CHANNEL FAILURE (HIGH)	0190010401
18	RECOVER FROM A S/G FEED FLOW CHANNEL FAILURE (HIGH)	0190020401
18	RECOVER FROM A S/G STEAM FLOW CHANNEL FAILURE (HIGH)	0190030401
18	RECOVER FROM A S/G STEAM PRESSURE CHANNEL FAILURE (LOW)	0190040401
18	RECOVER FROM A S/G PRESSURE CHANNEL FAILURE (HIGH)	0190050401
18	OVER FROM A STEAM FLOW CHANNEL FAILURE (LOW)	0190060401
18	OVER FROM A S/G FEED FLOW CHANNEL FAILURE (LOW)	0190070401
18	RECOVER FROM A S/G LEVEL CHANNEL FAILURE (LOW)	0190080401
18	RECOVER FROM RCS TEMPERATURE CHANNEL FAILURES	1050020401
18	PERFORM REQUIRED ACTIONS FOR COLD LEG TEMPERATURE CHANNEL FAILS HIGH	1050030401
18	PERFORM REQUIRED ACTIONS FOR COLD LEG TEMPERATURE CHANNEL FAILS LOW	1050040401
18	PERFORM REQUIRED ACTIONS FOR HOT LEG TEMPERATURE CHANNEL FAILS HIGH	1050050401
18	PERFORM REQUIRED ACTIONS FOR HOT LEG TEMPERATURE CHANNEL FAILS LOW	1050060401
19	PERFORM REQ ACTIONS/RECOVER FROM A DROPPED ROD	0010010401
19	REALIGN THE CONTROL RODS	0010070401
19	OPERATE WITH A STUCK ROD	0010120401
20	PERFORM REQ ACTIONS/RECOVER FROM AN URGENT FAILURE	0010040401
20	MITIGATE THE CONSEQUENCES OF A CONTINUOUS ROD WITHDRAWAL/INSERTION	0010080401
21	PERFORM THE REQUIRED ACTIONS FOR AN EMERGENCY BORATION	0000120501
21	EMERGENCY BORATE DURING AN ATWS	0040016101
22	PERFORM THE REQUIRED ACTIONS FOR HIGH ACTIVITY IN THE RCS	0000130501
23	PERFORM REQUIRED ACTIONS FOR A TURBINE TRIP	0000060501
24	ALLY TRIP THE REACTOR UPON FAILURE OF THE AUTOMATIC TRIP SYSTEM	0120025001

LICENSED OPERATOR REQUALIFICATION PROGRAM
TASKS SELECTED FOR CONTINUING TRAINING
SIMULATOR/CLASSROOM TRAINING

ITEM NO.	TASK DESCRIPTION	TASK ID NO.
25	PERFORM REQ ACTIONS/RECOVER FROM A PRESSURE CHANNEL FAILURE (LOW)	0100010401
25	PERFORM REQ ACTIONS/RECOVER FROM A PRESSURE CONTROL CHANNEL FAILURE (HIGH)	0100020401
26	PERFORM REQUIRED ACTIONS FOR A REACTOR TRIP	0000020501
26	VERIFY AUTOMATIC ACTIONS COMPLETE FOLLOWING A REACTOR TRIP	0120015001
27	PERFORM REQUIRED ACTIONS FOR A STEAM LINE RUPTURE	0000050501
28	PERFORM REQ ACTIONS/RECOVER FROM A SOURCE RANGE FAILURE	0150010401
28	PERFORM REQ ACTIONS/RECOVER FROM AN INTERMEDIATE RANGE FAILURE	0150020401
28	PERFORM REQ ACTIONS/RECOVER FROM A POWER RANGE FAILURE	0150030401
29	CONTROL STEAM GENERATOR LEVELS FROM THE CCR USING THE #22 ABFP	0610050301
30	MONITOR THE CRITICAL SAFETY FUNCTIONS (STATUS TREES)	5030015101
31	MONITOR THE CRITICAL SAFETY FUNCTIONS (STATUS TREES)	5030035001
32	LINEUP FOR COLD LEG RECIRCULATION	0060035101
33	TRANSFER TO HOT LEG RECIRCULATION	0060045101
34	PERFORM REQ ACTIONS/RECOVER FROM A FAILURE OF THE C STEAM PRESS TRANSMITTER	0390060401
35	FILL THE ACCUMULATORS USING THE TOPPING PUMP	0200010101
36	FILL THE ACCUMULATORS USING THE HHSI PUMPS	0200020101
37	RECOVER FROM A LOSS OF SECONDARY HEAT SINK (MFW AVAILABLE)	0590026301
38	RESPOND TO A LOSS OF HEAT SINK (CONDENSATE AVAILABLE)	0560016301
39	RECOVER FROM A LOSS OF HEAT SINK USING FEED AND BLEED	0070016301
40	ART AN RCP	0030020301

LICENSED OPERATOR REQUALIFICATION PROGRAM
TASKS SELECTED FOR CONTINUING TRAINING
SIMULATOR/CLASSROOM TRAINING

ITEM NO.	TASK DESCRIPTION	TASK ID NO.
41	RESET CONTAINMENT ISOLATION SIGNALS	0130035001
42	RESET PHASE A ISOLATION SIGNAL USING THE INTERLOCK BYPASS KEYS	0130045001
43	RESET SAFETY INJECTION	0130015101
44	ESTABLISH EXCESS LETDOWN	0040035001
45	DEPRESSURIZE THE RCS USING AUXILIARY SPRAY	0040045001
46	IDENTIFY AND ISOLATE A LOCA OUTSIDE CONTAINMENT	0060065101
47	SECURE CONTAINMENT SPRAY	0260035101
48	OLATE A FAULTED S/G	0350015201
49	VENT THE ACCUMULATORS	0200015201
50	MANUALLY INITIATE CONTAINMENT SPRAY	0260015101
51	MANUALLY PLACE THE FAN COOLER UNITS IN THE INCIDENT MODE	0220035101

Total Records Displayed: 87

LICENSED OPERATOR REQUALIFICATION PROGRAM
TASKS SELECTED FOR CONTINUING TRAINING
ON-THE-JOB/CLASSROOM TRAINING

ITEM NO.	TASK DESCRIPTION	TASK ID NO.
52	STARTUP THE EDG'S	0640010404
53	SYNCHRONIZE THE EDG'S	0640020404
54	SHUT THE MSIV'S LOCALLY	0390040401
55	OPERATE THE ATMOSPHERIC STEAM DUMPS LOCALLY	0390010401
56	OPERATE THE AFW FEED REGULATORS LOCALLY	0610020401
57	START THE #22 AFWP LOCALLY	0610010401
58	RESET THE #22 AFWP	0610050401
59	PLACE A STATIC INVERTER IN SERVICE	0820040201
60	PLACE THE 1 KVA TOPAZ INVERTER IN SERVICE	0820050201
61	SHED UNNECESSARY DC LOADS	0840015501
62	LOCALLY START THE FCU AND SWP FOLLOWING A BLACKOUT WITH THE CONTROL ROOM	0220020401
63	STARTUP A 125V DC BATTERY CHARGER	0630040304
64	TRIP THE REACTOR FROM THE CABLE SPREADING ROOM	0840015001
65	TRIP THE REACTOR FROM THE 480/6.9 KV SWITCHGEAR	0840025001
66	START THE #21 ABFP FROM THE SAFE SHUTDOWN POWER SUPPLY	0840030401
67	OPERATE THE #24 SERVICE WATER PUMP FROM THE SAFE SHUTDOWN POWER SUPPLY	0840100401
68	OPERATE THE #23 CCW PUMP ON THE ALTERNATE SAFE SHUTDOWN POWER SUPPLY	0840110401
69	OPERATE THE #23 CCP FROM THE SAFE SHUTDOWN POWER SUPPLY	0840120401

LICENSED OPERATOR REQUALIFICATION PROGRAM
TASKS SELECTED FOR CONTINUING TRAINING
ON-THE-JOB/CLASSROOM TRAINING

ITEM NO.	TASK DESCRIPTION	TASK ID NO.
70	ALIGN 23 CCP TO CONTROL PZR LEVEL FROM ALTERNATE POWER SUPPLY USING SCOOP TUBE	0840080401
71	OPERATE THE #21 SI PUMP ON SAFE SHUTDOWN POWER SUPPLY	0840130401
72	ALIGN THE #21 SI PUMP TO THE ALTERNATE POWER SUPPLY	0840140401
73	OPPERATE THE #21 RHR PUMP ON THE SAFE SHUTDOWN POWER SUPPLY	0840090401
74	OPERATE THE ZURN STRAINERS MANUALLY	0760010404
75	RESET AND OPEN THE #22 ABFP STEAM SUPPLY SHUTOFF VALVES	0610060301
76	RESET FCU SERVICES AND CCR VENTILATION	0220015101
77	TROL S/G LEVELS LOCALLY USING #21 ABFP	0610060401
78	CONTROL S/G LEVELS LOCALLY USING #23 ABFP	0610070401
79	LINEUP CITY WATER TO THE CHARGING PUMPS WITH THE PUMPS SHUTDOWN	0040040404
80	LINEUP CITY WATER TO THE CHARGING PUMPS WITH THE PUMPS OPERATING	0040050404
81	LINEUP CITY WATER TO THE SIS AND RHR PUMPS WITH THE PUMPS SHUTDOWN	0060010404
82	LINEUP CITY WATER TO THE SIS AND RHR PUMPS WITH THE PUMPS OPERATING	0060020404
83	LINEUP PRIMARY WATER TO THE HHSI AND RHR PUMPS WITH THE PUMPS SHUTDOWN	0060030404
84	LINEUP PRIMARY WATER TO THE HHSI AND RHR PUMPS WITH THE PUMPS OPERATING	0060040404

Total Records Displayed: 33

ATTACHMENT II

ACTIVE LICENSE CERTIFICATION

NAME: _____ CONTROL NUMBER: _____

LICENSE POSITION: SWS____ SRO____ RO ____ SRO_____ (Refueling Only)*

REQUIREMENTS

1. Qualifications and Status of License are current and valid.

_____
Operations Manager

2. Completed forty (40) hours (1 [12] hour shift for Refueling SRO) of under instruction watchstanding in the position to which you will be assigned.

DATE	HOURS	SWS INITIALS

3. The forty (40) hours of Under Instruction Watchstanding include:

a. Plant Tour: _____ SWS Initials _____ Date

b. Review Shift Turnover Procedures:

_____ SWS Initials _____ Date

4. Final Qualification: _____ Operations Manager Signature _____ Date

***FOR REFUELING SRO COMPLETE 1, 2, 36, 4**

ENCLOSURE 1

Attendance List

March 14, 1990 Meeting to Discuss Results of Mid-Cycle Inspection of Indian Point 2 Steam Generator Girth Welds

Name	Title	Organization
Donald Brinkman	Sr. Project Manager	NRC/NRR/PDI-1
Ray Burns	Chemistry Engineer	Con Ed-Indian Point III
Charles J. Durkin, Jr.	VP - Engineering	Con Ed
Sam Sinha	Stress Analysis, Mech-Eng	Con Ed
Warren Bamford	Fellow Engineer, Matls Tech.	Westinghouse
John Houtman	Mgr., Struct. Mech.	WNSD
Donald Lindgren	Sr. Engineer, Nuclear Safety	Westinghouse, NATD
Peteris Skulte	Sr. Eng. Mech Eng	Con Ed
Robert R. Laubham	Sr. Eng. W Nuclear Safety	Westinghouse, NATD
Stephen P. Swigart	Project Manager	Westinghouse
Serge Brozski	Sr. Q.A. Engineer	Con Ed
G.C. Millman	Section Leader, RES/EMEB/ES	NRC
Gutti V. Rao	Sr. Engineer, Materials	Westinghouse
Robert A. Hermann	Chief, Metallurgy Section	NRC/EMCB
Barry Elliot	Sr. Matl. Eng.	NRC
William P. Bennet	Project Eng.	Con Ed
Jack Strosnider	Chief, Materials & Processes Sec.	NRC/Region I
Alfred Lohmeier	Reactor Engineer	NRC/Region I
C.D. Sellers	Sr. Mtls. Eng.	NRC/NRR/EMCB
H. Kaplan	Reactor Eng.	NRC/REG. 1
George Galanes	Principal Met. Engr.	Commonwealth Ed
A.R. Vaia	Mgr, S/G Services Engr	W NSD
M.R. Hum	Materials Eng.	NRC
J.D. Neighbors	Sr. Project Manager	NRC
M. Marina	Manager Engineering	Con Ed
R. Lambert	Mechanical Eng.	Con Ed
J. Ellwanger	Nuclear Safety Prin. Eng.	Con Ed
Michael L. Miele	Gen'l Mgr. Technical Services	Con Ed
C.Y. Cheng	Chief, Material & Chemical Eng.	NRC
J.E. Richardson	Director, Div. Eng. Tech.	NRR
S. Varga	DRP/NRR	NRC
R. Capra	Director, PDI-1, NRR	NRC
C.W. Jackson	Manager Nuclear Safety & Lic.	Con Ed
Angie P. Young		NRC/NRR/OEAB

ENCLOSURE 2

STATUS REPORT

INDIAN POINT UNIT NO. 2

MID-CYCLE STEAM GENERATOR INSPECTION

MARCH 14, 1990

Con Edison

AGENDA

1. Introduction
2. Recent Actions, Operating History, and Radiation Exposure
3. Midcycle Inspection Results and Current Repairs
4. Boat Sample Results
5. Integrity Analysis
6. Root Cause
7. Future Plans

Summary

1. Four Planned Shutdowns and One Reactor Trip
2. 183.3 Hours at Hot Standby for Current Cycle Compaired with 502.2 Hours for the Previous Cycle
3. Hot Standby Defined as RCS > 200 F <u>With</u> Auxiliary Feedwater in Operation.
4. Parameters Recorded and Evaluated.

MAJOR EVENTS

1. JUNE 28, 1989 TO JULY 28, 1989
   - COVERS RCS HYDRO AND PLANT STARTUP FROM 1989 REFUELING OUTAGE

2. OCTOBER 24, 1989 TO NOVEMBER 23, 1989
   - COVERS SHUTDOWN ACTIVITIES IN CONNECTION WITH REPAIR ACTIVITIES ON INSPECTION PORTS AND STARTUP

3. DECEMBER 10, 1989 TO DECEMBER 14, 1989
   A) #22 STEAM GENERATOR CHEMISTRY LINE LEAK - HOT SHUTDOWN
   B) REACTOR TRIP ON LOW TURBINE OIL PRESSURE CONTROL

4. DECEMBER 23, 1989 TO DECEMBER 30, 1989
   - REACTOR SHUTDOWN FOR REPAIR OF PRESSURIZER SPRAY VALVE

5. FEBRUARY 23, 1990 TO APRIL 3, 1990
   - SHUTDOWN FOR MIDCYCLE STEAM GENERATOR INSPECTION

PLANT OPERATING CHARACTERISTICS

"CONSOLIDATED EDISON SHALL MONITOR THE NUMBER OF TRIPS, HOURS AT HOT STANDBY AND WATER CHEMISTRY EXCURSIONS. IF ANY OF THE ABOVE EXCEEDS THE PERFORMANCE DURING THE ENTIRE PREVIOUS FULL OPERATING CYCLE, YOU SHALL PROMPTLY INFORM THE STAFF."

Recent Actions to Alleviate Steam Generator Indications

1. Removal of downcomer flow resistance plate - June of 1989

2. Installation of timer to delay closure of the main feedwater bypass regulating valves/ reactor trip - September of 1989

3. Installed taps on condensate discharge and return lines for oxygen scavenging skid.

CYCLE 10 CHEMISTRY TRANSIENTS

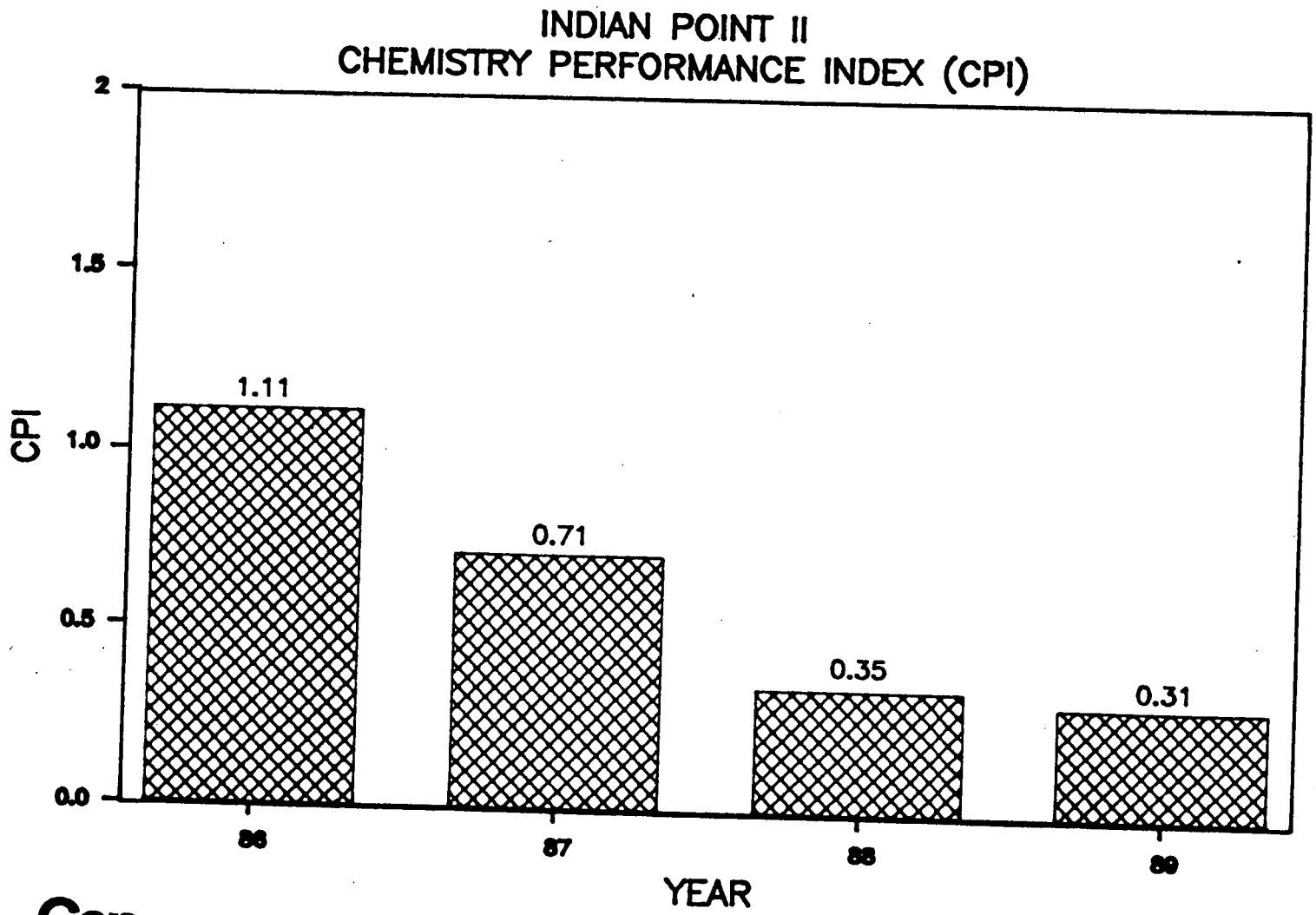
CONDENSER LEAKS

JULY 17, 1989 (power reduction to 80% required – 60ppb Cl max)
AUGUST 23, 1989
SEPTEMBER 11, 1989
NOVEMBER 24, 1989
DECEMBER 26, 1989

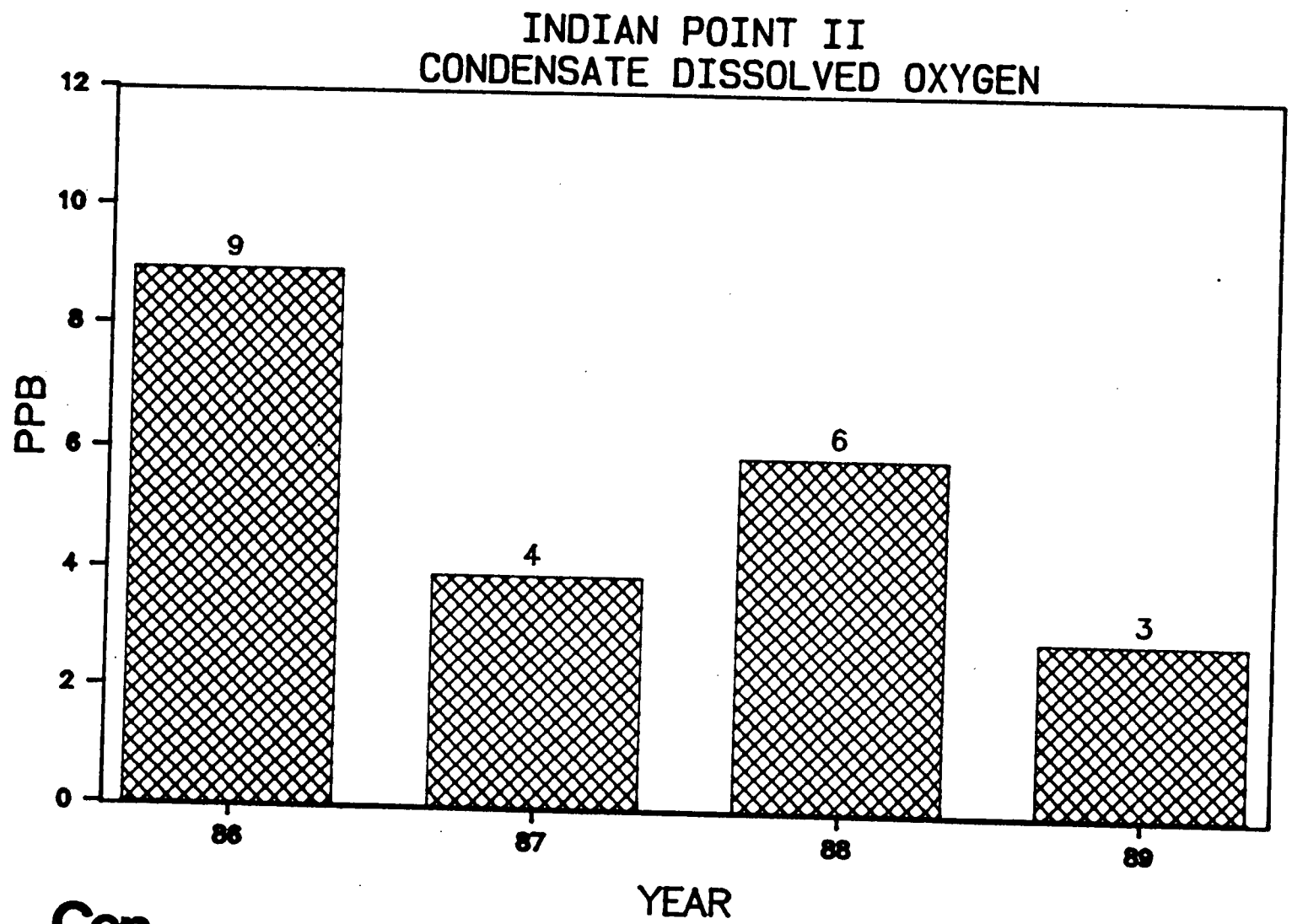
UNIT START UP

JULY 2, 1989 (6 day, 8 hour hold at 30% power)
NOVEMBER 16, 1989 (30 hour hold)
DECEMBER 11, 1989 (6 hour hold)
DECEMBER 13, 1989 (6 hour hold)
DECEMBER 26, 1989 (6 hour hold)

