Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

August 1, 1994

Mr. William T. Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Application for Amendment to Facility Operating License:

Byron Station Units 1 and 2
(NPF-37/66; NRC Docket Nos. 50-454/455)

"Steam Generator Interim Plugging Criteria"

References: See Attachment J

Dear Mr. Russell:

Pursuant to 10 CFR 50.90, Commonwealth Edison Company (ComEd) proposes to amend Appendix A, Technical Specifications of Facility Operating Licenses NPF-37 and NPF-66. The proposed amendment request revises Technical Specification 3/4.4.5, STEAM GENERATORS, to incorporate a 1.0 volt steam generator (SG) tube interim plugging criteria (IPC) beginning with Unit 1 Cycle 7. The proposed changes clearly indicate that IPC is only applicable to Unit 1. The License for Unit 2 is affected only due to the fact that Units 1 and 2 use common Technical Specifications.

The attached safety analysis shows that this proposal will have minimal impact on safety. Amending Technical Specification 3/4.4.5 would allow tubes with flaw indications at the tube support plates to remain in service regardless of depth provided they meet the following criteria:

- Axial Outer Diameter Stress Corrosion Cracking (ODSCC) is the dominant degradation mechanism,
- Indications are within the tube support plate length,
- Indications are less than or equal to 1.0 volt as measured per the applicable inspection guidelines,
- Indications greater than 1.0 volt and less than or equal to 2.7 volts and are not confirmed as ODSCC by rotating pancake coil (RPC) probe inspection, and
- Indications are located outside of the "wedge" areas of the SGs.

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Change: NRC PDR

This philosophy is in general agreement with the draft NUREG-1477, "Voltage-Based Interim Plugging for Steam Generator Tubes - Task Group Report". It is also consistent with recent Safety Evaluations issued to Braidwood, Catawba, Joseph M. Farley, and Donald C. Cook Nuclear Power Plants for application of IPC, and the Safety Evaluation issued to Palo Verde Unit 2 prior to restart following the March 14, 1993 tube rupture event. Additional considerations from the Braidwood Safety Evaluation Report (SER) for IPC dated May 7,1994, the Braidwood technical support document for IPC (WCAP-14046), public comment resolutions on draft NUREG-1477 from the February 8, 1994 NRC/industry meeting, and the EPRI technical support documents for ODSCC at support plates, have been incorporated into the proposed Byron Unit 1 IPC amendment.

In support of this request, the following information is attached:

Attachment A:	Description and Safety Analysis Of The Proposed Changes
Attachment B:	Interim Plugging Criteria Methodology
Attachment C:	Corrective Actions to Address Steam Generator Degradation
Attachment D:	System Operational Measures/Defense In Depth
Attachment E:	Revised Technical Specification Pages
Attachment F:	Evaluation of Significant Hazards Considerations
Attachment G:	Environmental Assessment
Attachment H:	WCAP-14046, "Braidwood 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May 1994, Proprietary and Non-proprietary versions
Attachment I:	Eddy Current Inspection Guidelines
Attachment J:	References

A brief discussion of the contents of Attachments B through D follows.

Attachment B: Interim Plugging Criteria Methodology

WCAP-14046, submitted in support of the Braidwood IPC request and included as Attachment H to the Byron request, serves as the basis for the methodology that will be used to evaluate Byron Unit 1 SGs. The methodology itself, included in Sections 5.0 and 6.0 of the WCAP, is generic in nature, with the application of the methodology being a plant specific effort. Attachment B summarizes the significant features of the methodology and addresses the application of this methodology to Byron Unit 1.

Attachment C: Corrective Actions to Address Steam Generator Degradation

Consistent with the Braidwood IPC request, this Attachment reviews the historical water chemistry data for Byron Unit 1 SGs and presents a discussion of the corrective actions that are in