Con Edison

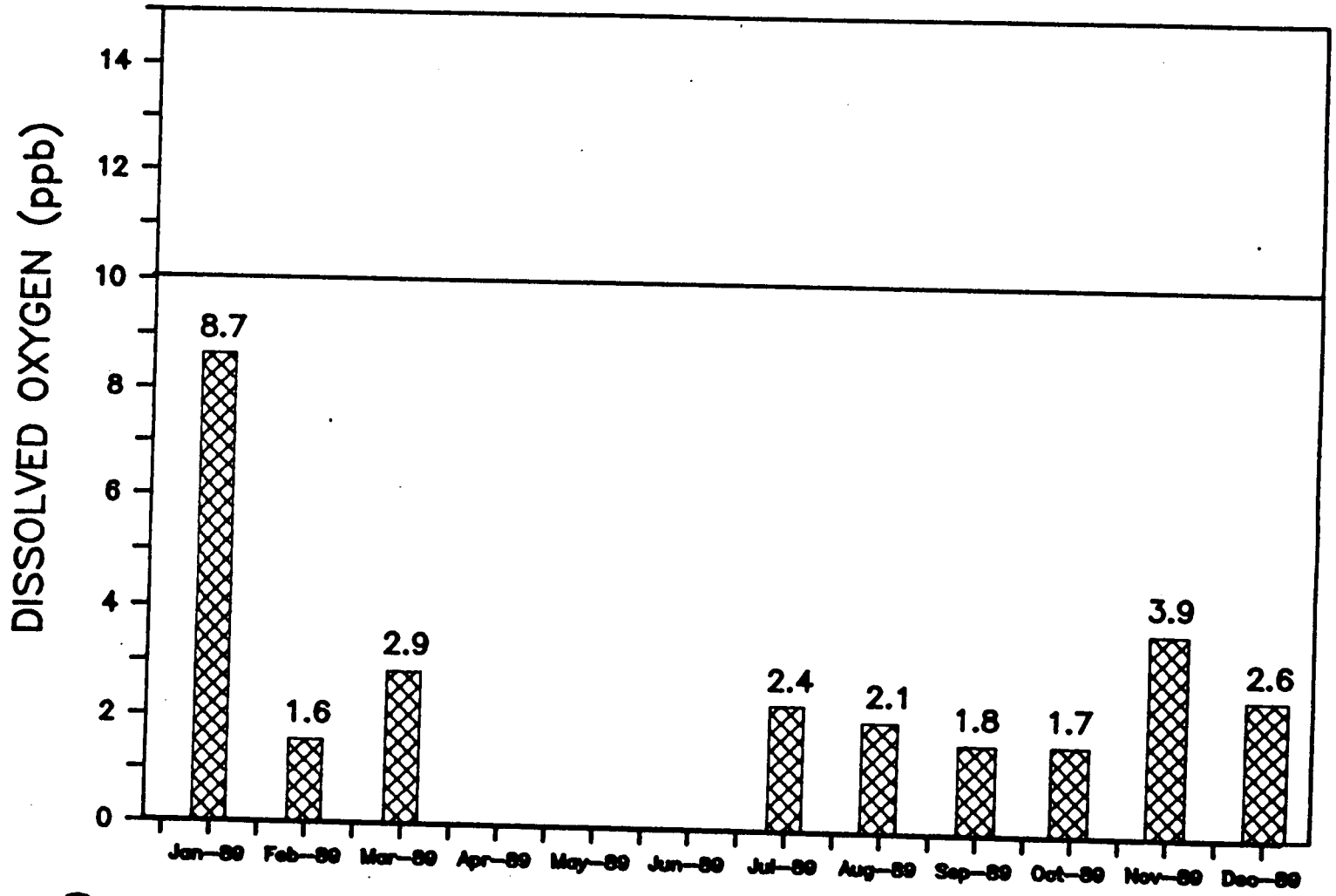

INDIAN POINT II
CHEMISTRY PERFORMANCE INDEX (CPI)
CPI
2
1.5
1.0
0.5
0.0
1.11
0.71
0.35
0.31
86
87
88
89
YEAR

Con Edison


INDIAN POINT II
CONDENSATE DISSOLVED OXYGEN
PPB
12
10
8
6
4
2
0
9
4
6
3
86
87
88
89
YEAR

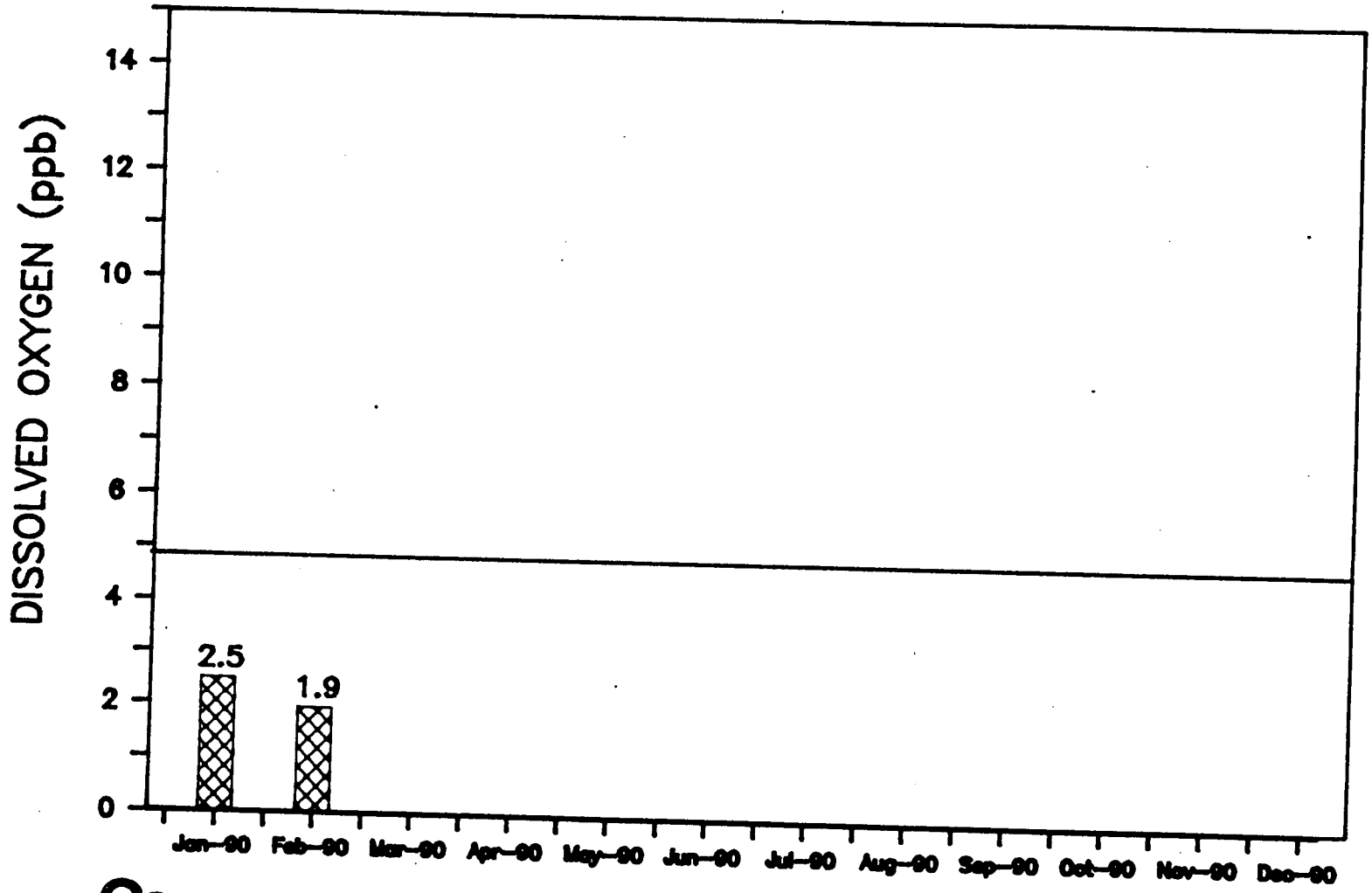
Con Edison

CONDENSATE PUMP DISCHARGE DISSOLVED OXYGEN – 1989

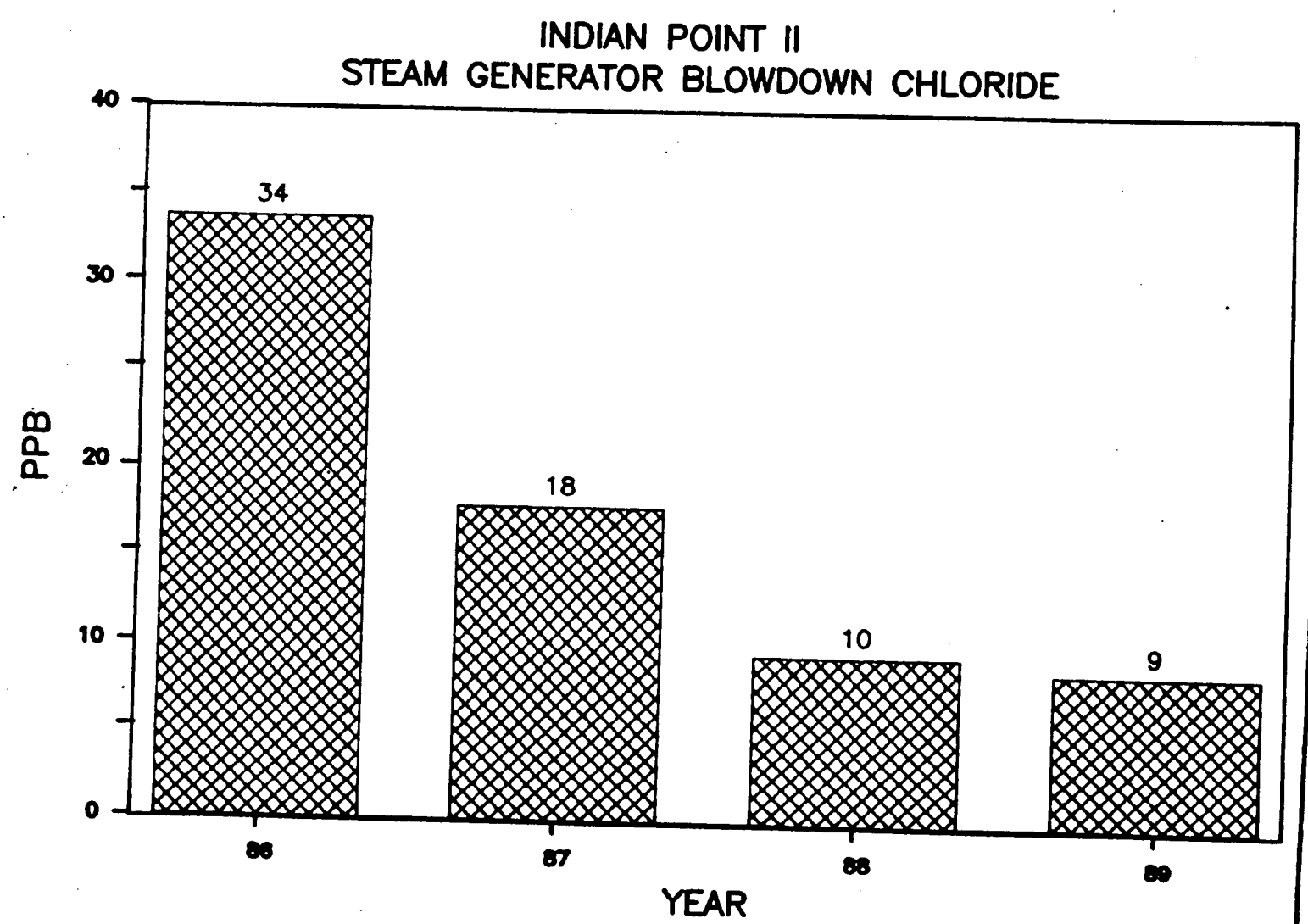


Con Edison

CONDENSATE PUMP DISCHARGE DISSOLVED OXYGEN – 1990

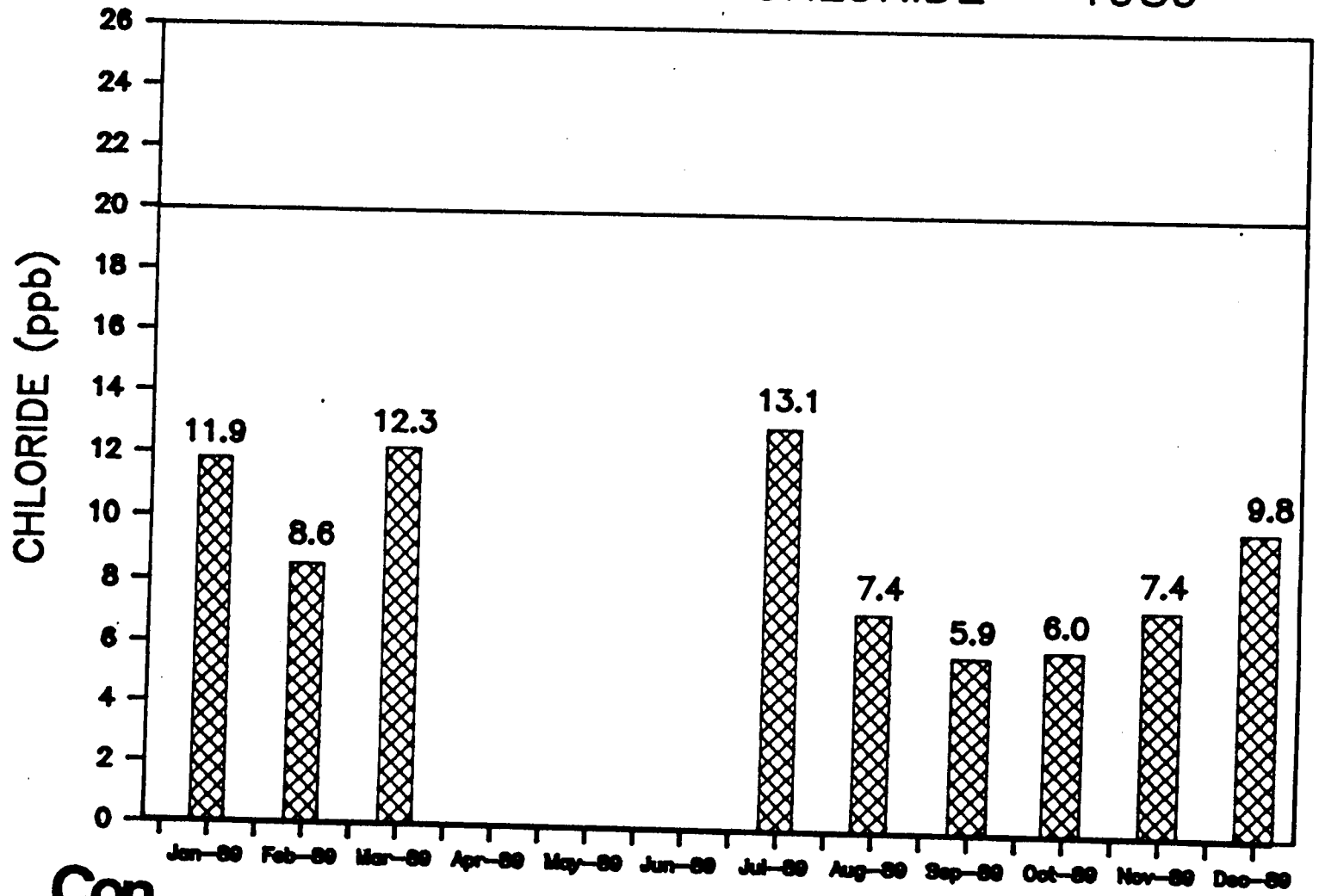


Con Edison


INDIAN POINT II
STEAM GENERATOR BLOWDOWN CHLORIDE
PPB
40
30
20
10
0
34
18
10
9
86
87
88
89
YEAR

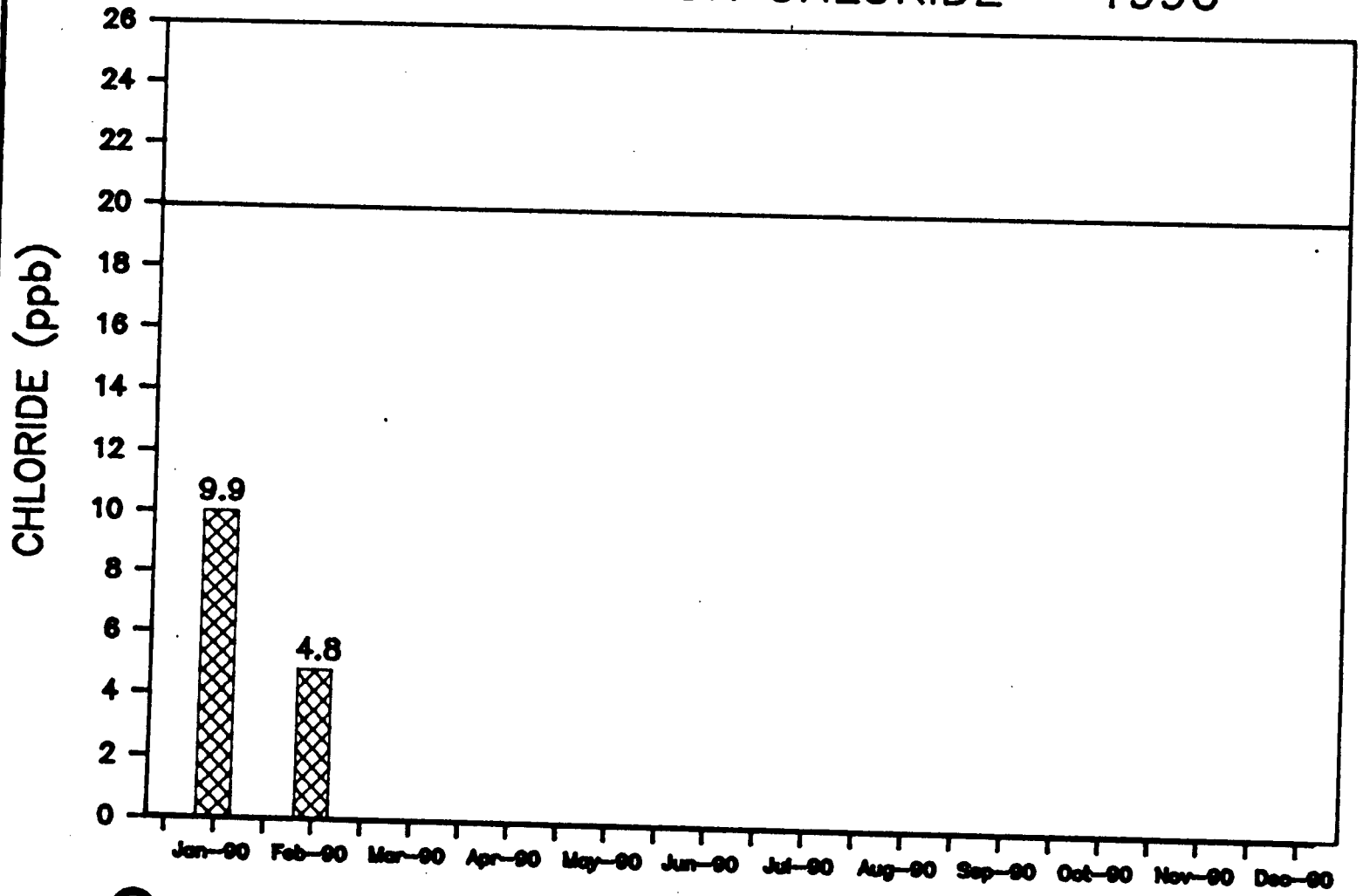
Con Edison

STEAM GENERATOR CHLORIDE – 1989



Con Edison

STEAM GENERATOR CHLORIDE – 1990



Con Edison

Radiological Exposure/Steam Generators

1987 - 199 Manrem

1989 - 135 Manrem

1990 - 145 Manrem

Con Edison

Indian Point 2
Steam Generators

Inspection and Repair

- o Inspection Program
- o Status to Date
- o Inspection Results
- o Repairs

INDIAN POINT 2
STEAM GENERATORS

INSPECTION PROGRAM

- GIRTH WELDS

  1/3 OF 22 AND 23
  IF INDICATIONS FOUND THERE
  100% 22 AND 23 AND 1/3 OF 21
  AND 24
  IF INDICATIONS FOUND, THEN 100%

  NOZZLES / BRACKETS

  LOWER 1/2 OF 22 AND 23
  IF INDICATIONS FOUND THERE
  LOWER 1/2 OF 21 AND 24
  IF INDICATIONS FOUND, THEN
  EVALUATE

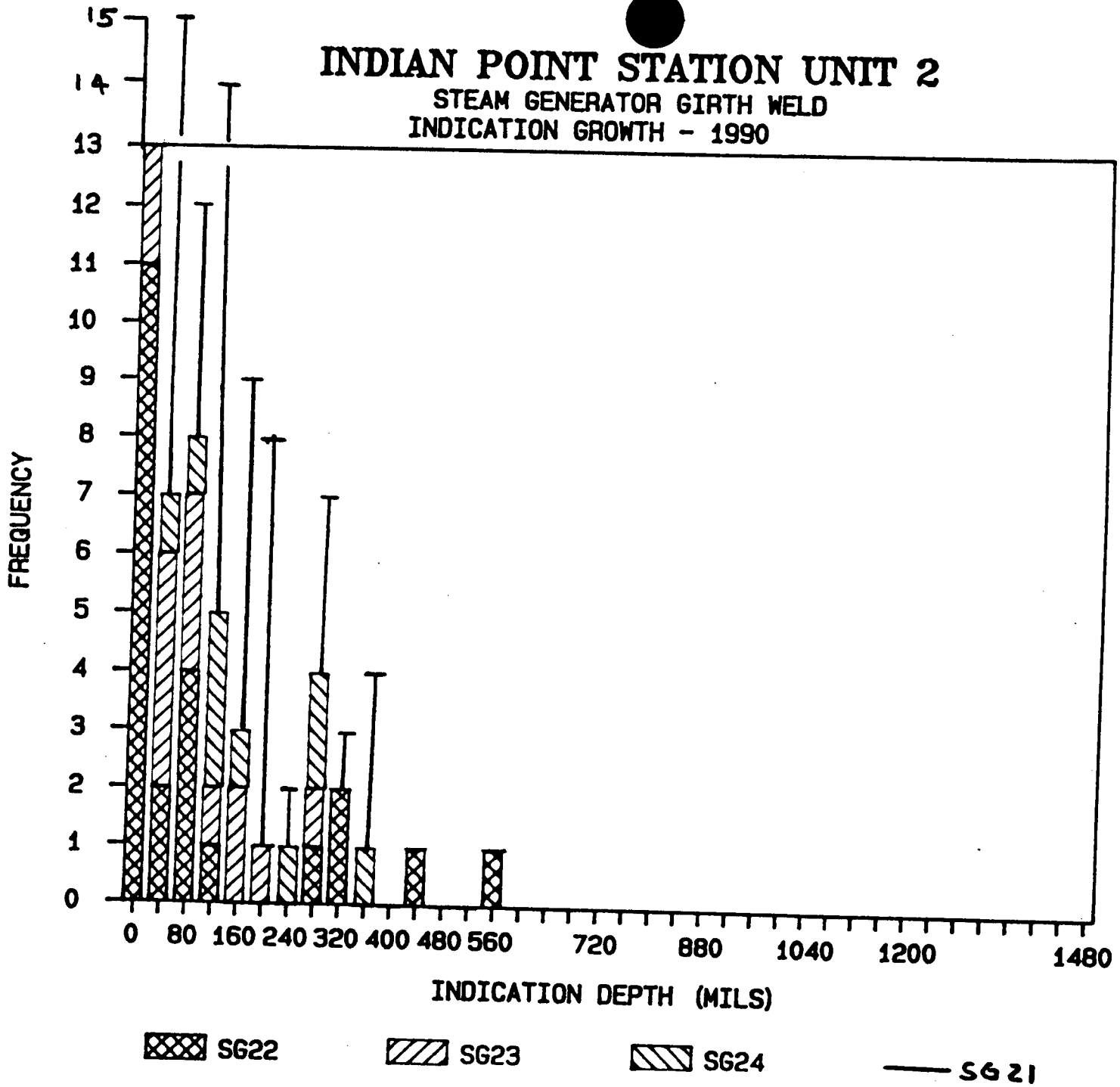
Indian Point 2
Steam Generators

Girth Weld

	SG21	SG22	SG23	SG24
1990 # of Indications	42**	26	14	10
Max. Depth*	0.36"	0.53"	0.28"	0.34"
Avg. Depth	0.13"	0.11"	0.08"	0.16"

* Depth of Grindout

**4 additional indications were seen above the MT inspection zone in an apparent temporary weld attachment


INDIAN POINT STATION UNIT 2
STEAM GENERATOR GIRTH WELD
INDICATION GROWTH - 1990
FREQUENCY
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
0
80
160
240
320
400
480
560
720
880
1040
1200
1480
INDICATION DEPTH (MILS)
SG22
SG23
SG24
SG21

3/11/90

Indian Point 2
Location of Girth Weld

Girth Weld Indications

SG	Original Surface	Previous Indications	Grindout Slopes	Total
21	31	3	8	42
22	7	3	16	26
23	10	2	2	14
24	9	0	1	10
Total	57	8	27	92

Indian Point 2
Steam Generators

Indications Present

	SG21	SG22	SG23	SG24
Nozzle	*	0	1	*
At FW Bracket	*	Yes	Yes	*

* Not Yet Inspected

INDIAN POINT 2
STEAM GENERATORS

REPAIRS

- o REMOVE INDICATIONS
- o REWELD AS NEEDED

GIRTH WELDS

SG 21 REMOVE INDICATIONS ONLY

SG 22 AFTER REMOVAL OF INDICATIONS BY GRINDING, NO AREA WILL BE LEFT WITH EXCAVATIONS GREATER THAN 0.70 IN.

SG 23 REMOVE INDICATIONS ONLY

SG 24 REMOVE INDICATIONS ONLY

METALLURGICAL INVESTIGATION OF THE RE-OCCURRENCE OF STEAM GENERATOR GIRTH WELD CRACKING AT IP 2

GUTTI V. RAO
WESTINGHOUSE NATD

MAJOR TASKS

1. SURFACE EXAMINATIONS IN THE AS-RECEIVED CONDITION (LIGHT OPTICAL & SEM TECHNIQUES)

2. NDE EXAMINATIONS (MT & PT)

3. METALLOGRAPHIC EXAMINATIONS BY (LOM, SEM TECHNIQUES)

4. FRACTOGRAPHIC EXAMINATIONS (LOM, SEM & TEM TECHNIQUES)

5. HARDNESS MEASUREMENTS (KNOOP & ROCKWELL)

6. CHEMISTRY EXAMINATIONS (EDS & MICROPROBE ANALYSES)

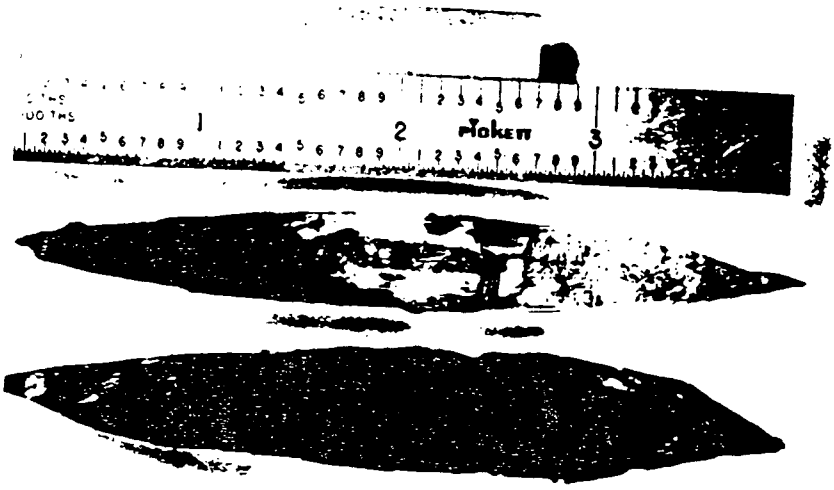
METALLURGICAL EXAMINATION OF BOAT SAMPLES

OBJECTIVES:

- TO ESTABLISH THE CAUSE AND MECHANISM OF CRACKING
- TO DEVELOP INFORMATION THAT WOULD BE HELPFUL IN FORMING CORRECTIVE ACTIONS

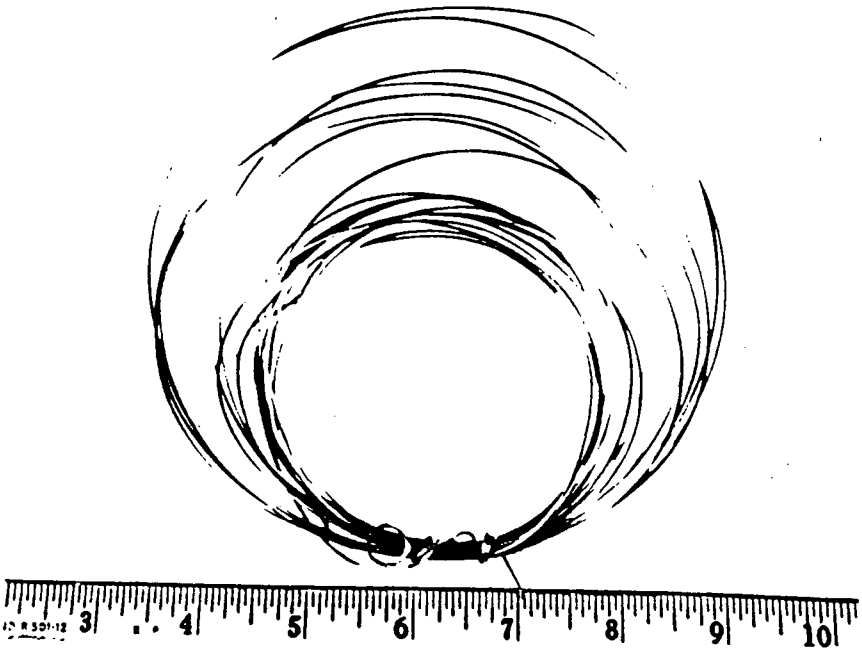
SAMPLES

o BOAT A	SG #21	SHALLOW GRINDOUT AREA
o BOAT B	SG #22	PREVIOUSLY WELD REPAIRED AREA



◄Boat – A, SG.21

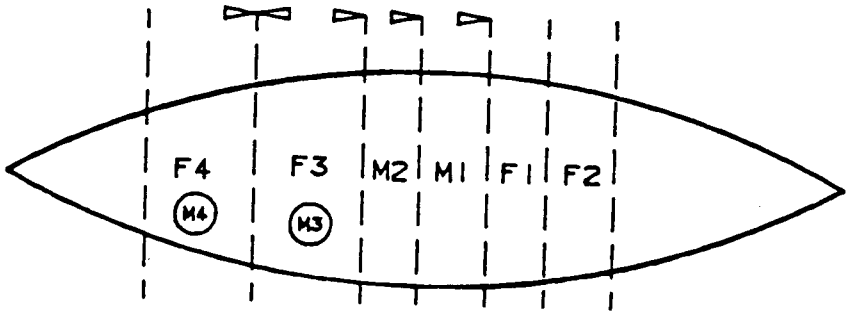
◄Boat – B, SG.22



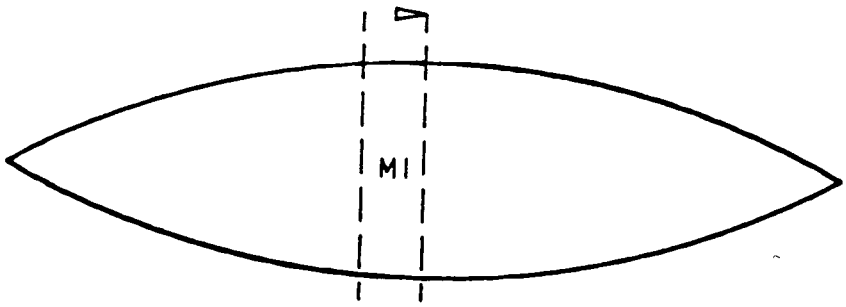
◄Sample "C"
(Weld Wire)

Macro Photos Illustrating the Appearence of the As–Received Samples

SECTIONING PROCEDURE OF THE TWO BOATS



PCE - A
SG.21

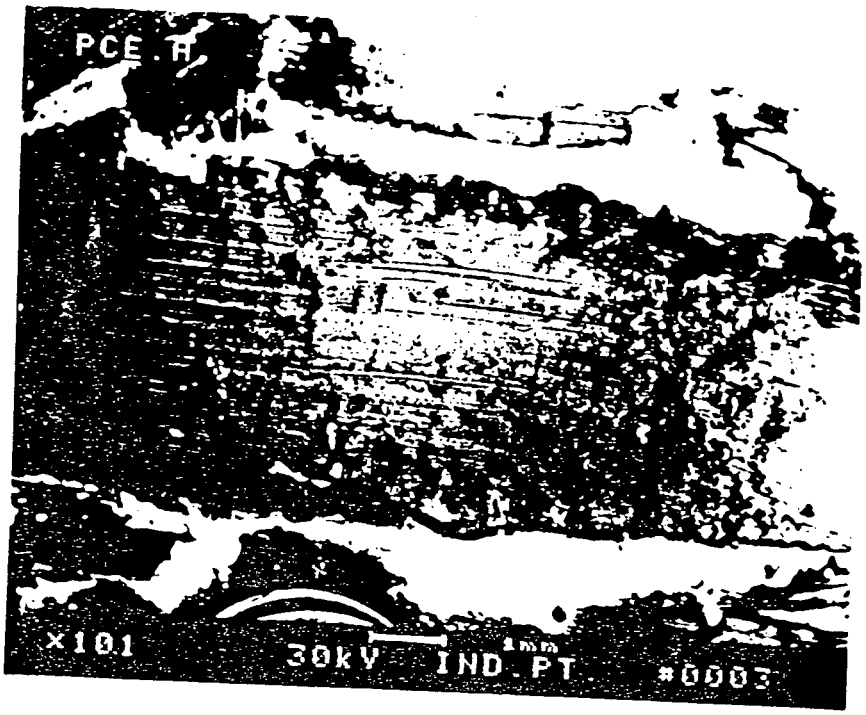


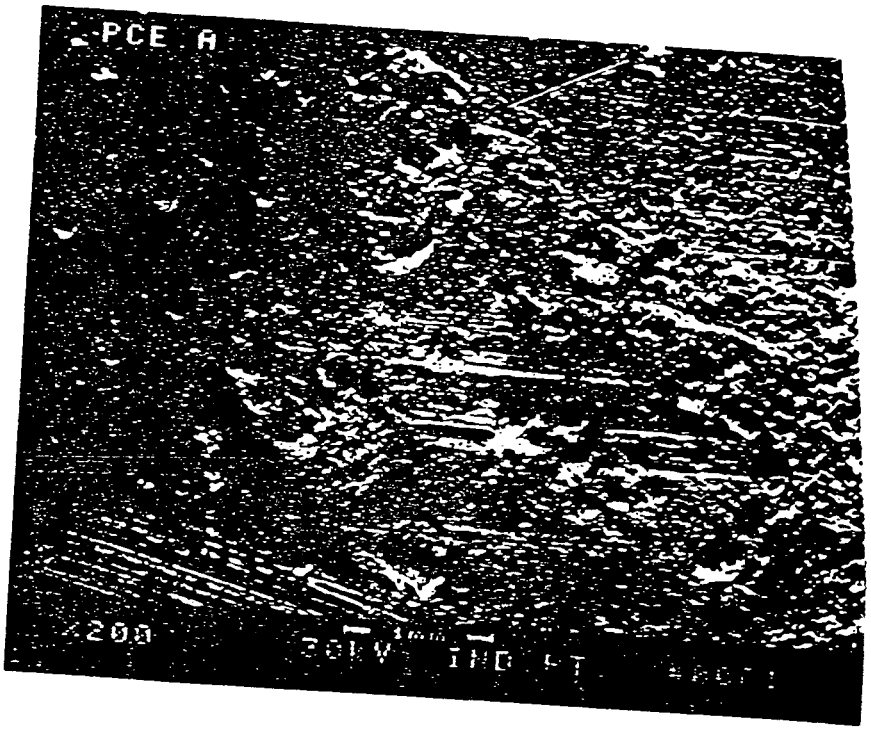
PCE - B
SG.22

6X

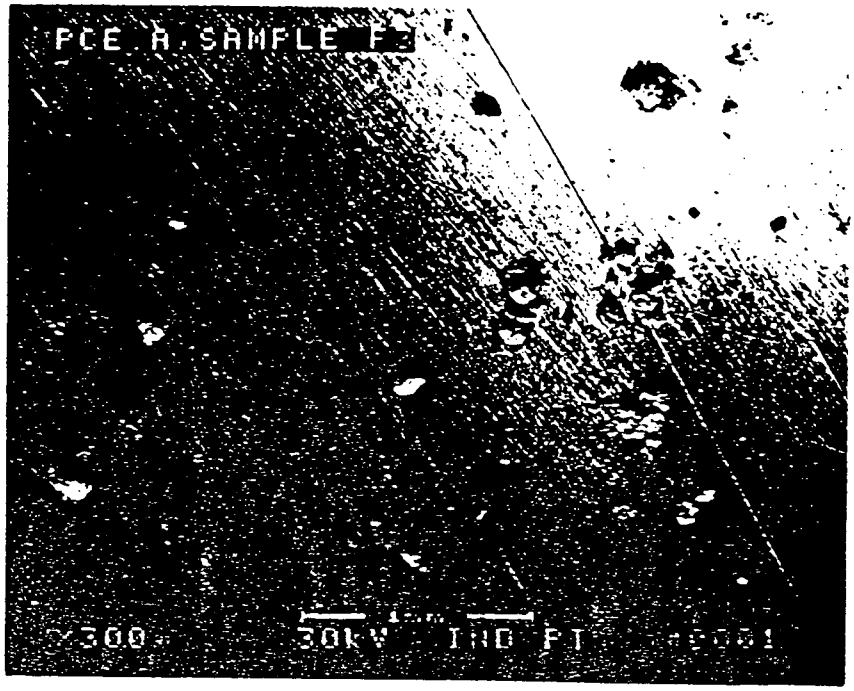
6X

Surface Examination Results of Boat — A, (SG.21) and Boat — B, (SG.22)

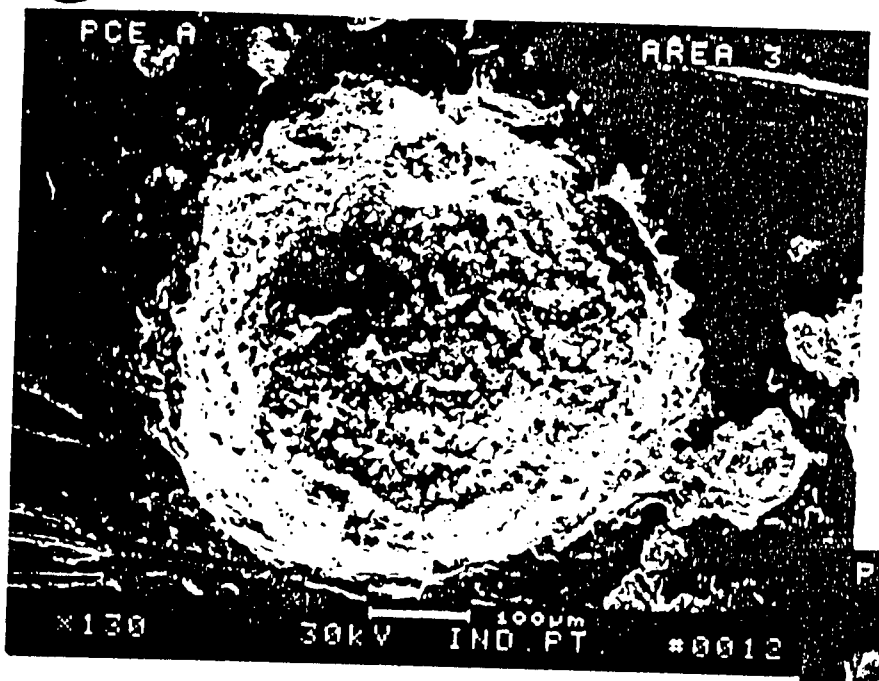


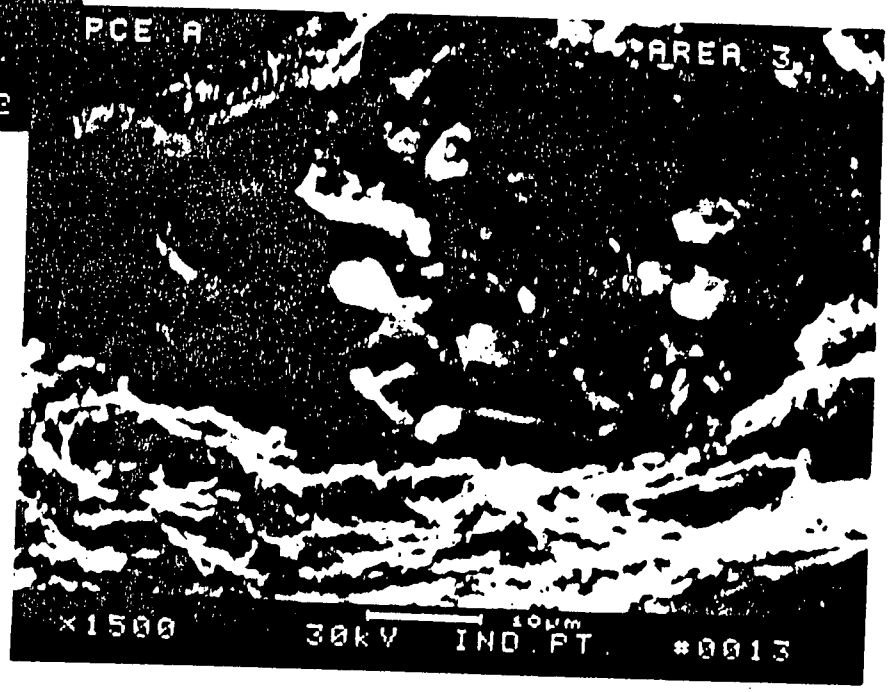


Surface Examination Results of Boat – A, (SEM)

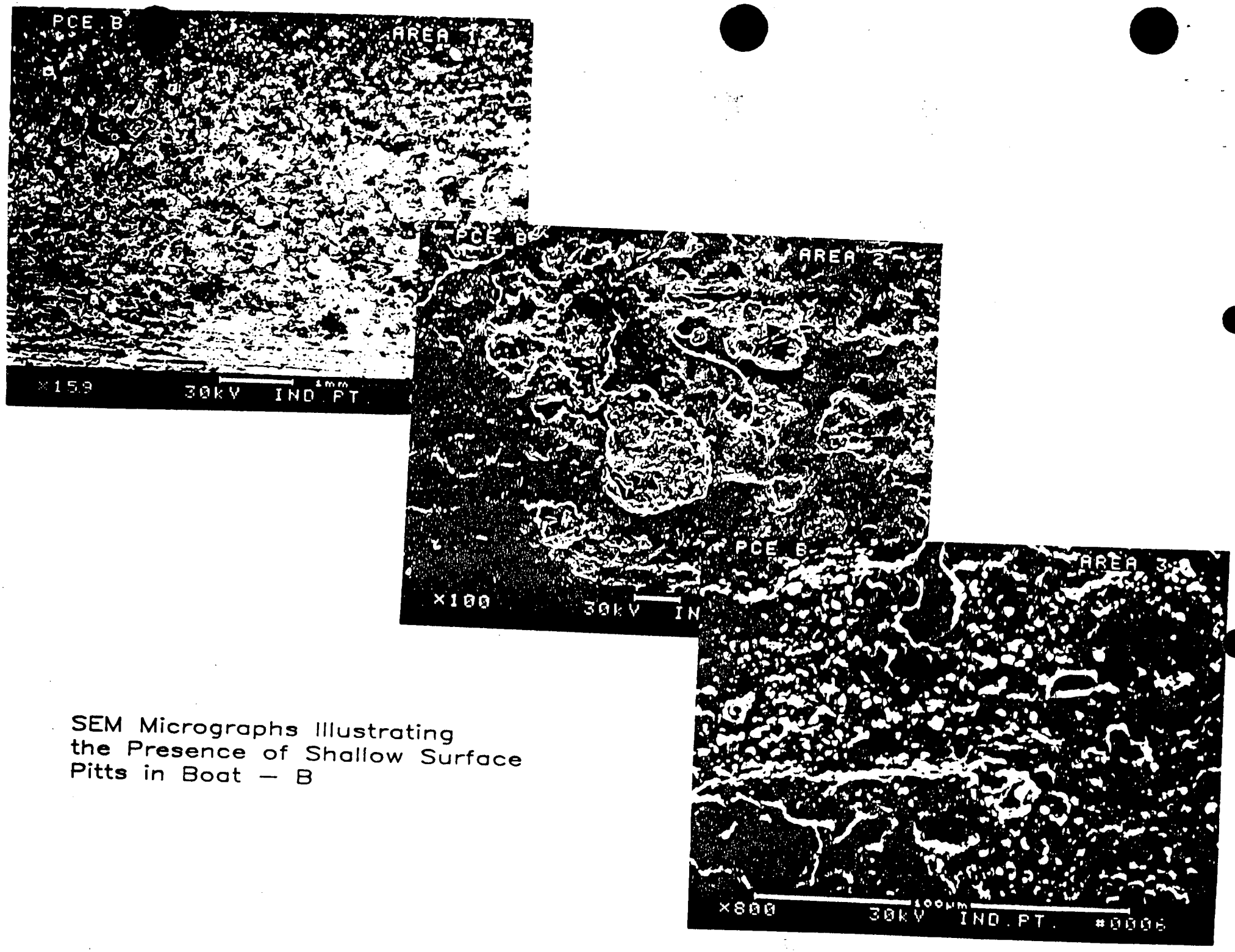


Surface Pitts Seen After Rough Grinding on Boat — A, SG.21 (SEM)

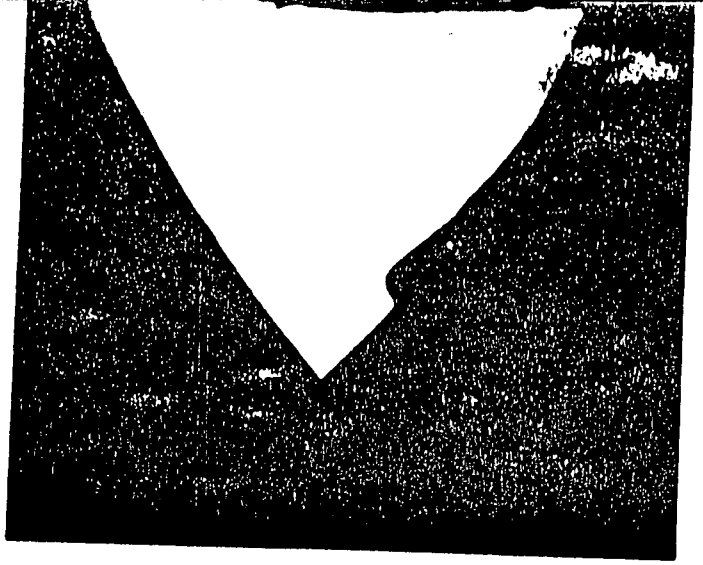


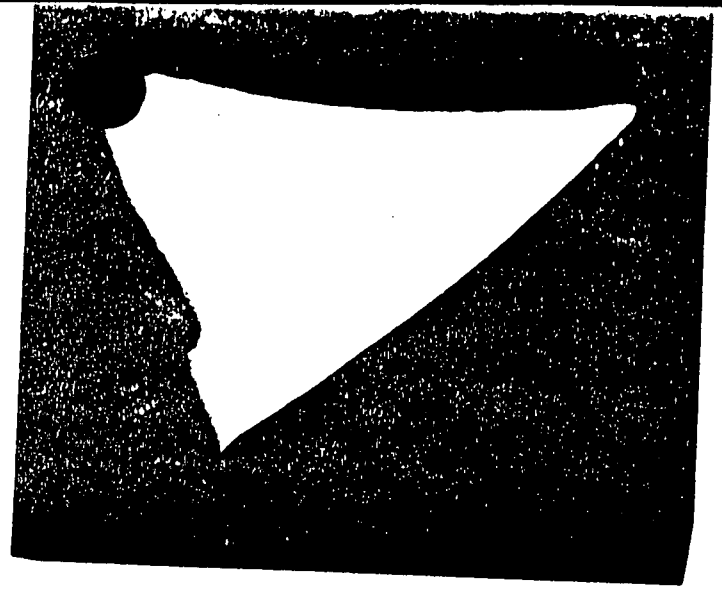


Scanning Electron Micrograph
Illustrating the Morphology
of a Surface Pit Seen
in Boat — A

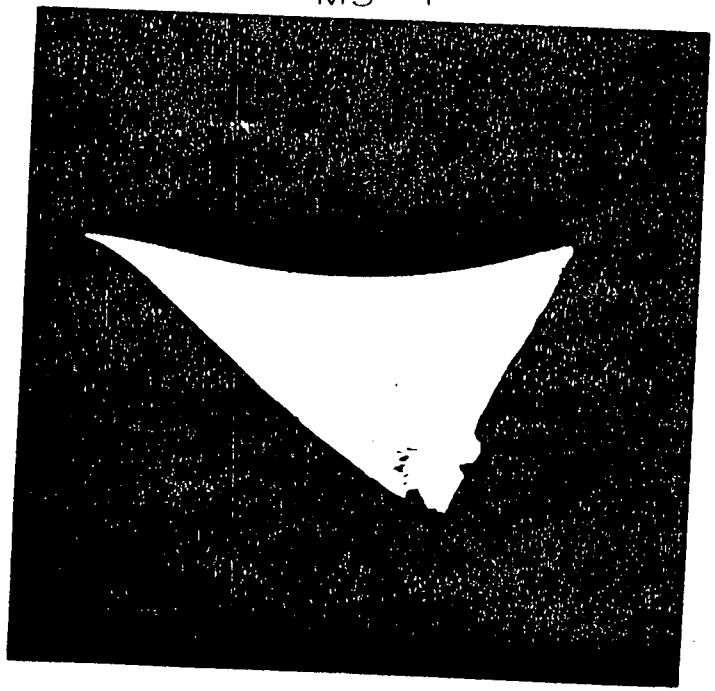


SEM Micrographs Illustrating the Presence of Shallow Surface Pitts in Boat — B

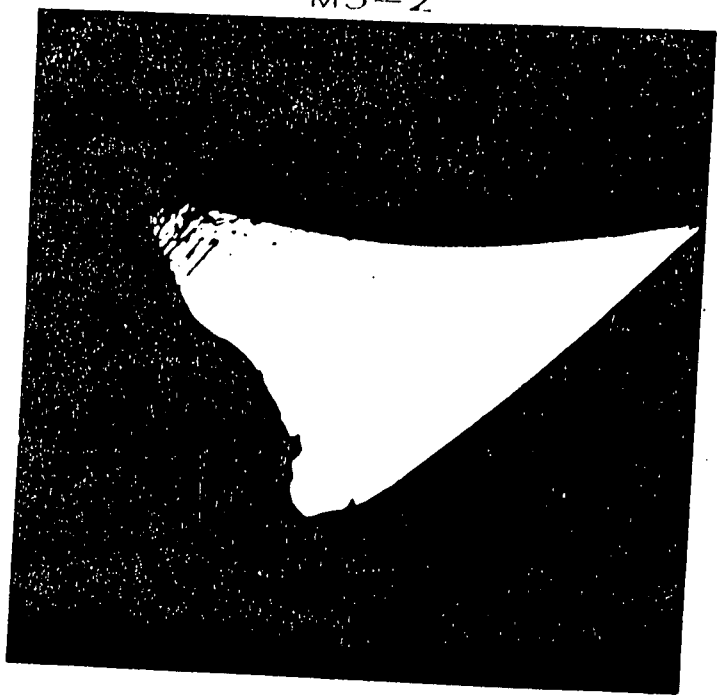




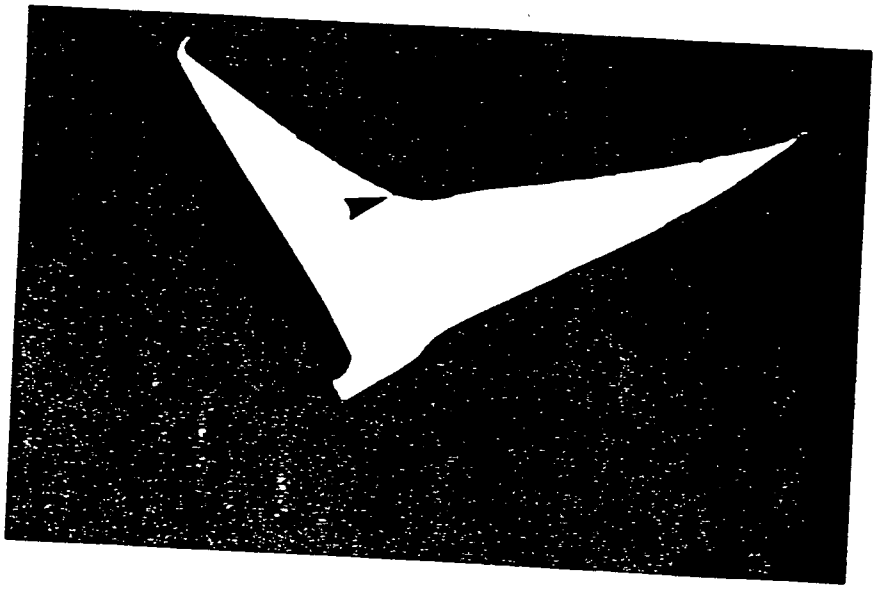
M3 – 1

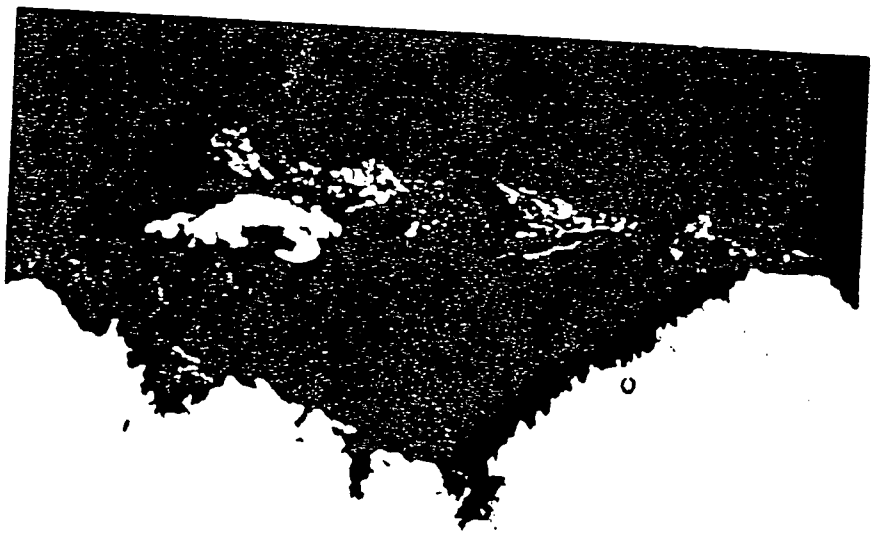


M3 – 2



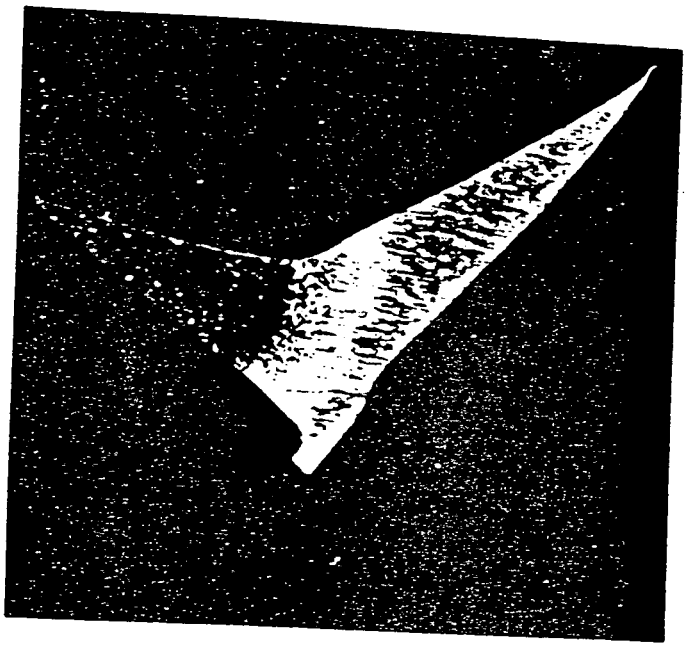
Metallography Results of the Various Transverse Sections of Boat – A (6.25X)





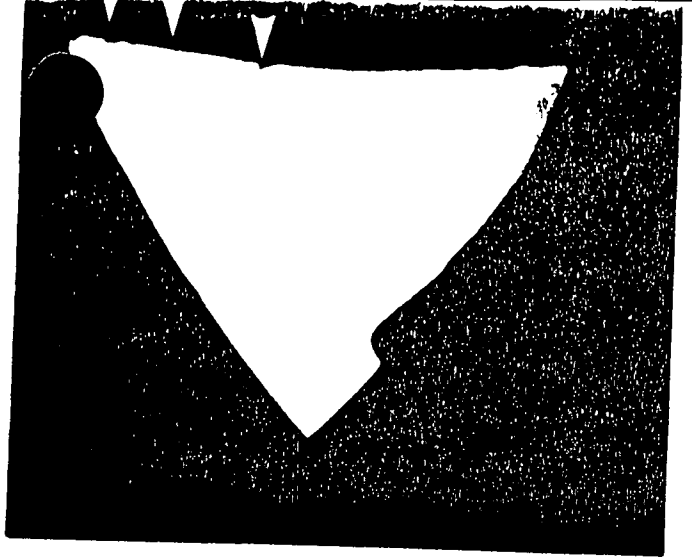
500X

Surface Pit in Boat — B, SG.22

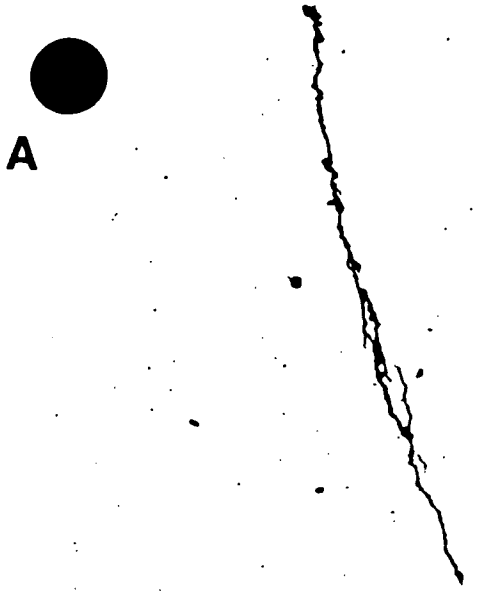


8X

Metallography Results Illustrating the Microstructure seen on a Transverse Section in Boat – B



6.25X



A

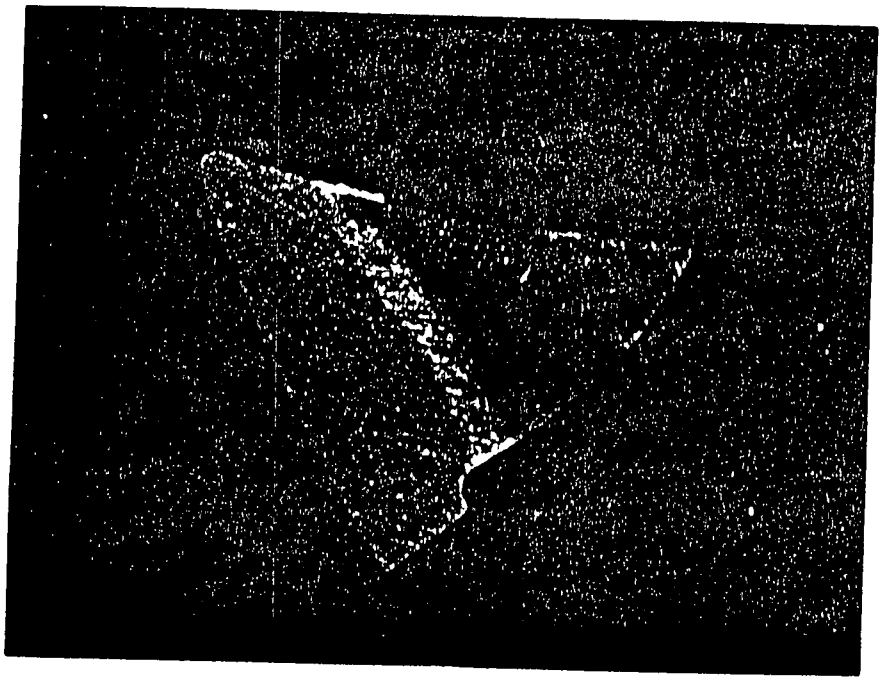
C

200X

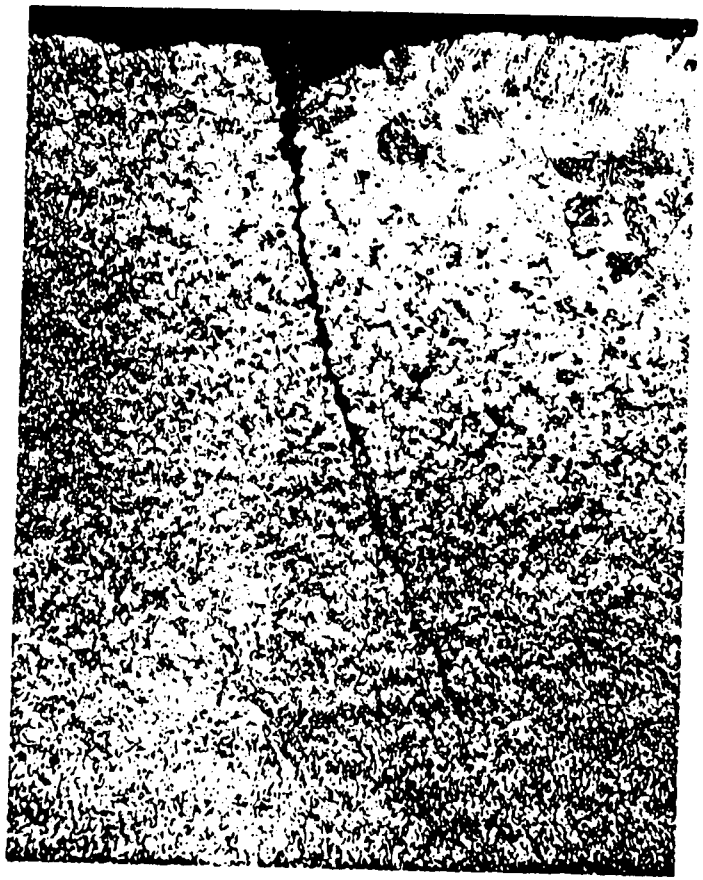
B

200X

Metallography Results Illustrating the Morphology of Cracking and Pits in Boat Sample — A

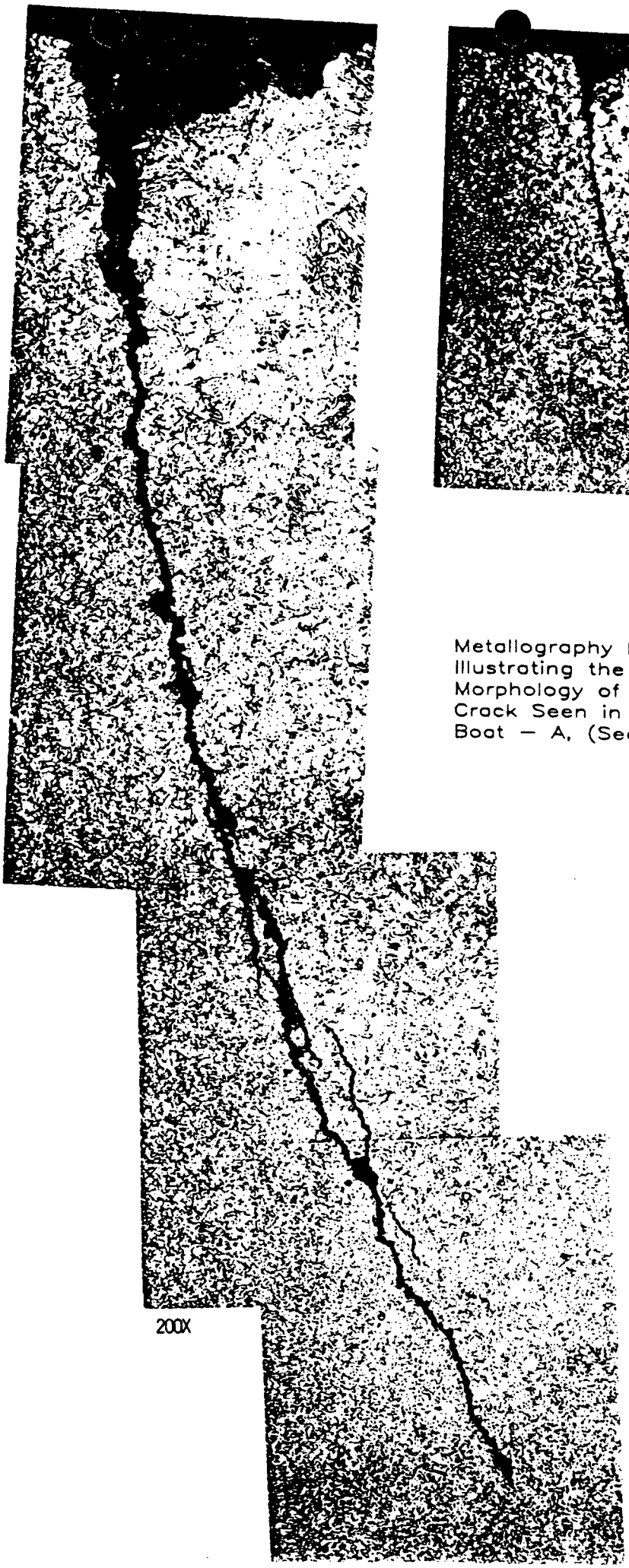


6.25X



50X

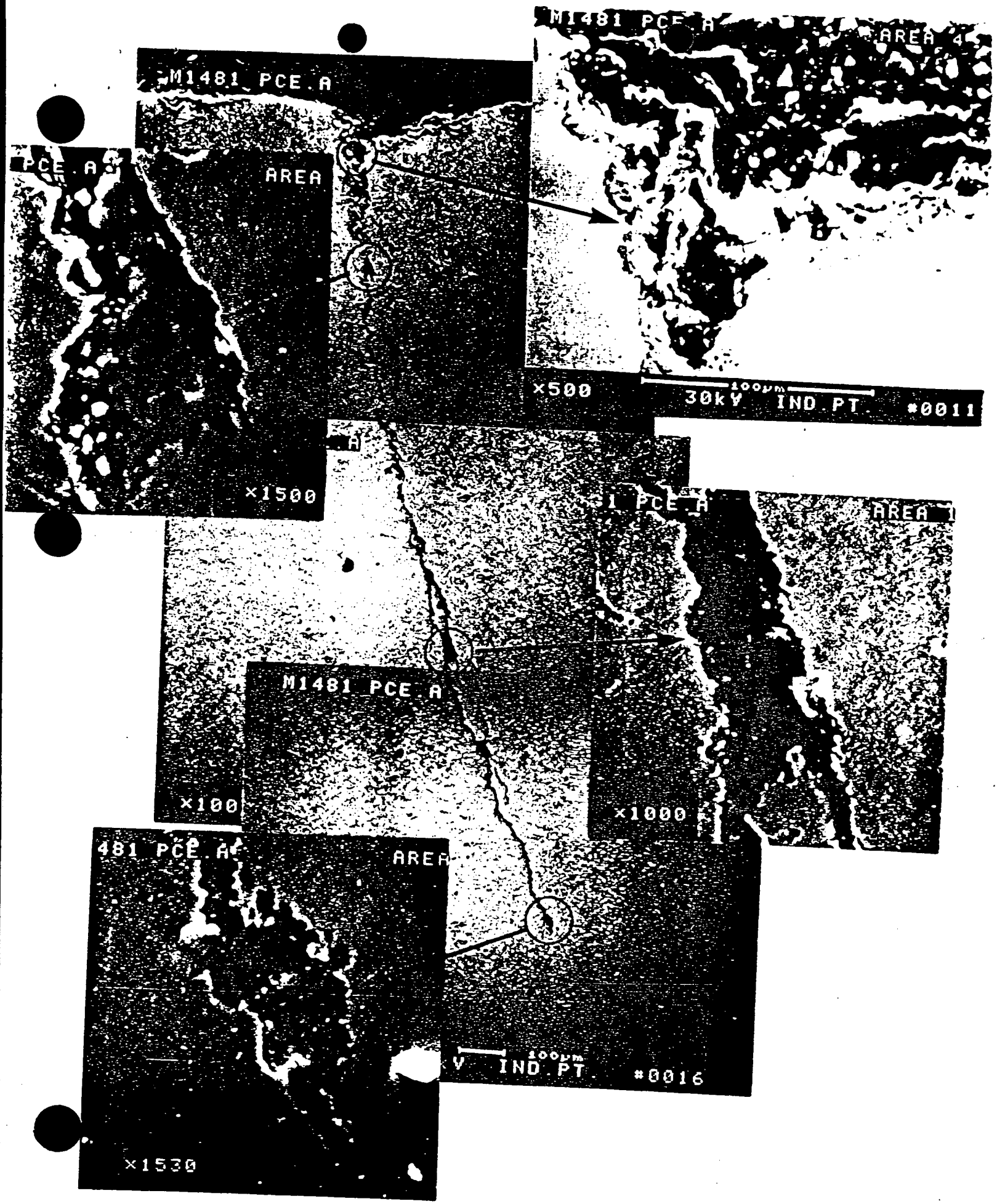
Metallography Results Illustrating the Morphology of the Crack Seen on Section M1 in Boat — A



200X

50X

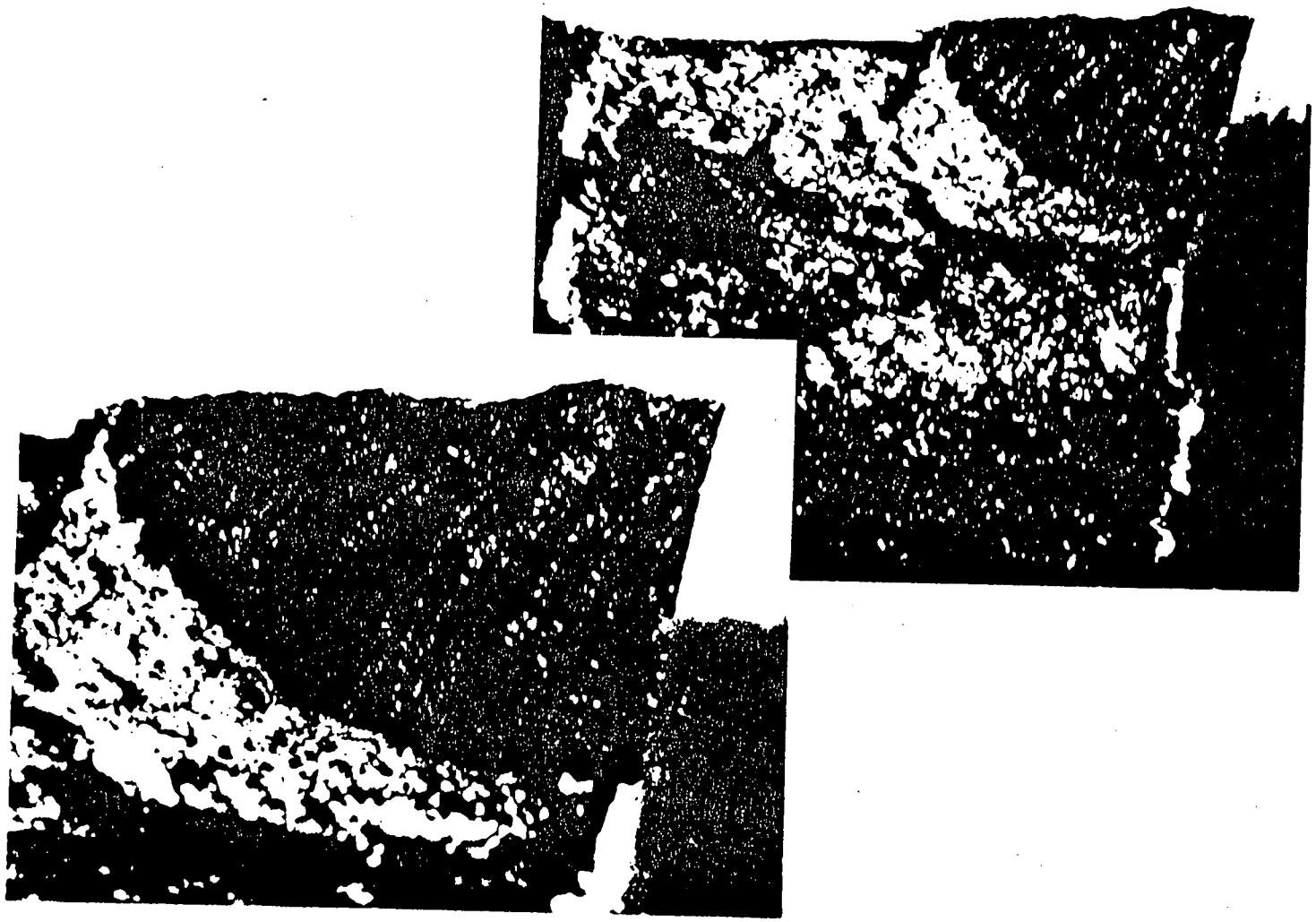
Metallography Results Illustrating the Morphology of the Crack Seen in Boat – A, (Section m1)



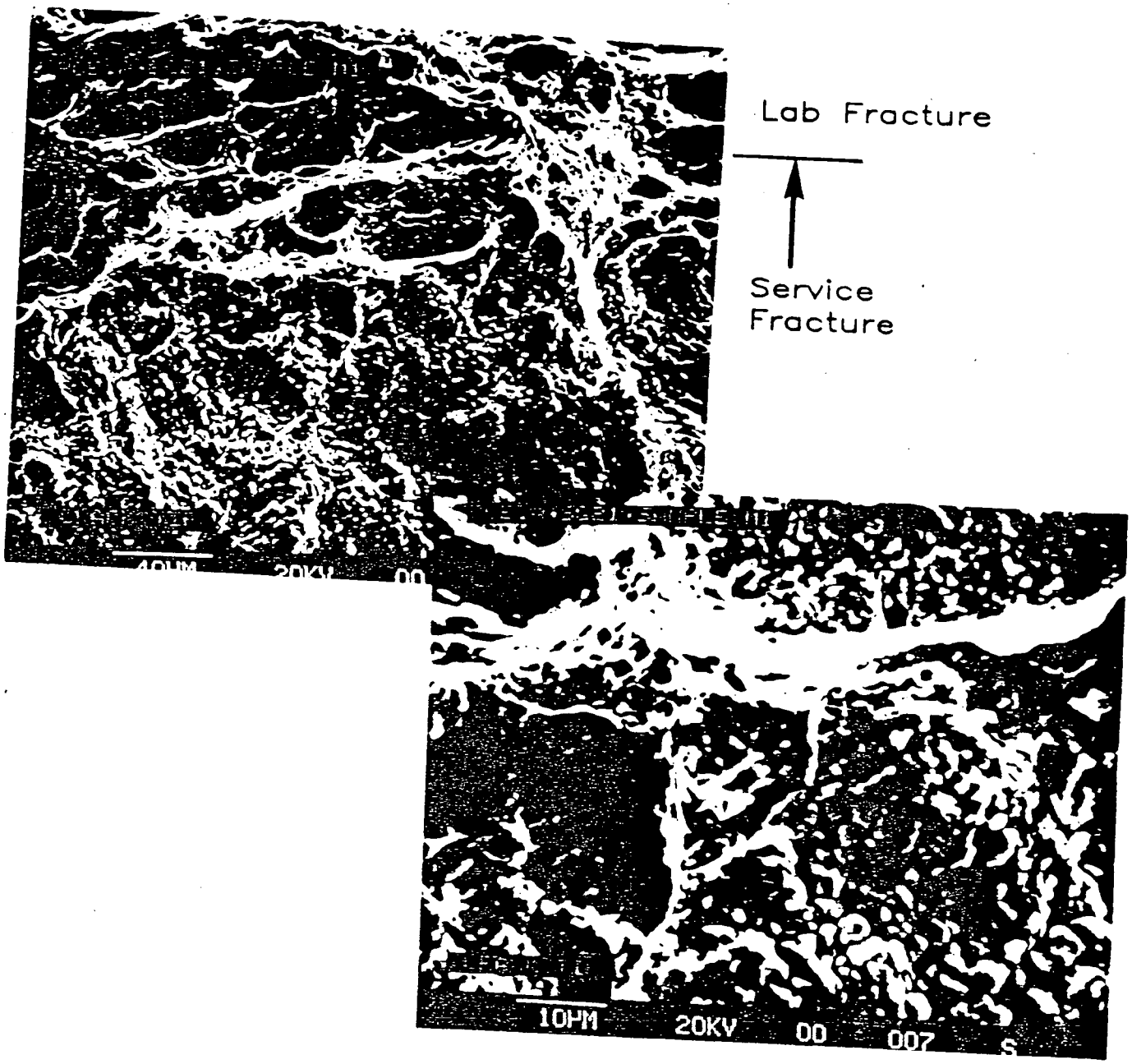
Scanning Electron Fractography Results Illustrating the Morphology of the Crack and Deposits Seen in Boat — A

16X

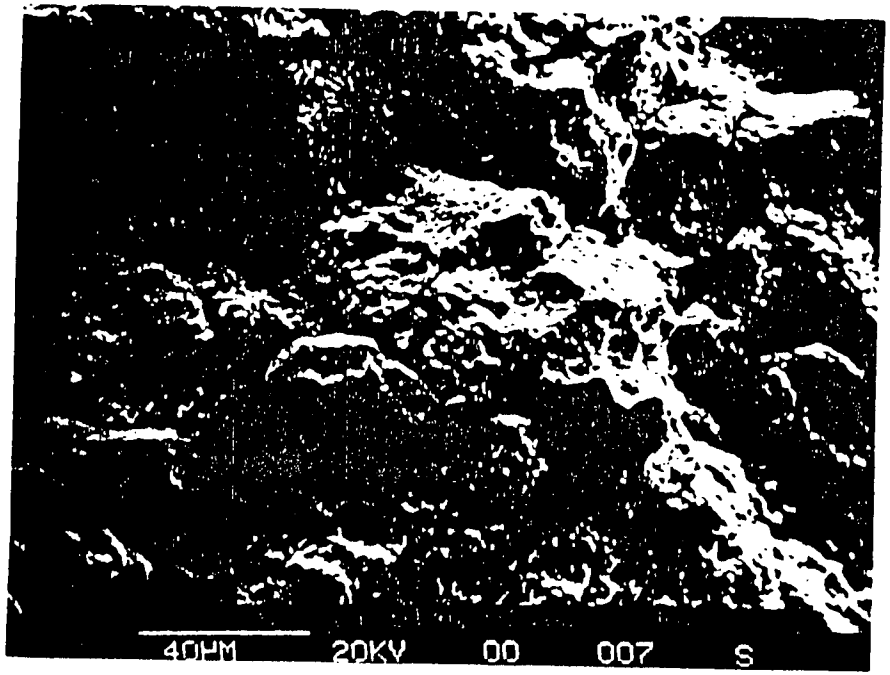
Light Optical Fractograph of a Freshly Opened Crack in Boat — A (Sample M1)

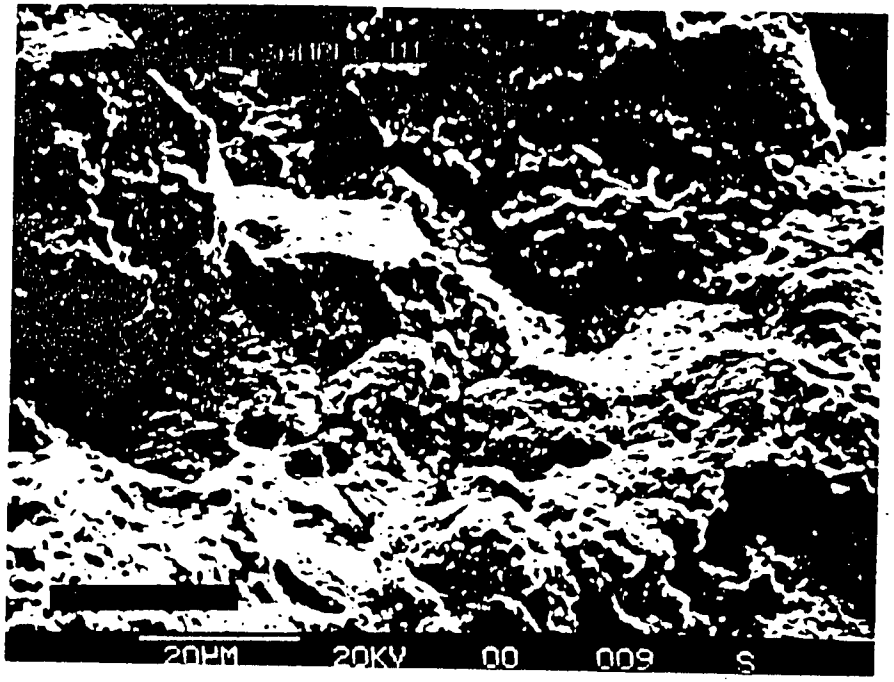


Light Optical Fractograph of a Freshly Opened Crack in Boat — A (Sample M1—1)

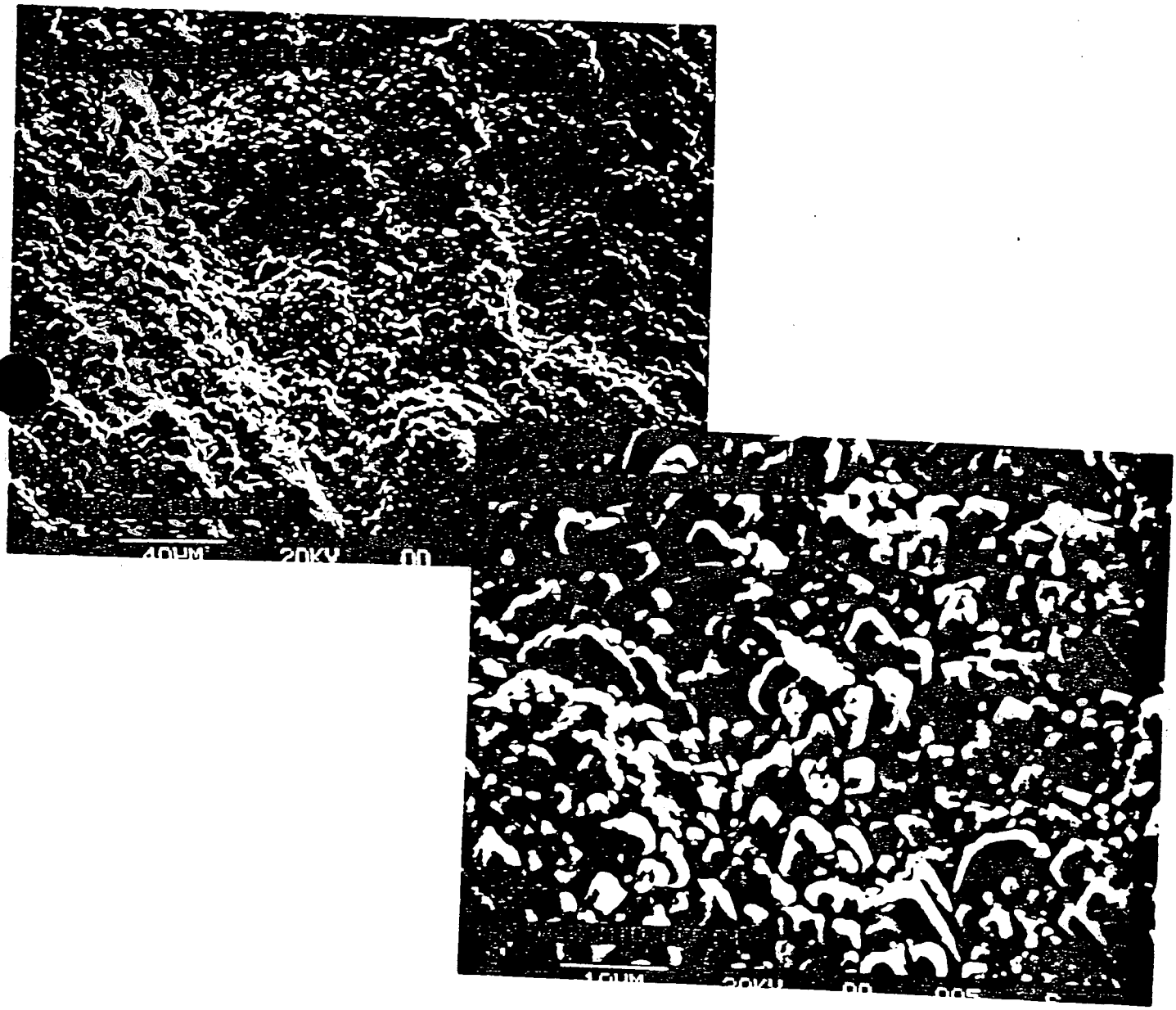


SEM Fractograph Illustrating the Fracture Morphology Seen at the Crack–Tip Region of a Freshly Open Crack in Boat – A

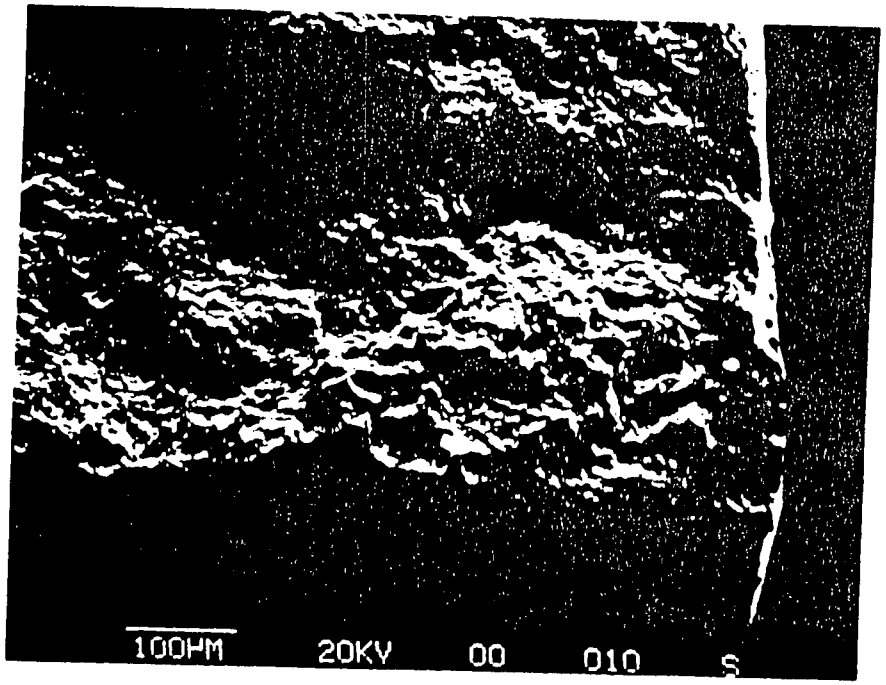




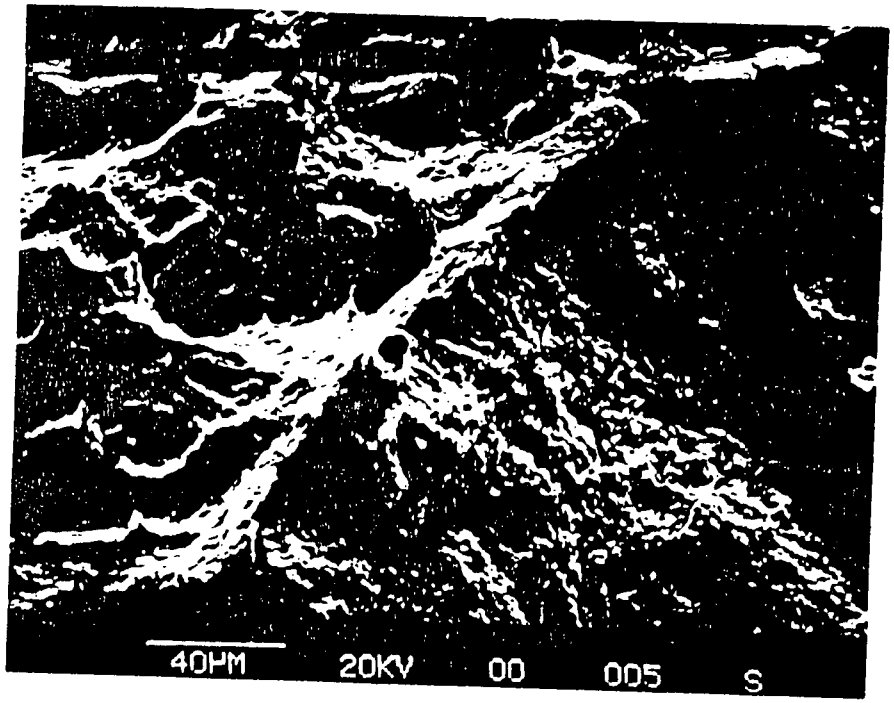
Fracture Morphology of a Freshly Opened and Endoxed Fracture Face of a Crack in Boat — A (Sample M1)



SEM Fractograph Illustrating the Morphology of the Oxide Deposits Seen at Mid—Crack Depth on a Freshly Opened Crack in M1



CRACK MOUTH REGION



CRACK-TIP REGION

Scanning Electron Fractographs Illustrating the Morphology Crack Mouth and Crack—Tip Regions of an Endoxed Fracture Face in Boat — A

SUMMARY OF HARDNESS TRAVERSE RESULTS

BOAT-A, X-M1 SG21

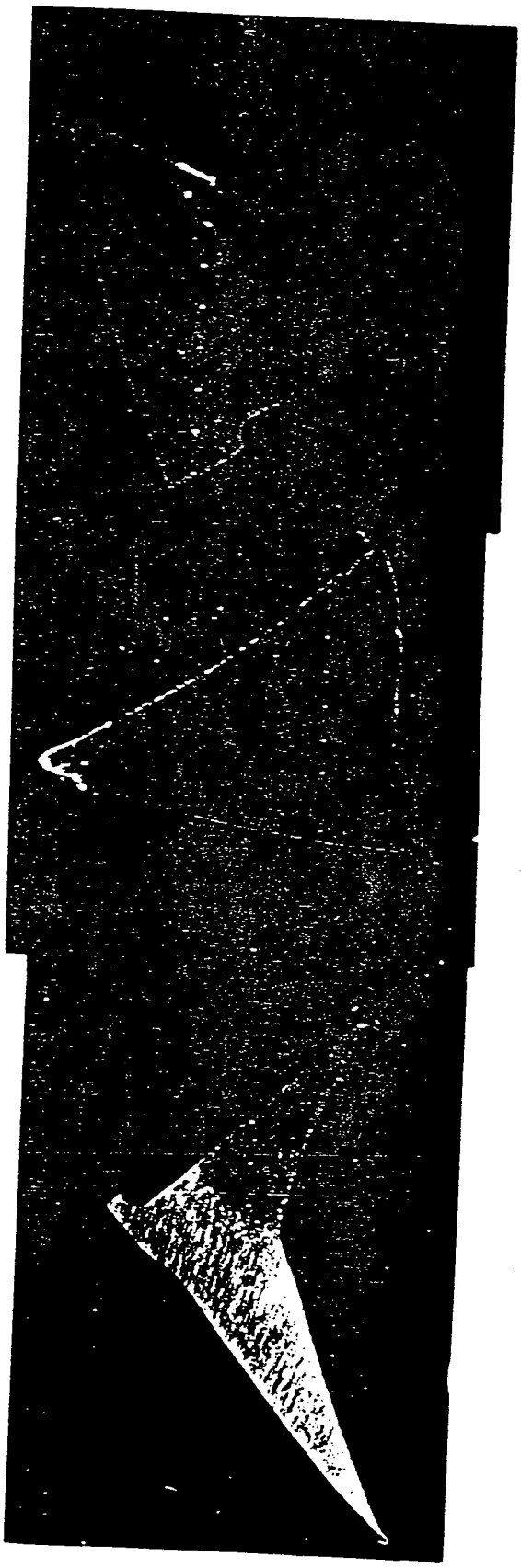
S. NO.	KNOOP HARDNESS	ROCKWELL EQUIVALENT	LOCATION
1	354	35 R_C	HAZ
2	317	31 R_C	HAZ
3	240	20 R_C	HAZ
4	296	28 R_C	HAZ
5	221	94 R_B	BASE
6	221	94 R_B	BASE
7	216	93 R_B	BASE
8	214	93 R_B	BASE

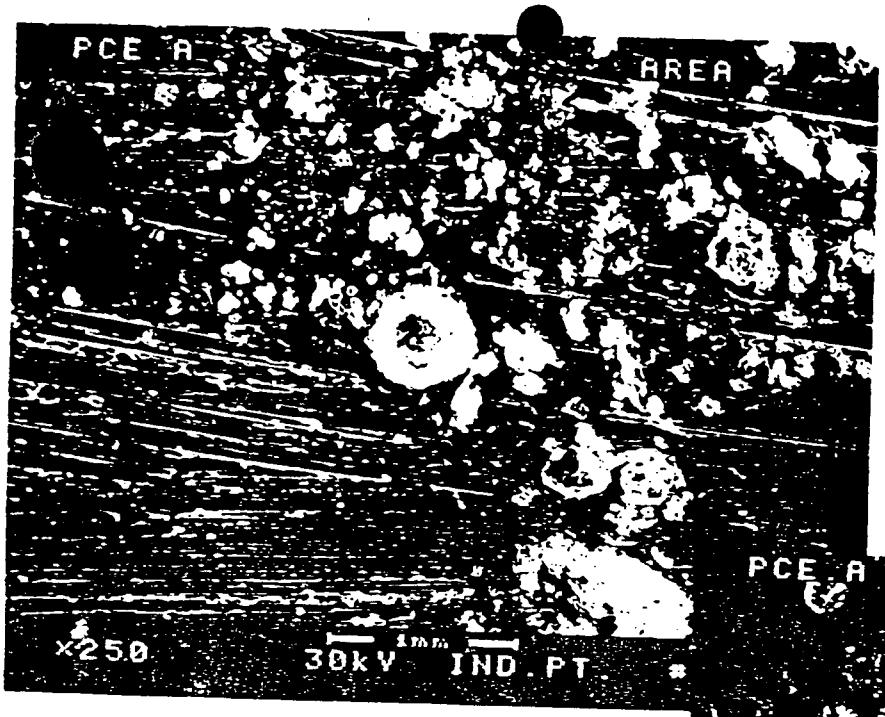
BOAT-A X-M2 SG21

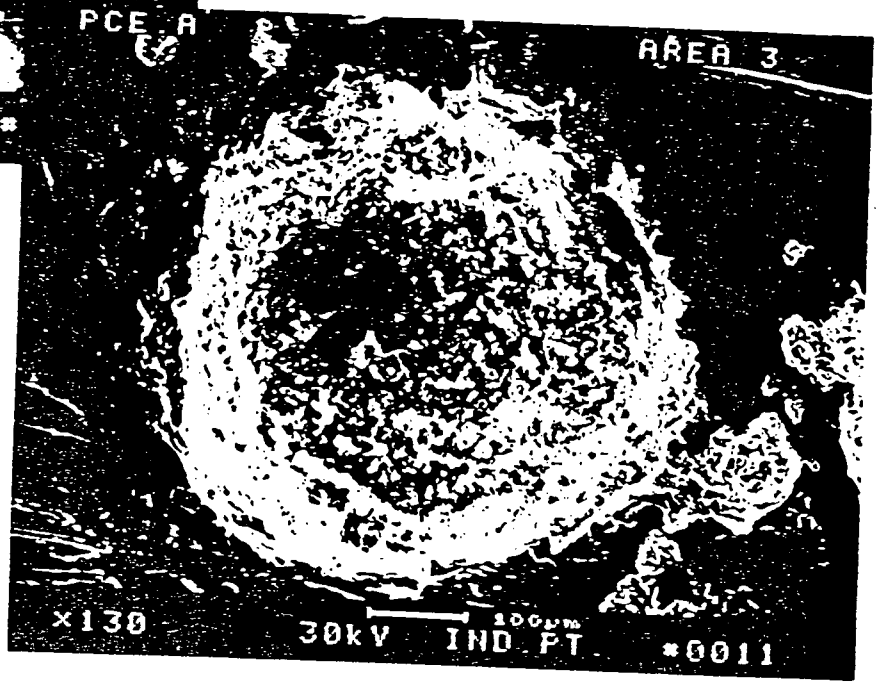
S. NO.	KNOOP HARDNESS	ROCKWELL EQUIVALENT	LOCATION
1	296	28 R_C	WELD
2	304	29 R_C	WELD
3	367	37 R_C	HAZ
4	277	25 R_C	HAZ
5	209	91 R_B	BASE
6	197	89 R_B	BASE
7	209	91 R_B	BASE

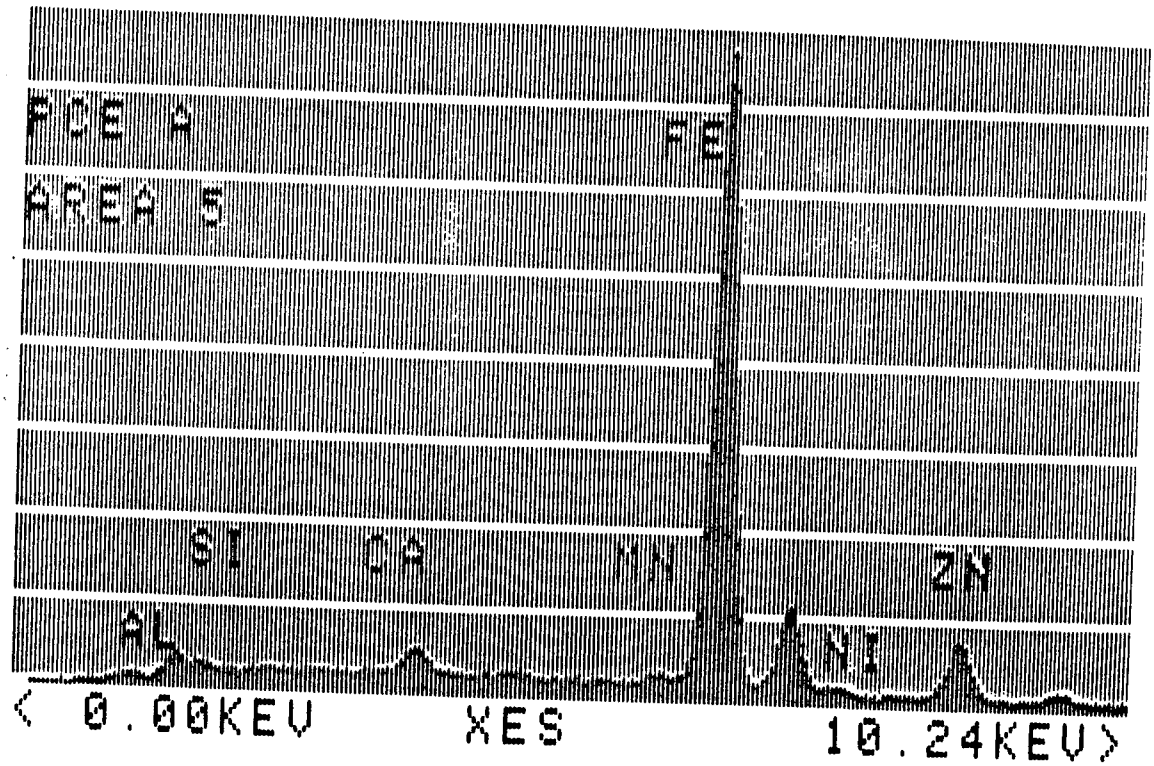
BOAT-B MID-SECTION SG 22

S. NO.	KNOOP HARDNESS	ROCKWELL EQUIVALENT	LOCATION
1	281	25 R_C	WELD
2	296	28 R_C	WELD
3	312	31 R_C	HAZ
4	254	23 R_C	HAZ
5	200	90 R_B	BASE
6	208	91 R_B	BASE
7	198	89 R_B	BASE

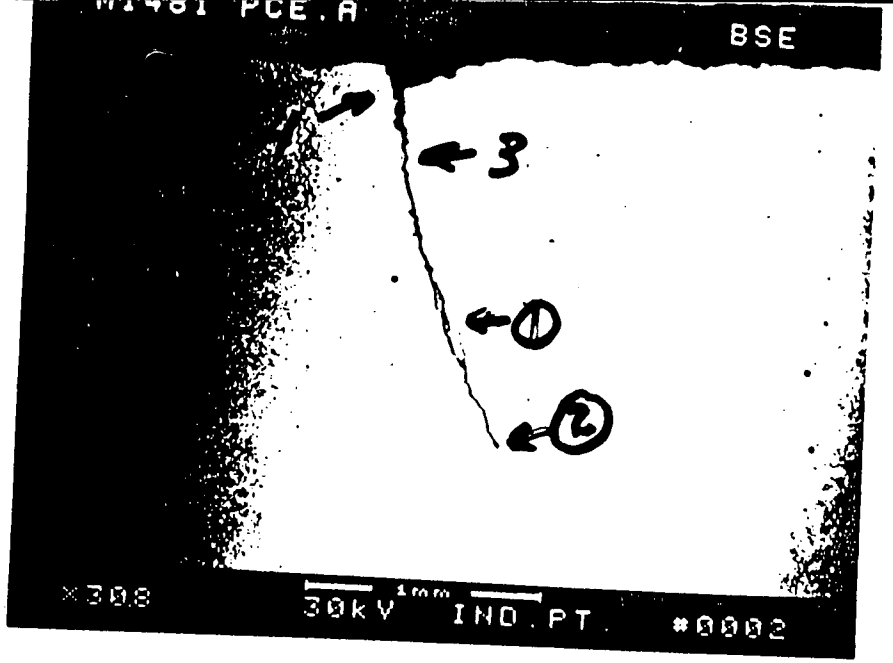


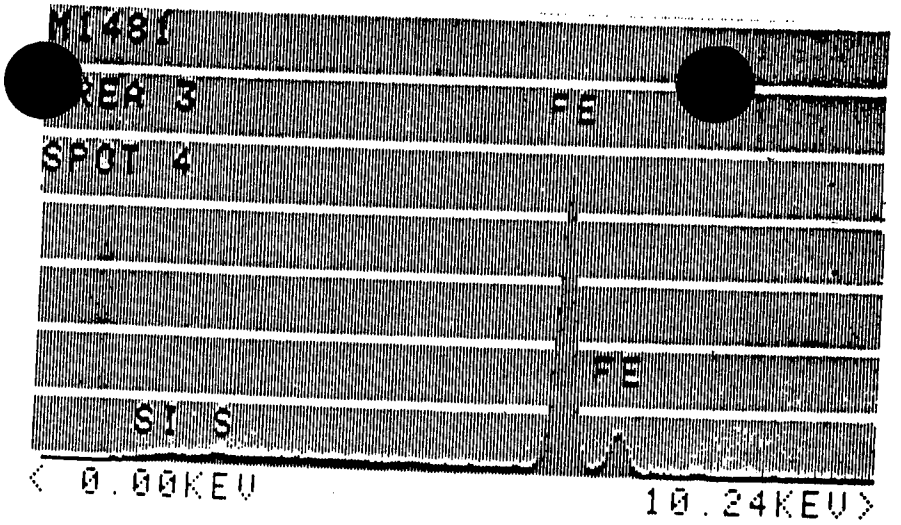


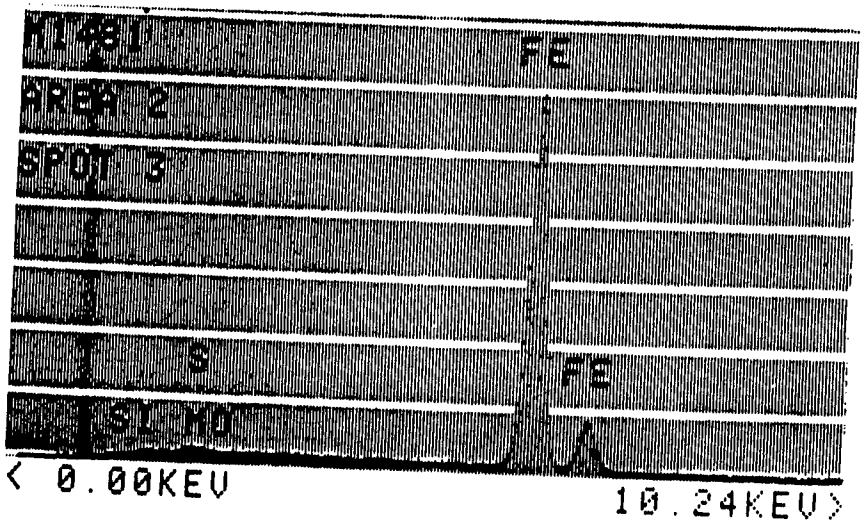


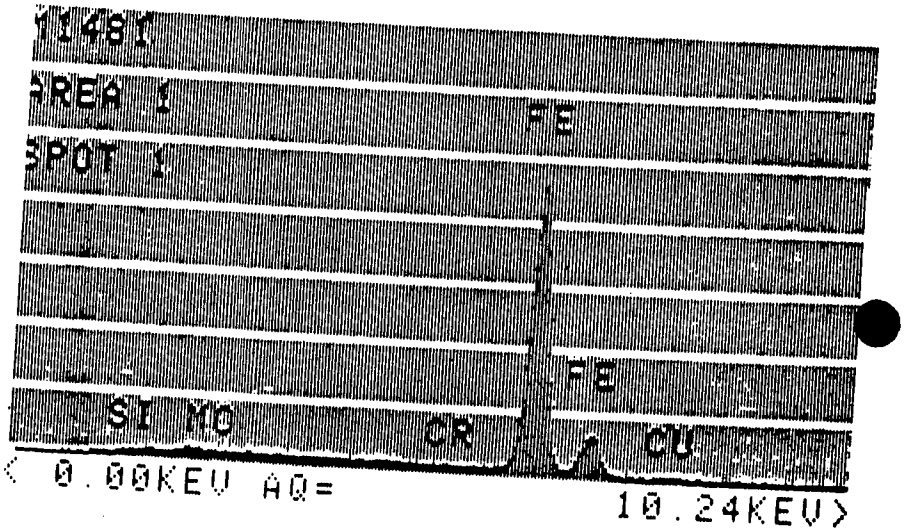


SEM Topography and EDS Chemistry Evaluations









EDS Chemistry Results of the Crack Deposits in Boat — A, Section M1

MicroProbe Analysis of Weld and Base Materials

Weight Percent Boat Sample Chemistry

Element	Stock Weld Wire	Weld Boat Sample A	New Weld Boat Sample B	Base Metal Boat Sample A	Old Weld Boat Sample B
SI	.644	.416	.473	.264	.265
CR	.008	.061	.053	.110	.089
MO	.443	.061	.444	.413	.590
MN	1.759	1.244	1.617	1.345	1.276
FE	96.364	96.113	95.422	96.512	95.747
NI	0.090	1.236	0.085	0.127	0.160
CU	0.268	0.292	0.328	0.352	0.459

* No Appreciable Differences in Weld Metal Chemistry

CONCLUSIONS

- SURFACE PITTING CAUSED BY THE PRESENCE OF OXYGEN, COPPER, AND LOCAL STRESSES CONTRIBUTED TO CRACK INITIATION.
- CRACK INITIATION BEHAVIOR IN THE PREVIOUS AND CURRENT BOATS APPEAR TO BE CONSISTANT WITH STRESS CORROSION MECHANISM
- TOTAL CRACK GROWTH APPEARS TO BE CUMULATIVE OF ENVIRONMENTALLY ASSISTED CRACK GROWTHS UNDER STATIC AND FLUCTUATING LOADS

FUTURE PLANS

1. DETERMINE BEST METHODS TO MINIMIZE OXYGEN.
2. CONTINUE MONITORING CONDITION OF CONDENSERS TO DETERMINE REMAINING LIFE WITHOUT DETRIMENT TO STEAM GENERATOR LIFE. CONDUCT CORROSION PRODUCT TRANSPORT SURVEY.

1991 REFUELING OUTAGE

1. EXAMINE 1/3 OF THE CIRCUMFERENCE OF FOUR STEAM GENERATORS GIRTH WELDS
2. INCLUDE ALL WELDED AREAS IN 22 GIRTH WELD
3. IF INDICATIONS ARE FOUND, THEN GO TO 100%
4. EXAMINE NOZZLE AREAS
5. REPAIR AS NEEDED
6. EVALUATE WELDED AREAS TO DETERMINE ARRESTING VALUE.
7. REPLACE ONE CONDENSER WITH TITANIUM TUBES

ENCLOSURE 3

Westinghouse Electric Corporation

Energy Systems

Nuclear and Advanced Technology Division

Box 355
Pittsburgh Pennsylvania 15230-0355

March 14, 1990
NS-NRC-90-3497

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Attention: Mr. Donald S. Brinkman
Senior Project Manager (IP2)
Project Directorate I-1
Division of Reactor Projects I/II

Subject: Indian Point Unit No. 2 Steam Generator Girth Weld Presentation Material

Dear Mr. Brinkman:

Enclosed is one copy of presentation materials entitled "Indian Point Unit No. 2 Steam Generator Girth Weld" (Proprietary) which were used during the March 14, 1990 meeting at the NRC offices in Rockville, Maryland. This information provides a current description of the program addressing the girth weld issue at Indian Point Unit No. 2.

The enclosed information contains information which is proprietary to Westinghouse Electric Corporation. Accordingly, we request that this information be withheld from public disclosure.

We will comply with the requirements of 10 CFR 2.790 to provide proprietary and non-proprietary versions of the above material together with an affidavit as soon as the proprietary information contained in the submittal has been specifically identified and the proprietary and non-proprietary versions have been prepared. We will submit the required number of copies of the proprietary and non-proprietary versions of the information and the required affidavit at that time.

In the meantime, we have provided a copy for your information and use. K. Holzle, Esq., of the NRC Office of the General Counsel, has advised Westinghouse that he concurs with this procedure.

We expect to be able to fully comply with the requirements for the proprietary and non-proprietary versions of the information and an accompanying affidavit within four weeks.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

W. J. Johnson, Manager
Nuclear Safety Department

cc: K. Holzle, Esq., Office of the General Counsel, NRC

R. A. Wiesemann, Manager, Regulatory & Legislative Affairs
Westinghouse NATD
P.O. Box 355
Pittsburgh, PA 15230

A. T. Paterson, EC E4-09

10 CFR 50.59(b) REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED IN 1989

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT 2
Docket No. 50-247
March, 1990

Preface

Discussed herein are summary descriptions of changes, tests, and experiments performed at Indian Point Unit 2 and completed in 1989. The changes presented herein have been evaluated and determined to meet the following criteria as established by 10 CFR 50.59(b):

Criteria

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

3. The margin of safety as defined in the basis for any Technical Specification has not been reduced.

It has, therefore, been concluded that none of these changes represents an unreviewed safety question.

TABLE OF CONTENTS

ITEM	TITLE	PAGE
1.	Organizational Changes	1
2.	Cycle 9/10 Refueling - Cycle 10 Operation	1
3.	Steam Generator Repairs and Modifications	2
4.	Steam Generator Tube Plug Removal and Replacement	3
5.	Replace Pressurizer Insulation	3
6.	Central Control Room Modifications Per Regulatory Guide 1.97	4
7.	Central Control Room Modifications to Enhance Operability	4
8.	Unit 1 Annunciator and Alarm Replacements	5
9.	Replace Meteorological Display Panels	5
10.	Gas Turbine Control Modifications	5
11.	Addition of Indicating Lights to Monitor Fused Circuits	6
12.	Replace Battery Charger and Battery 22	6
13.	Supply Power to DC Auxiliary Pumps from Unit 1 Batteries	6
14.	Emergency Diesel Generator Building Ventilation Controls Enhancement	7
15.	Changes to Manage Emergency Diesel Generator Loading	7
16.	Install Alternate Feed for Lighting Transformer 23	8
17.	Installation of Time Delay Relays for Low Flow Feedwater Valves	8
18.	Add Fuses to 480 Volt Bus Voltmeter Circuits	9
19.	Installation of Current Limiters on Non-IE Cables	9
20.	Provide Backup Nitrogen to Boration Valve	9

ITEM	TITLE	PAGE
21.	Compressed Air Backup	9
22.	Emergency Diesel Generator Cooling Setpoint Changes	10
23.	Install Temperature Monitors on Pressurizer Auxiliary Spray Line	10
24.	Install a Spare Flow Transmitter for the Plant Vent Flow Monitor	10
25.	Improve Flow Control for Steam Generator Blowdown Sample	11
26.	Reroute Cables for Health Physics Computer	11
27.	Replace Residual Heat Removal Heat Exchanger Outlet Temperature Sensor	11
28.	Install Isolation Valves for Steam Heating Coil for Refueling Water Storage Tank	11
29.	Emergency Diesel Generator Starting Air Supply Improvements	12
30.	Additional Radiation Monitor	12
31.	Central Control Room Air Conditioning Modifications	12
32.	Provide Redundant Signals to Close Steam Generator Blowdown Isolation Valves	12
33.	Nitrogen Backup for Steam Supply Valves to Auxiliary Feedwater Pump	13
34.	Primary Auxiliary Building Ventilation Improvements	13
35.	Install Position Indication for Sampling Valves	13
36.	Install Isolation Valves and Decontaminate the Regenerative Heat Exchanger	13
37.	Provide Shutoff for Lube Oil Heaters of Emergency Diesel Generators	14
38.	Provide Automatic Makeup to Instrument Air Closed Cooling System	14

ITEM	TITLE	PAGE
39.	Installation of Flowmeters to Facilitate Inservice Testing	14
40.	Annunciation System for Heat Trace and Deluge Alarms	15
41.	Add Polymer to Liquid Wastes	15
42.	Provide for Operation of RHR/LPI Valves from Control Room	15
43.	Removal from Service of the Gross Failed Fueled Detector	15
44.	Replace Battery Room Exhaust Fan	16
45.	Rewire Bypass Breakers of Reactor Trip System	16
46.	Install Additional Residual Heat Removal Flow Monitor and Perform Flow Tests	16
47.	Install Reactor Coolant System (RCS) Draindown Level Monitoring System	16
48.	Installation of DORIS Computer	17
49.	Delay Generator Trip on Safety Injection and High Steam Generator Level	17
50.	Install Back Draft Damper in Cable Spreading Room	17
51.	Installation of Helicoflex O-Ring to Seal Reactor Vessel Closure Head	18
52.	Generic Replacement of Valves and Blind Flanges	18
53.	Improve Toxic Gas Monitoring System	18
54.	Supply Cooling Water to Secondary Plant Components from Non-essential and River Water Headers	18
55.	Replace Transmitters Monitoring Service Water Flow to Containment Fan Cooler Units	19
56.	Modify Instrument Connection on Service Water Header	19
57.	Change Power Sources for Service Water Strainers	19

ITEM	TITLE	PAGE
58.	Replace and Relocate Cables in Manhole 21	19
59.	Install Drain from Service Water Strainer Pit	20
60.	Shielding Support Holes in Reactor Head Lifting Rig	20
61.	Generic Replacement of Copper Instrument Tubing With Stainless Steel Tubing	20
62.	Generic Pump Seal Replacement	20
63.	Generic Piping Restraint Modifications	21
64.	Generic Solenoid Valve Replacement	21
65.	Radiation Monitoring Betterment Program	21
66.	Replace Instrument Valve in Reactor Coolant System	22
67.	Defeat Turbine Supervisory Instrumentation Trip	22
68.	Containment of Leakage in Service Water Return Line	22
69.	Use of Sealant to Repair Packing Leak in Pressurizer Spray Valve	22
70.	Install Resistor in Control Circuit of Emergency Diesel Generator (EDG)	23
71.	Replacement of RWST Purification Pump	23
72.	Alternate Method to Transfer Refueling Water to the RWST	23
73.	Provide Alternate Power for Battery Chargers	24
74.	Clear Alarm for Fire Main Loop Isolation Valves	24
75.	Temporarily Deactivate Oil Level Alarms for One Reactor Coolant Pump	24
76.	Temporarily Reroute IVSWS Supply Tubing	24
77.	Use of Temporary Sight Gauge for Component Cooling Surge Tank Level	24
78.	Use of Spare Breaker for EDG Auxiliaries	25

ITEM	TITLE	PAGE
79.	Disable Condenser Water Box Level Alarm	25
80.	Provide Temporary Evacuation Alarm in Containment	25
81.	Operate Containment Purge Exhaust Fan Without Supply Fan	25
82.	Use of Fire Protection Water to Clear Unit 1 Screen Channels	25
83.	Upgrade Containment Sump Grating	26
84.	Install Temporary Pressure Relief Fan	26
85.	Generic Replacement of Instrument Valves	26
86.	Supply Power to Construction Trailers	26
87.	Deoxygenate Water in Condensate Storage Tank	26
88.	Secure Two Non-essential Service Water Pumps to Repair Test Valves	27
89.	Supply Power to Telephone Equipment from the Security Emergency Diesel Generators	27
90.	Replace Security TV Cameras	27
91.	Replace Emergency Security Power Cable	28

1. Organizational Changes

Organizational changes have been made to modify administrative control of the facility (both Indian Point Units 1 and 2), including management of safety-related activities. Significant organization changes are described below.

Watch Engineer positions have been created, reporting to the Operations Manager. The Watch Engineers replace the Shift Technical Advisors (STAs) in all capacities and meet the training and qualification requirements of the STAs. They meet technical specification requirements and other commitments regarding the activities of the STAs and are suitably separated from hands-on plant operations so their independent responsibility for the performance of their safety review duties is not inhibited.

The Safety Assessment and Regulatory Affairs groups have been merged into one group. The new group is the Nuclear Safety and Licensing Section, and its manager reports to the Vice-President Nuclear Power.

A new Material Procurement Section has been created to centralize all purchasing, procurement, materials control, and related activities. The manager of the section reports to the General Manager of Nuclear Power Generation.

The Fire Protection, Safety and Security Section reports directly to the Vice-President Nuclear Power.

The Emergency Planning Section reports to the Manager of Administrative Services.

The Maintenance Section and the Projects and Planning Section have been merged under the Maintenance Manager.

The Radiation Protection, Chemistry, and Radwaste groups have been merged under the Radiation Protection Manager.

These changes bring the Watch Engineers (STAs) directly into the operating organization, consolidate groups with closely related functions, and modify administrative controls of plant activities. These organizational changes are permitted by the technical specifications, and have been determined not to decrease the effectiveness of the safety functions provided by the overall plant organization. No unreviewed safety question was involved.

2. Cycle 9/10 Refueling - Cycle 10 Operation

Indian Point Unit 2 was refueled following the end of its ninth cycle of operation. A total of 72 fuel assemblies were removed, 15 from region 10 and 57 from region 9. They were replaced by region 12 fuel with enrichments of 3.6 and 4.2 w/o. The cycle 10 loading consists of 53 region 10 assemblies, 68 region 11 assemblies, and 72 region 12 assemblies.

The region 12 assemblies are Westinghouse optimized fuel assemblies (OFA), while regions 9, 10 and 11 are LOPAR (low parasitic) fuel. The use of OFA fuel at

Indian Point Unit No. 2 was approved by the NRC via License Amendment No. 140 to DPR-26, issued May 19, 1989. OFA have removable top nozzles with 3-leaf holddown springs, seven zircaloy mid-grids, and larger fuel rod plenums than LOPAR fuel. The cycle 10 core has 60 wet annular burnable poison assemblies, four new secondary sources, and three old secondary sources. In Cycle 10 the OFA fuel does not contain thimble plugs. However, a number of the remaining LOPAR fuel assembles still contain thimble plugs.

Technical specification requirements for boric acid storage tank level, accumulator pressure, and accumulator level were revised.

A report entitled "Reload Safety Evaluation, Indian Point Nuclear Plant, Unit 2, Cycle 10" was prepared by Westinghouse Electric Corporation and reviewed and accepted by Con Edison. The report presents an evaluation for cycle 10 which demonstrates that the core reload, including the design changes discussed above, did not adversely affect the safety of the plant. All accidents and malfunctions of equipment important to safety previously evaluated in the UFSAR have been addressed and it was determined that there is no increase in probability of occurrence or consequences of previously analyzed accidents. The enrichment of the OFA fuel is within the limits of the Spent Fuel Pool Analysis.

The margins of safety are not reduced, and no unreviewed safety question was involved in the Cycle 10 reload and operation.

3. Steam Generator Repairs and Modifications

Modifications were performed in conjunction with inspection and repair of the steam generators. These modifications included removal of asbestos insulation and replacement with asbestos-free insulation, removal of the internal downcomer resistance plates, and modification of existing inspection ports. Additionally, visual inspection revealed the existence of foreign objects which could not be removed from the steam generators.

The replacement insulation has equivalent insulating properties and meets original performance requirements for exposure to sodium hydroxide and boric acid. It is seismically installed.

The downcomer resistance plates were initially removed to obtain access for removal of indications observed in the girth weld at the top of the transition cone. The plates were not reinstalled since their permanent removal was determined to improve the thermal hydraulic conditions at the girth weld location during plant operation. Their permanent removal does not affect the structural integrity of the steam generators. The circulation ratio and therefore the flow rate and mass in the tube bundle are increased by approximately one percent. Effects on tube fatigue, steam generator thermal-hydraulics, tube rupture analysis, and LOCA and non-LOCA accident analyses were evaluated. Previous analyses of susceptibility of tubes with unsupported U-bends to vibration-induced fatigue rupture had resulted in recommendations for preventively plugging 6 tubes. These analyses were re-evaluated for removal of the downcomer resistance plates and it was found that no additional tubes required plugging. Thermal-hydraulic evaluations showed operation to be hydrodynamically stable with a

large stability margin. Dynamic loading on the tube bundle in the event of a steam line break would be less without the resistance plates. The analyses of the tube rupture, LOCA and non-LOCA accidents showed that there is no significant increase in their consequences.

Inspection port modifications included enlargement of several of the existing steam generator inspection ports from one inch to two inches, and new closure designs for those ports. They were performed in accordance with requirements of the ASME B&PV Code, Section III, and did not affect the ability of the steam generators to meet design requirements for normal operation, seismic events and accident conditions. A conservative fatigue usage factor reanalysis determined remaining steam generator lifetime.

An evaluation was performed for operation during Cycle 10 with foreign objects in the steam generator secondary side which could not be retrieved or otherwise accounted for during the Cycle 9/10 refueling outage. The evaluation addressed tube integrity considerations, effects on water chemistry, potential for migration out of the steam generator, and impact on safety-related instrument sensing lines. The analysis showed that no significant adverse effects would result from operation during cycle 10.

None of the above steam generator modifications or conditions increased the probability or consequences of an accident evaluated in the UFSAR, created the possibility for an accident of a different type, or decreased the margin of safety as defined in the technical specification bases. No unreviewed safety question was involved.

4. Steam Generator Tube Plug Removal and Replacement

In order to assess the condition of steam generator tube plugs made from the same heat as plugs that caused a large primary to secondary leakage event at another plant, eight plugs were pulled from the hot leg of steam generator 22. The tube surfaces from which the plugs had been removed did not meet finish requirements for direct replacement with the standard mechanical plug. Therefore, the tubes were plugged with a modified mechanical plug which was inserted one-half inch deeper into the tube. To preclude interaction with the transition between the expanded and unexpanded portions of the tube, plugs were modified by removing the top lands (which are above the structural and sealing lands). The replacement plugs were otherwise identical to the standard plug design. There was no change in plug function or structural integrity, and no unreviewed safety question was involved.

5. Replace Pressurizer Insulation

Pressurizer insulation was replaced with non-asbestos materials. The replacement insulation has equivalent insulating properties and meets original performance requirements for exposure to sodium hydroxide and boric acid. It does not weigh more than that being replaced. Previous seismic and stress analyses were not impacted. No unreviewed safety question was involved in this replacement.

6. Central Control Room Modifications Per Regulatory Guide 1.97

To implement Con Edison commitments regarding Regulatory Guide 1.97 Revision 2, several modifications were made to central control room indicators and their channels.

Various indicators, indication channels, and indication isolation devices were upgraded to meet seismic qualification and channelization/separation requirements. The upgraded channels provide information to allow the operators to monitor reactivity control, core cooling, reactor coolant inventory, heat sink and containment integrity. Approximately 17 indicators and 25 isolation devices were removed and replaced.

The core exit thermocouple monitoring system provides information for diagnosing the onset of inadequate core cooling and ascertaining the effectiveness of mitigation actions. It was modified to provide two channels powered from independent class IE sources. Each channel displays core exit temperature, reactor coolant pressure, and subcooling margin, and also provides data to the safety assessment system and the plant's PROTEUS computer.

Current indicators have been provided in the control room for each pressurizer heater group. They enable the operators to detect pressurizer heater malfunctions and take actions to maintain proper pressure. The ammeters and their current transformers are seismically mounted to ensure that they will not affect safety-related systems in a seismic event.

The range of the reactor coolant system cold leg wide range temperature indicators was expanded from 0 to 600° F to 0 to 700° F to improve post-accident monitoring capability. The change required minor set-point changes to maintain the function of the overpressure protection circuit.

The modifications described above enhance the information available to the operators and improve reliability without affecting the functional capability of the channels. Additional electrical loading is minimal and was analyzed for acceptability. Appropriate seismic requirements are met and margins of safety are not reduced. No unreviewed safety question was involved.

7. Central Control Room Modifications to Enhance Operability

Numerous modifications were made to the instruments and controls in the central control room to improve operability and implement human engineering enhancements identified in the detailed control room design review. These betterments included the following:

- Removal of instrumentation and controls which are no longer used.
- Replacement of containment recirculation fans low flow indicating lights with "push-to-test" lights.
- Replacement of recorders which monitor conditions in the condensate, feedwater, main steam and reheater drain systems.
- Replacement of steam generator narrow-range level recorders and installation of wide-range level recorders.

- Provision of additional indicators for charging flow, letdown flow, letdown pressure, volume control tank level, and voltage on the 6900 volt buses.
- Provision of valve position indicating lights for the pressurizer spray valves.
- Modifications to provide increased uniformity of labels, demarcation and scales on meters and recorders to improve readability.
- Relocation of instruments and controls to functionally group them and improve operator recognition and response.

These modifications do not affect the functional capability of affected or interfacing systems. During the physical modification process, monitoring and control capabilities were maintained as required for the plant condition at the time. Power supply loading is within system capabilities. All components are seismically mounted such that a seismic event would not cause them to affect the functioning of any safety-related equipment. Technical specifications and their bases are not involved. These modifications did not involve an unreviewed safety question.

8. Unit 1 Annunciator and Alarm Replacements

A new annunciator panel was installed to provide alarm capability for unit 1 alarms which are still active (many unit 1 alarms have been retired). Fire protection alarms have been relocated to put them in a direct line of sight from the operators. Electric power supplies are from unit 1 power feeds and are not class IE. This modification does not affect any safety related equipment and does not impact any technical specification or basis. No unreviewed safety question was involved.

9. Replace Meteorological Display Panels

New meteorological data display panels and printers were installed in the central control room, the emergency operations facility, and the meteorological building. The new panels provide 15 minute averages for parameters specified in the technical specifications. The control room panel and printer are seismically restrained and do not interface with existing safety-related equipment. No change in the technical specifications or bases, and no unreviewed safety question was involved in this modification.

10. Gas Turbine Control Modifications

Gas turbine generators physically located at the Indian Point plant site are available as an alternate offsite power source. The control and instrumentation circuits of gas turbine GT-1 were modified to provide that:

- GT-1 can be started from the Indian Point 2 control room. In case these controls are disabled they can be isolated and GT-1 controlled locally.
- Load current from the Gas Turbine substation to the Indian Point 2

6900 volt busses can be monitored in the control room.

- Breakers between the Gas Turbine substation and the Indian Point 2 6900 volt busses can be closed without first opening the breakers from the station auxiliary transformer.

These modifications increase the reliability of the offsite power supply and minimize the number of starts of the emergency diesel generator units. They did not involve any technical specification or bases, or an unreviewed safety question.

11. Addition of Indicating Lights to Monitor Fused Circuits

Indicating lights were added to indicate availability of power to the actuation and timing circuits for degraded voltage in the 480 volt busses, the actuation circuits for undervoltage auxiliary relays, and the auxiliary feedwater pump autostart actuation and logic circuits. The power feed for the train "B" relay which actuates the auxiliary boiler feedwater pump was relocated from the line side to the load side of its fuse.

These modifications provide for periodic verification of power to these circuits. Their functions are not affected. The lights and wiring are seismically restrained to assure that there is no potential impact on other safety related components. No unreviewed safety question was involved.

12. Replace Battery Charger and Battery 22

Battery 22 and its charger, which are part of the 125 VDC power system, were replaced. In the replacement, the capacity of the battery was increased from 1200 to 1800 amphours, and the capacity of the charger was increased from 200 to 250 amps. The battery fuse, fusebox and racks were also replaced.

Analyses demonstrated that combustible hydrogen concentrations will not occur in the battery room and that the new components will be operable after a safe shutdown earthquake. There was no change in functional requirements. The modification did not involve a change in any technical specification. No unreviewed safety question was involved in this modification.

13. Supply Power to DC Auxiliary Pumps from Unit 1 Batteries

The power supplies for the main turbine-generator emergency oil pump and the boiler feed pump emergency oil pump were changed from the Unit 2 batteries to the Unit 1 battery panels. These pumps, which are located in the turbine-generator building, are not safety-related.

Unit 1 batteries are not safety-related and have sufficient spare capacity to serve these loads. The Unit 2 batteries are safety related and provide power to safety-related systems. Relieving them of these loads gives them an additional margin of capacity. Safety related equipment and functions were not adversely affected by this modification, and no technical specification or basis

was involved. This modification did not involve an unreviewed safety question.

14. Emergency Diesel Generator Building Ventilation Controls Enhancement

The controls for the emergency diesel generator (EDG) building ventilation system were modified to eliminate the potential for a single failure to disable the ventilation system. The air-operated louvers and dampers were modified to fail open on a single failure of the pneumatic control system, loss of power to either solenoid valve or a single mechanical failure of either solenoid valve. Redundant solenoid valves have been provided to control them. Six thermostats have been provided, one for each of the five fans and one high building temperature thermostat. Four fans are capable of adequately ventilating the building. In addition, a connection was provided to an alternate air supply and a pressure switch was added to actuate both a control room EDG trouble alarm and a low air pressure alarm at the local EDG control panel. A new pressure gauge was also installed in the air supply tubing to the louver and damper actuators to indicate the status of the pneumatic control system.

These modifications increase system reliability. They meet seismic requirements to ensure the system will remain functional after an earthquake. This change did not adversely impact any technical specification or its basis, and no unreviewed safety question was involved.

15. Changes to Manage Emergency Diesel Generator Loading Sequence

In the event of loss of power, the emergency diesel generators start and provide power as required for safe shutdown or accident mitigation. The various loads, which are divided among the three diesel generators, are started in preset sequences. Equipment and procedures have been modified to limit peak loads on the diesel generators.

When service water pumps 21, 22 and 23 are on the non-essential header, the starting sequence is now 23-22-21 (previously 21-22-23). Similarly, when pumps 24, 25, and 26 are on the non-essential header, the sequence is 26-25-24 (previously 24-25-26). At the start of the recirculation phase, if both containment spray pumps are running, pump 22 will continue running and pump 21 will stop. Previously, pump 22 would have stopped and pump 21 would have continued to run. To compensate for load shifts associated with these changes, the motor driven Auxiliary Feedwater pumps will be stopped just prior to initiating the recirculation switching sequence. In addition, the recirculation switching sequence is modified.

The emergency operating procedures (EOPs) have been rewritten so that the emergency diesel generators will not be overloaded under any analyzed circumstances. The procedures address loss of offsite power coincident with a safety injection signal and provide caution statements concerning manually applying loads. Additional modifications have been implemented to support the EOP changes and were separately discussed and evaluated.

The impact of these changes on all applicable FSAR analyses has been evaluated

to ensure acceptability. Evaluations also addressed potential effects on component and system functional capabilities.

The above changes to equipment and procedures limit the loading on the Emergency Diesel Generators and do not increase the probability or consequences of any analyzed accidents. The technical specification bases were preserved and no unreviewed safety question was involved.

16. Install Alternate Feed for Lighting Transformer 23

A manual transfer switch has been installed to allow lighting transformer 23 to receive power from either engineered safeguards bus 5A or from bus 2 in the Technical Support Center (TSC). The transformer serves lights in part of the Primary Auxiliary Building. In the event of loss of offsite power, bus 5A receives power from emergency diesel generator 21 and TSC bus 2 is powered by the TSC diesel-generator.

This modification allows the load on EDG-21 to be reduced. The lighting transformer is not safety related and does not support any security systems. The TSC diesel-generator has adequate capacity to carry the lighting load. The transfer switch is seismically supported. The lighting system does not involve any technical specification or basis and no unreviewed safety question was involved.

17. Installation of Time Delay Relays for Low Flow Feedwater Valves

On a reactor trip coincident with low primary system average temperature, feedwater to the steam generators is shut off by automatic closure of the main feedwater regulating valves and the low flow feedwater regulating valves. The auxiliary feedwater pumps are started to supply feedwater under these conditions. The isolation circuit for the low flow feedwater regulating valves has been modified to delay their closure until steam generator level has been restored to approximately 15 inches above the feedwater nozzle. Continued main feedwater flow (at approximately 416^{0} F) during this period will mix with the auxiliary feedwater flow and reduce the temperature change experienced at the steam generator girth weld. The time delay chosen was also analyzed to ensure that no overcooling of the reactor coolant system and/or generation of a safety injection signal would occur as a result of this continued main feedwater flow.

This modification does not affect the separate safety-grade circuitry which initiates feedwater isolation on high steam generator level or a safety injection signal.

This modification does not create the possibility of an accident or malfunction of a different type than any previously analyzed and does not increase the probability or consequences of any accident previously analyzed. The modification did not affect any technical specification or basis. No unreviewed safety question was involved.

18. Add Fuses to 480 Volt Bus Voltmeter Circuits

A voltmeter is provided in the control room to monitor voltage on the 480 v busses, with a switch to select which bus is monitored. The switch and voltmeter are connected to the potential transformers which supply the bus undervoltage relays. Fuses have been installed at the 480 v busses to isolate the switch and voltmeter in the event of a short circuit or other fault, allowing the undervoltage relays to continue to function normally.

This modification enhances the existing busses by protecting against spurious operation of the undervoltage protection feature in the event of failure of the switch or voltmeter. It did not affect any technical specification or basis and no unreviewed safety question was involved.

19. Installation of Current Limiters on Non-IE Cables

Current limiters have been installed in three non-IE feeder circuits from safety-related busses to provide enhanced fault protection for these non-Class IE cables. The current limiters (which are similar to fuses) clear faults quickly to protect against exceeding the thermal limits of these cables. They provide additional protection beyond that afforded by the Class IE breakers that are provided for these feeders.

While the existing IE breakers in the feeder circuits assure that these cables (which do not serve safety related functions) will not affect the safety-related circuits with which they are associated, these new Class IE current limiters provide enhanced fault protection for the feeder circuits. They were seismically installed and do not affect the functioning of any circuit. No unreviewed safety question was involved.

20. Provide Backup Nitrogen to Boration Valve

Valve LCV-112B is an air-operated valve which opens to allow the charging pumps to draw water from the refueling water storage tank. A backup nitrogen supply has been furnished to allow operation of the valve in the event the normal air supply is lost. The backup nitrogen supply is seismically restrained to prevent any interaction with the normal air supply or any safety related equipment. No technical specifications or bases were affected by this modification and no unreviewed safety question was involved.

21. Compressed Air Backup

An additional source of compressed air to the station air system (which serves as a backup to the instrument air compressors) has been supplied in the form of a bottle trailer and associated regulator, piping, and valves. Air would be supplied from this source only in the event the station air pressure falls below 70 psig. The station air system is protected by a relief valve with sufficient capability to preclude overpressurization by this backup supply. Check valves are provided to prevent any potential depressurization due to failure of the

backup supply. The backup supply cannot degrade any safety related system or equipment. It does not impact any technical specification or basis. No unreviewed safety question was involved.

22. Emergency Diesel Generator Cooling Setpoint Changes

Setpoints have been raised by for the following, which are related to the emergency diesel generators:

- Jacket water high temperature alarm was raised from 175^{0} F to 185^{0} F,
- Jacket water high-high temperature alarm and cooling (service) water valves full open signal was raised from 185^{0} F to 190^{0} F, and
- Lube oil high temperature alarm and cooling water valves full open signal was raised from 185^{0} F to 205^{0} F.

These setpoint changes were necessitated by the increase in allowable service water inlet temperature approved by the NRC via License Amendment No. 143 to DPR-26, issued August 7, 1989. The resulting lube oil and jacket water temperatures are within safe limits, as approved by the equipment vendor. No unreviewed safety question was involved.

23. Install Temperature Monitors on Pressurizer Auxiliary Spray Line

To obtain data for evaluation of the potential for thermal stratification, temporary equipment has been installed to monitor temperatures on auxiliary pressurizer spray line 64. Two resistance temperature detectors were strapped to the line and connected to a data logger. The RTDs, cables, and data logger do not affect the functions of any system. The installation has been seismically analyzed and restrained to ensure there was no impact on existing systems. No unreviewed safety question was involved in this temporary installation.

24. Install a Spare Flow Transmitter for the Plant Vent Flow Monitor

Plant vent flow rate is used in assessing plant radioactive effluent releases during normal operation, accidents and post-accident recovery conditions. The flow measuring system was installed in accordance with the criteria of Regulatory Guide 1.97, Revision 3. It was modified to increase its reliability and maintainability by relocating the flow transmitter inside a weatherproof enclosure and by installing a spare transmitter in the enclosure. The new transmitter is an installed spare and does not affect interfaces with plant electrical systems.

The capability of automatically recording the effluent release rate was temporarily interrupted while the modification was being installed, but the noble gas monitor remained functional and vent flow rate was available from other instrumentation.

This modification meets or exceeds the design basis of the flow measuring system

and does not change its functional performance. The enclosure is seismically mounted to assure that safety related equipment will not be affected in the event of earthquake. The margins of safety defined in the technical specifications and bases were not affected by this modification, and no unreviewed safety question was involved.

25. Improve Flow Control for Steam Generator Blowdown Sample

The system that monitors and controls sample flow to the steam generator blowdown radiation monitor was redesigned and replaced. The new system provides automatic regulation and improves the reliability of the flow balance to the radiation monitor which is set by the operator, thereby improving the capability for recognition of leakage from the primary to the secondary side of the steam generators. All new equipment is downstream of the blowdown isolation valves. It is seismically designed to remain functional following a seismic event. No technical specification or basis was affected by this modification. No unreviewed safety question was involved.

26. Reroute Cables for Health Physics Computer

Cables associated with the health physics computer were found to be installed in a manner inconsistent with the plant channelization requirements. They were removed and replaced with new cables which are properly routed. Seismic loading criteria were maintained in this installation. The cables are not safety related. This modification did not impact any technical specification or basis or involve an unreviewed safety question.

27. Replace Residual Heat Removal Heat Exchanger Outlet Temperature Sensor

The resistance temperature detector which monitors reactor coolant returning from the residual heat removal heat exchanger was replaced, along with its resistance-to-current converter. The new components are environmentally qualified, implementing a Con Edison commitment regarding Regulatory Guide 1.97, Rev. 2.

This channel is used to monitor performance of the heat exchangers. No initiation of safeguards equipment or operator action to mitigate an accident is dependent on this temperature channel. The response characteristics, drift and instrument error are within the original specifications. The margins of safety as defined in the basis of the technical specifications were not changed and no unreviewed safety question was involved.

28. Install Isolation Valves for Steam Heating Coil for Refueling Water Storage Tank

Isolation valves have been installed at the inlet and outlet of the steam heating coil of the refueling water storage tank, and some piping components were replaced. The isolation valves were installed to safety related, seismic class

I requirements so the seismic integrity of the tank is not changed. Other components are seismically restrained. The modified system is capable of the same functional performance as the original, with heating coil isolation capability added. This modification did not involve the basis of a technical specification and no unreviewed safety question was involved.

29. Emergency Diesel Generator Starting Air Supply Improvements

The systems that supply compressed air for starting the emergency diesel generators have been modified to improve the quality of the air by reducing moisture and rust. Moisture separators, drainers and strainers were added and some piping and components were replaced with stainless steel items. The system is safety related and is seismically supported throughout. The changes do not degrade its safety function; rather, reliability is improved. No modification of a technical specification or basis was required and no unreviewed safety question was involved.

30. Additional Radiation Monitor

In satisfaction of a Con Edison commitment regarding Regulatory Guide 1.97, Rev 2, a radiation monitor was added in the primary auxiliary building between motor control centers 26AA and 26BB. The monitor provides an indication of the habitability of the area. It does not adversely affect any systems or components required for accident prevention or mitigation and its failure would not result in release of radioactivity. No unreviewed safety question was involved.

31. Central Control Room Air Conditioning Modifications

The control panel for the central control room air conditioning system was redesigned and replaced to improve its functional layout and implement human engineering enhancements identified by the detailed control room design review. A humidifier was added to the system to improve personnel comfort.

The modifications were executed with the plant in the shutdown condition so that control room air cleanup capability would not be required. They did not adversely affect safety systems and did not affect the technical specifications or their bases. No unreviewed safety question was involved.

32. Provide Redundant Signals to Close Steam Generator Blowdown Isolation Valves

Two redundant logic channels are provided to automatically start the auxiliary boiler feed pumps on the occurrence of low-low steam generator water level or other conditions. One of these channels also initiates closure of the steam generator blowdown isolation valves. A relay was added so the second channel provides a redundant signal to initiate closure. The new relay and associated cable are qualified consistent with Class IE requirements.

This modification increases the number of channels that can automatically

initiate valve closure from one to two, thereby providing added assurance that the valves will close when required. It does not affect any control functions. The new relay is seismically qualified and mounted and is of the same type as existing relays in the circuit. The margin of safety as defined in the bases of the technical specifications was not reduced, and no unreviewed safety question was involved.

33. Nitrogen Backup for Steam Supply Valves to Auxiliary Feedwater Pump

A backup nitrogen supply has been added for the steam shutoff valve and the steam control valve for the steam driven auxiliary feedwater pump. This will allow normal operation of the valves in the event of loss of instrument air. In the event of loss of both air and nitrogen, the valves may be manually operated.

This modification improves the overall operation of the auxiliary feedwater pump if instrument air is not available. The supply system meets seismic Class I requirements and did not affect the basis for any technical specification. No unreviewed safety question was involved.

34. Primary Auxiliary Building Ventilation Improvements

The supply air handling unit in the primary auxiliary building was replaced, along with the heating steam supply and condensate return piping. The supply and return air ducts were modified to improve ventilation and eliminate air stagnation. Provisions were made for operation of two exhaust fans during periods of a high concentration of airborne radioactivity. Steel barriers were placed between the steam piping and two safety related motor control centers to protect them in the event of leakage or a break.

The modified design meets existing design requirements. The supply air handling unit, new duct work, piping, and shields are seismically supported. The technical specifications and bases were not affected by this modification, and no unreviewed safety question was involved.

35. Install Position Indication for Sampling Valves

Position indicating lights have been added in the control room for twelve existing post-accident sampling valves. This satisfies a Con Edison commitment to provide position indication as a backup variable in monitoring containment integrity per Regulatory Guide 1.97, Rev 2. The lights provide indication only and their failure would not affect the functional capability of the valves. The lights and associated cable and conduit are seismically supported so they will not affect the performance of any safety related systems. No technical specification basis is affected and no unreviewed safety question was involved.

36. Install Isolation Valves and Decontaminate the Regenerative Heat Exchanger

Additional isolation valves were installed in all four lines of the regenerative

heat exchanger, and connections for chemical cleaning were added. After all fuel had been removed from the reactor, the exchanger was isolated and chemically decontaminated.

The decontamination process (CANDEREM) was a product of extensive industry development and it was determined by analysis and material testing that the decontamination chemicals would have no significant deleterious effect on the heat exchanger and other components they contacted. During normal operation, each in-line valve is open and each cleaning connection is double-isolated by a closed valve and a cap or flange. The chemicals were confined to the equipment being decontaminated by double isolation and were thoroughly removed. Chemistry sampling was performed prior to replacing the fuel. The installation of isolation valves is seismic Class I and does not impact the functional capability of the system. It did not involve any technical specification basis and no unreviewed safety question was involved.

37. Provide Shutoff for Lube Oil Heaters of Emergency Diesel Generators

Each emergency diesel-generator has a thermostatically controlled heater to maintain its lube oil above a minimum required temperature when it is not in operation. In accordance with the manufacturer's recommendation, relays were added to lock out power to each heater when its EDG is in service. This precludes the heater from being a potential source for explosive reaction with oil vapors within the crankcase. The modification did not impact the functional capability of the system and no technical specification basis was affected. No unreviewed safety question was involved.

38. Provide Automatic Makeup to Instrument Air Closed Cooling System

A makeup line was provided to automatically supply city water to maintain the inventory of the instrument air compressors closed cooling water system. Inventory had been manually maintained. Operation of the compressors is not affected. Piping passing through a 480 volt switchgear room is capable of maintaining its integrity in the event of a safe-shutdown earthquake. No unreviewed safety question was involved in this modification.

39. Installation of Flowmeters to Facilitate Inservice Testing

To facilitate inservice testing, locally-indicating flowmeters have been installed in the discharge lines of the boric acid transfer pumps and in the recirculation line of the steam driven auxiliary feedwater pump.

These installations are able to withstand a safe shutdown earthquake and maintain their integrity. They do not affect the operation of the pumps and systems, and did not involve any technical specification or basis or an unreviewed safety question.

40. Annunciation System for Heat Trace and Deluge Alarms

A new annunciator assembly was installed in panel "SO" to replace the lights which indicated status of heat trace circuits. The replacement does not affect the functioning of the heat tracing but it improves the reliability and ability to detect failures. The new system includes a reflash capability which allows annunciation of additional circuit failures even in the presence of an existing signal. The new annunciator is seismically mounted in the control room. The modification does not create the possibility for an accident or malfunction not previously evaluated. Although the technical specifications require some heat trace circuits to be operable, they are not affected by this modification. No unreviewed safety question was involved.

41. Add Polymer to Liquid Wastes

Equipment was installed in Unit No. 1 to add a polymer to liquid radwastes before treatment. The purpose is to test the effect of the polymer on the ability to coagulate colloidal particles, allowing them to be removed by filtration. This equipment is connected to waste processing piping. It does not interface with or affect systems required to prevent or mitigate an accident. Any leakage from the system would be contained. No technical specification or basis was affected and no unreviewed safety question was involved.

42. Provide for Operation of RHR/LPI Valves from Control Room

Valves 1863 and 883 are in the recirculation line for the residual heat removal (RHR) pumps, which also serve as the low pressure injection (LPI) pumps. The valves are closed during normal reactor operation when LPI operability is required. Manual valve 1863 was replaced with a non-powered motor operator/valve combination capable of future motor operation upon provision of an appropriate electric power supply. The capability for manual operation of the valve was maintained.

This modification does not affect the capability of fulfilling both the normal and safeguards functions of the RHR system. The new motor operator/valve combination is capable of withstanding a safe-shutdown earthquake without affecting the functional capability of the system and without affecting other safety-related systems. The margin of safety as defined in the technical specification and its bases was not reduced. No unreviewed safety question was involved.

43. Removal from Service of the Gross Failed Fueled Detector

The gross failed fuel detector was intended to monitor gross activity in the reactor coolant for rapid assessment of fuel behavior. It was installed prior to initial plant operation as a research and development project, and has not operated successfully. It has been procedurally removed from service. Primary coolant activity is monitored in accordance with the technical specifications by other means.

The system has no accident mitigation function. There are no regulatory requirements associated with this system, which is not safety related, and it is not referenced in any technical specification or basis. No unreviewed safety question was involved in removing it from service.

44. Replace Battery Room Exhaust Fan

The exhaust fan for battery room 24 was replaced. The replacement fan has a larger flow rating and draws slightly more power than the original. It is seismically mounted. The function of the fan is not affected. The power supply is capable of supplying the increased load. No unreviewed safety question was involved.

45. Rewire Bypass Breakers of Reactor Trip System

The bypass breakers of the reactor trip system are racked in and active to provide trip capability when the normal trip breakers are surveilled or maintained during reactor operation. Wiring changes have been made to the bypass breakers to make them identical to and interchangeable with the trip breakers. The modification facilitates maintenance of the trip system. Neither the function nor the failure modes of the breakers and the trip system was changed. No unreviewed safety question was involved.

46. Install Additional Residual Heat Removal Flow Monitor and Perform Flow Tests

The industry has found that vortexing, air entrainment, and air binding of residual heat removal pumps may occur during reactor shutdown with a lowered water level. An indicator for residual heat removal pump flow has been installed in the control room to provide the operators with an aid for detecting and diagnosing the onset of this condition. This instrument, which has a range of 0 to 3000 gpm, is in addition to the previously-existing flowmeter.
With all fuel removed from the reactor, testing was performed to correlate indicated flow with other available data.

The design and installation of the new flowmeter and associated equipment do not degrade the functioning of safety related equipment. No unreviewed safety question was involved in the addition of this flowmeter and testing with fuel out of the reactor.

47. Install Reactor Coolant System (RCS) Draindown Level Monitoring System

An enhanced system was installed to measure reactor coolant level with the reactor shutdown and the RCS in a draindown condition. The enhanced system uses an ultrasonic level sensor. It provides control room indication and annunciation and an input signal to the plant PROTEUS computer. The existing level indication will remain as a backup. Level monitoring is intended to aid the operator in avoiding vortexing of the RHR supply which could cause pump cavitation. The added level sensor is strapped to the bottom of hot leg #21, and does not

penetrate the reactor coolant boundary. System components are seismically restrained. This modification did not involve an unreviewed safety question.

48. Installation of DORIS Computer

The necessary support hardware for a computer system for daily operations reporting and inspection (DORIS) has been installed, including power supplies for work stations at various locations in the plant. Additional software development and field testing will be completed prior to placing the system into operation. It is not safety related, and components have been seismically restrained as required to prevent interaction with safety related systems. Seismic design of cable trays is not affected. No unreviewed safety question was involved with this modification.

49. Delay Generator Trip on Safety Injection and High Steam Generator Level

Circuitry has been modified to introduce a 30 second delay in tripping the generator on occurrence of safety injection signal or a high steam generator level signal. The purposes of the delay are to assure full reactor coolant flow by maintaining power to the reactor coolant pumps and to limit overspeed of the turbine generator. The turbine trips immediately on occurrence of the two signals; other turbine and generator trips are not affected. The fast transfer of station auxiliary loads from the unit auxiliary transformer to the station auxiliary transformer occurs at the time the generator is tripped. Emergency diesel generator starting on the occurrence of a safety injection signal is not affected.

This change has been evaluated for effects on instrument and control system, effects on loss of coolant accident analyses, and effects on analyses of other accidents. The evaluations showed that the consequences of accidents previously evaluated are not increased by this modification. The modification does not create the possibility of an accident of a different type than those evaluated in the safety analysis report, and the margin of safety as defined in the basis of the technical specifications was not decreased. No unreviewed safety question was involved.

50. Install Back Draft Damper in Cable Spreading Room

A back draft damper was installed at the entrance to the cable spreading room, as part of the ventilation system for the electrical penetration tunnel. Its purpose is to assure that proper temperatures are maintained in the battery rooms. A thermometer was installed in each battery room to monitor the temperatures. The additional components were seismically mounted. The back draft damper did not interfere with the adjacent fire damper. No unreviewed safety question was involved.

51. Installation of Helicoflex O-Ring to Seal Reactor Vessel Closure Head

A Helicoflex O-ring was installed in the inner O-ring track to seal the reactor vessel closure head and prevent weepage during reactor operation, replacing the self-energized ring that had previously been used. The helicoflex ring has an internal coil spring to provide increased sealing force against the vessel head and flange. This type seal has been successfully used for similar vessels. The helicoflex ring did not impact the technical specifications and bases, or any accident, and no unreviewed safety question was involved.

52. Generic Replacement of Valves and Blind Flanges

Obsolete manual and check valves, 2 inches and smaller, are being replaced on an "as-needed" basis with Conval valves. In addition, when valves are replaced on vents and drains which are 3/4 inch and smaller, blind flanges are replaced by Swagelok fittings. The new valves and fittings are functionally equivalent to those being replaced. Components meet or exceed existing specifications, materials are compatible, and system functions were not changed. Seismic capabilities are verified as appropriate. No unreviewed safety question was involved by this modification.

53. Improve Toxic Gas Monitoring System

The sampling pumps of the toxic gas monitoring system were relocated and their piping was modified to allow any of the three pumps in each of the two redundant monitor trains to supply airflow to any or all of the three toxic gas monitors in its train. (Previously each pump supplied flow for one monitor.) The pumps were relocated to improve the reliability of the monitors by reducing vibration. The modification was implemented in accordance with the requirements for the system. This modification did not reduce the margin of safety as defined in the technical specifications and bases. No unreviewed safety question was involved.

54. Supply Cooling Water to Secondary Plant Components from Non-essential and River Water Headers

Plant operating procedures have been modified to allow cooling water to be supplied to various coolers associated with the steam and power conversion systems from the non-essential header of the service water system, or from the river water system of Indian Point Unit 1. These loads include, among others, the turbine lube oil coolers, and the boiler feed pump lube oil coolers which had previously been cooled by the essential service water header.

The loads relocated from the essential header are not required for accident mitigation. These changes reduce demand on the Essential Service Water System, providing additional margin between system capability and essential service water requirements for accident mitigation. No technical specifications or bases were affected and no unreviewed safety question was involved.

55. Replace Transmitters Monitoring Service Water Flow to Containment Fan Cooler Units

The fan cooler units (FCUs), which are part of the containment air recirculation cooling and filtration system, are called upon to cool the containment atmosphere under accident conditions. Heat is removed by circulation of service water through FCU cooling coils. The transmitters which monitor service water flow were removed and replaced with environmentally and seismically qualified units. The control room indicators for these channels were also replaced. Other than qualification, system design and function are unchanged. This modification did not involve an unreviewed safety question.

56. Modify Instrument Connection on Service Water Header

A temporary repair to a leaking pressure tap on the service water header was reported in the 1988 report. The temporary repair has been made permanent by removing the leaking pressure tap and converting the temporary pressure housing to a permanent tap. There is no change in the function of the tap. It meets applicable requirements of the system specifications and is seismically qualified. This modification did not involve an unreviewed safety question.

57. Change Power Sources for Service Water Strainers

Each service water pump has a rotary strainer in its discharge. To enhance the reliability of the system, the power sources for the rotary strainers were changed so each strainer receives power from the same source as its associated service water pump. Additionally, the interlocks that allow a strainer to run only if the associated service water pump is running were removed. This interlock did not have any effect on the safety function of the system. Cables and circuit breakers are Class IE, and circuit breakers and conduit are seismically designed. The additional loads are within the capabilities of the power sources. No unreviewed safety question was involved by this modification.

58. Replace and Relocate Cables in Manhole 21

To improve the separation of redundant trains, cables feeding service water pumps 23 and 24 from their transfer switch were removed from manhole 21 and replaced with cables routed through above-ground conduits. These cables are part of the alternate safe shutdown system. The ductbank between manhole 21 and the turbine building was also replaced, as were control and instrumentation cables for the condenser circulating water pumps. The arrangement of the power conduits entering manhole 21 and the cable routing within the manhole were revised to improve cable separation.

During the modifications described above, and during repairs to the power feeds to other service water pumps, temporary feeds were provided. These feeds assured that reliable power would be available for operation of the service water pumps. Temporary power was supplied from alternate safe shutdown sources, other unit 1 and unit 2 sources, and from a dedicated standby diesel generator located near

the service water pumps. The reactor temperature was maintained below 140^{0} F while the temporary feeds were used.

The functional performance of the service water system and the 480 v distribution system is not altered by this modification. The ductbank and safety related cable and conduit are designed to seismic I requirements, and criteria for separation of redundant cables are met. The margins of safety defined in the technical specifications and bases are not affected by this modification. Functional performance capability of the system met applicable requirements during the modifications and repairs. An unreviewed safety question was not involved.

59. Install Drain from Service Water Strainer Pit

To provide a diverse, passive means of drainage, a pipe has been installed from the service water strainer pit to the intake bay. It is equipped with an isolation valve and a swing check valve to prevent backflow in case the river level should ever rise above el. 5'8". The modification does not affect the capability of the service water structures to withstand an earthquake and will not affect the function of the service water system. No unreviewed safety question was involved.

60. Shielding Support Holes in Reactor Head Lifting Rig

Ten holes having a diameter of 11/16 inch were drilled in the outer kickplate of the reactor head lifting rig. The holes were used to support temporary shielding while performing a modification in the reactor head area. The reactor was defueled while the shielding was used.

An engineering analysis determined that the holes do not alter the capability of the lifting rig to perform its function or its seismic capability. It was not the subject of any technical specification or basis. No unreviewed safety question was involved.

61. Generic Replacement of Copper Instrument Tubing With Stainless Steel Tubing

Copper instrument tubing used throughout the plant for both safety and non-safety related systems is being replaced with stainless steel tubing on an as-needed basis or when maintenance is performed on an existing copper system. Stainless steel tubing meets or exceeds the requirements of the copper tubing it replaces and will follow the same routing as the tubing being replaced. All tubing is seismically supported as required. This modification did not involve the basis of a technical specification or an unreviewed safety question.

62. Generic Pump Seal Replacement

Shaft seals (both packed and mechanical) of pumps are being replaced on an "as-needed" basis with Chesterton mechanical seals which are expected to have an

extended service life compared to the previous seals. The replacement seals meet or exceed the design requirements (including leakage limits) of the original seal, perform the same function, do not adversely affect the pump, and maintain material compatibility. Seismic capabilities are not affected. There was no unreviewed safety question.

63. Generic Piping Restraint Modifications

Piping restraints and supports are being replaced and/or removed on an "as-needed" basis. Evaluations conducted for each individual application have demonstrated that the revised design will not adversely affect the operation, performance or function of its system and will maintain material compatibility. The evaluations also address the need for temporary supports during installation. These modifications are not applicable to the removal of any restraint or support required by a technical specification. The evaluations discussed above assure that the revised configuration will perform the same function as the original and that design requirements are met during and after modifications. There was no change to the bases of the technical specifications and no unreviewed safety question was involved.

64. Generic Solenoid Valve Replacement

Exact replacements for certain solenoid valves are no longer available. Existing valves are being replaced on an "as-needed" basis. The following criteria are applied to replacements:

- The replacement performs the same function as the original.
- The replacement meets or exceeds the specification requirements of the original.
- Seismic analyses or reviews are performed when required for any differences between the original and the replacement valve and/or installation.

The replacement valves are functionally equivalent to the original valves (or represent an upgrade) and the specific evaluation of such is documented for each valve replacement. These changes did not involve an unreviewed safety question.

65. Radiation Monitoring Betterment Program

As part of a continuing program to improve the reliability and maintainability of the process radiation monitoring system, two additional monitors were installed, replacing two monitors of the existing system. The new monitors, R45 and R57, monitor the condenser air ejector discharge and nuclear service building sewage, respectively. Their alarms are annunciated in the control room, and R45 communicates with the Safety-Related Display Control Module. R45 is capable of remaining functional after a steam generator tube rupture coincident with a loss of offsite power.

Since these channels and equipment in their areas are not required to be

functional after a seismic event, the new monitors, the local control units and associated piping are not seismic Class I. The margins of safety as defined in the bases for the technical specifications are not changed. This modification did not involve an unreviewed safety question.

66. Replace Instrument Valve in Reactor Coolant System

A 1/2" valve in the reactor coolant system was removed and replaced with a 3/8" valve. Also replaced were fittings and tubing associated with the valve, which is an isolation valve for two pressure transmitters. Applicable system requirements for materials, design and construction are met by this replacement. System functioning was not altered and the technical specifications were not affected. No unreviewed safety question was involved in this replacement.

67. Defeat Turbine Supervisory Instrumentation Trip

To prevent inadvertent trips during troubleshooting activities, the turbine supervisory instrumentation trip defeat switch was placed in the defeat position (for approximately two hours) on two occasions. The alarm functions were not affected. The turbine supervisory instrumentation is provided for protection of the turbine-generator unit and serves no nuclear safety function. This instrumentation was not the subject of any technical specification or basis and no unreviewed safety question was involved.

68. Containment of Leakage in Service Water Return Line

A wrap was placed over a pinhole leak downstream of the last isolation valve in the piping which returns service water from the containment fan coolers to the river. The leak was located in the neck of a flange in the primary auxiliary building. Radiography was used to initially and periodically verify that the remaining wall thickness was more than adequate. The wrap was installed to preclude unnecessary leakage and was not required to preserve necessary piping integrity. The cover does not affect the functioning of any system and the material used (epoxy modified polybutadiene rubber) does not adversely affect the pipe. No unreviewed safety question was involved.

69. Use of Sealant to Repair Packing Leak in Pressurizer Spray Valve

An attempt was made to repair a packing leak on pressurizer spray valve PCV-455B by drilling a small hole through the bonnet into the seal area and injecting a predetermined amount of sealant. The alteration of the bonnet was allowed by code. The sealant is compatible with the reactor coolant and the valve materials. The amount injected was controlled to assure that no significant amount of sealant could enter the spray piping. Although the sealant was not expected to inhibit valve operation, potential valve failure modes were evaluated. This valve is not assumed to change position in any of the bounding accident analyses in the UFSAR. No unreviewed safety question was involved.

70. Install Resistor in Control Circuit of Emergency Diesel Generator (EDG)

A resistor in the run relay circuit of EDG 21 was found to be defective, inhibiting normal shutdown of the EDG. Another resistor was installed in parallel with it, restoring the functioning of the circuit. The autostart and emergency stop circuits of the EDG were not affected by the failure or repair. No unreviewed safety question was involved by this repair.

71. Replacement of RWST Purification Pump

To permit continued purification of the contents of the refueling water storage tank (RWST), the RWST purification pump was replaced by a monitor pump for a period of several months, following which a permanent replacement pump was installed. The monitor pump and the original pump have the same size and configuration, although their head-flow curves differ. This difference was addressed in the operating procedure. The temporary pump and connecting lines were seismically evaluated. However, because the shaft seal was not evaluated, the temporary pump was isolated from the tank when the reactor coolant system temperature was in excess of 350^{0} F.

The hydraulic performance of the permanent replacement pump is the same as that of the original. The new pump, which uses mechanical seals, has been seismically analyzed to assure that system boundary integrity will be maintained in the event of a safe-shutdown earthquake.

The RWST purification pump does not perform any accident mitigation or safe shutdown function and neither the temporary substitution nor the permanent replacement pump affected the functional capability of the RWST. No technical specification bases were affected and no unreviewed safety question was involved.

72. Alternate Method to Transfer Refueling Water to the RWST

Refueling water was transferred from the fuel transfer canal to the spent fuel pit by a submersible pump. Water from the pit was transferred to the refueling water storage tank (RWST) via the spent fuel pit purification system. This two step process was performed to ensure that the RWST, when filled, would not exceed the activity limit in the technical specifications.

Potential failure modes and accidents were considered, including siphoning via the submersible pump, leakage into the transfer canal, alternate suction paths from the spent fuel pool, and excessive discharge rate from the spent fuel pool. The installation of the submersible pump and the operating procedures together assured that the possibility of an accident or malfunction which had not been previously considered was not created. The consequences and probability of previously analyzed accidents were not increased. Normal spent fuel pit water level and cooling were maintained during the water transfer. The margins of safety were not decreased and no unreviewed safety question was involved.

73. Provide Alternate Power for Battery Chargers

A temporary feed to battery charger 23 was provided while the 480 volt bus which normally supplies its power was out of service for maintenance. The temporary feed was used (at a different time) to supply charger 24 while its supply was undergoing maintenance. The source of power for the temporary feed was selected to be separate from the sources for other chargers. The reactor was in the cold shutdown mode while these feeds were used. No technical specification or basis and no unreviewed safety question was involved.

74. Clear Alarm for Fire Main Loop Isolation Valves

To clear the annunciator for the fire main loop isolation valves, a jumper has been installed to bypass annunciation for several valves associated with Unit 1. These valves have been locked open, which meets the requirements of the fire protection plan. There is no impact on Unit 2 systems and no unreviewed safety question was involved.

75. Temporarily Deactivate Oil Level Alarms for One Reactor Coolant Pump

To clear a ground indication in the instrument bus, the oil level alarm switches for reactor coolant pump 23 were temporarily disconnected. Indication and annunciation for high bearing temperature continued to be operable to provide information to the operators. The alarm is not safety related and is not required by the technical specifications. No unreviewed safety question was involved.

76. Temporarily Reroute IVSWS Supply Tubing

The Isolation Valve Seal Water System (IVSWS) is designed to supply sealing water between the two isolation valves in certain lines penetrating the containment structure in the event of a loss of coolant accident. The tubing supplying sealing water to the line from the reactor coolant pumps seals was temporarily replaced. The new supply line was connected through a vent. The rerouted supply was a functional equivalent of the normal supply and was evaluated to assure the continued seismic capability of the system. No unreviewed safety question was involved.

77. Use of Temporary Sight Gauge for Component Cooling Surge Tank Level

While the normal level instrumentation for the component cooling system surge tank was under repair, a temporary sight gauge was installed and used. The temporary gauge was visually checked at intervals of four hours, and was required to be isolated by closed valves at all other times. Functions served by the level alarms were reviewed and it was concluded that other instruments are available which would be capable of indicating failures that would affect the component cooling water system. Functional capability of the system was maintained and there was no unreviewed safety question.

78. Use of Spare Breaker for EDG Auxiliaries

The breaker feeding the motor control center which supplies auxiliaries for emergency diesel generator (EDG) 23 became defective. An existing spare breaker in another cubicle of the same switchgear was temporarily used to supply power to the MCC. The spare breaker is identical to the original, and performs the same function. No unreviewed safety question was involved.

79. Disable Condenser Water Box Level Alarm

The control room alarm for condenser outlet water box level was temporarily disconnected. Other alarms are available to indicate circulating water system malfunctions, including inlet water box level. Removal of this alarm has no impact on any safety system or any accident analysis in the UFSAR. No unreviewed safety question was involved.

80. Provide Temporary Evacuation Alarm in Containment

While the normal containment evacuation alarm was under repair, portable alarm systems were installed. These systems served the same function as the normal system. They had no functional or physical interfaces with safety related equipment, and design loadings of structures would not be exceeded in a seismic event. No unreviewed safety question was involved.

81. Operate Containment Purge Exhaust Fan Without Supply Fan

The interlock between the supply and exhaust fans of the containment purge system was temporarily removed to allow the exhaust fan to be operated without the supply fan. The resulting slight negative pressure in the containment is within the technical specification limit and well within the capabilities of the affected components. The safety related function (isolation) of the system was not affected, and no unreviewed safety question was involved.

82. Use of Fire Protection Water to Clear Unit 1 Screen Channels

The fire protection system was used as a source of water to remove debris from the channel guides which support the Unit 1 riverwater screens. The flowrate was restricted to 120 gpm by an orifice installed between the hose and the hydrant from which flow was taken. This flowrate plus the largest demand assumed in the fire protection plan is within the capability of the fire protection system even with the failure of a single pump. There is no safety related equipment in the area. No unreviewed safety question was involved.

83. Upgrade Containment Sump Grating

Stainless steel mesh was installed to cover edge gaps around the grating over the containment sump. The grating (which has a 1" x 4" mesh) serves to keep

large particles from entering the sump during the recirculation phase of a loss-of-coolant accident. A 1/4" mesh screen in the sump prevents smaller particles from reaching the pump suctions. The installation around the grating edges will help keep large particles from reaching the screen, and provide additional assurance that it can perform its safety function. This modification assures that the grating can perform its function. It did not reduce the margin of safety or involve an unreviewed safety question.

84. Install Temporary Pressure Relief Fan

Normal containment pressure changes during power operation are handled by a pressure relief line which has isolation valves and a fan. Air from the line is filtered and discharged to the plant vent. The pressure relief fan, which has a flow rating of 1500 cfm, was temporarily replaced by a fan rated at 1000 cfm. The smaller fan serves the same function as the original, but pressure relief is at a lower rate. The replacement fan was seismically mounted. Functioning of system interlocks (isolation valve closure, etc.) was not affected. Installation of this temporary fan did not increase the consequences or probability of occurrence of an accident, and did not involve an unreviewed safety question.

85. Generic Replacement of Instrument Valves

Instrument valves, valve manifolds, and other tubing valves are being replaced on an "as-needed" basis. Engineering analyses for each installation assure that seismic qualification is maintained as required and that replacement valves meet or exceed the requirements for the system in which they are installed and provide functional equivalence to the valves being replaced. These replacements did not impact the basis of a technical specification and they did not involve an unreviewed safety question.

86. Supply Power to Construction Trailers

Construction trailers have been provided with a permanent electrical power supply from a 13.8 KV/480 V substation. The substation is not safety related and does not supply power required for safe shutdown or accident mitigation. The power supply cannot adversely affect any safety related system, structure or component and does not impact any technical specification or basis. No unreviewed safety question was involved.

87. Deoxygenate Water in Condensate Storage Tank

A portable system was installed and is operated to recirculate and remove oxygen from the water in the condensate storage tank. Connections for the system were installed in non-safety-related condensate piping. Oxygen reduction minimizes corrosion in the steam generators and the condensate and feedwater systems.

The inlet connection to the deoxygenation system is downstream of a pre-existing

automatic isolation valve which closes in the event the tank level falls to a preset point. The return to the storage tank is through a check valve. These valves assure that the tank contents are sufficient for safe reactor shutdown in the event of failure in the non-safety related condensate piping, including the deoxygenation Unit. Filters are provided to assure that particulates from the unit will not enter the tank. The system location assures that its failure would not affect safety-related equipment and piping is capable of withstanding the maximum pressure available from the Unit. No unreviewed safety question was involved with the installation and operation of the system.

88. Secure Two Non-essential Service Water Pumps to Repair Test Valves

Two non-essential service water pumps were secured for a short time to allow maintenance work on two valves. Plant operations for these evolutions were governed by a Temporary Operating Instruction (TOI). Provisions were made for operation with the non-essential service water header isolated from the essential header and supplied by one service water pump and also for the nonessential header tied to the essential header and supplied by one pump. Both configurations are allowed for limited time periods by the technical specifications. Contingencies were provided for in the TOI. The TOI, along with the plant emergency operating procedures, assured that adequate cooling would be available in the event of postulated accidents or malfunctions. These evolutions did not involve an unreviewed safety question.

89. Supply Power to Telephone Equipment from the Security Emergency Diesel Generators

The purpose of this modification was to supply power to telephone equipment from a distribution panel that has backup power capability from the Security Emergency Diesel Generators. This modification introduces no new failure modes, no safety system interactions and could not adversely affect security equipment or facilities. Fire barrier penetrations were resealed in accordance with the site Fire Protection Program.

90. Replace Security TV Cameras

Five Pan-Tilt-Zoom mechanisms and controllers for scanning cameras in the field were replaced with new mechanisms and controllers. Additionally, a new fixed camera was installed in the field.

Fiber optic cables were installed. A cable tray loading analysis showed that all resulting cable tray loads were satisfactory. The Fire Protection Program was not affected by this modification. No safety system interfaces were involved and no unreviewed safety question was introduced as a result of this modification.

91. Replace Emergency Security Power Cable

This modification involved the replacement of a section of emergency security power cable. The new cables use the existing cable conduit and some additional conduit that was put in place for part of the cable run.

It was determined that there is no impact on the Fire Protection Program as a result of this modification. There are no electrical interfaces between any safety related system and the Security Emergency Diesel Generators and equipment involved. There is no change in the purpose or the functioning of the Security Emergency Diesel Generator as a result of this replacement. There was no reduction in the margin of safety and no unreviewed safety question as a result of this modification.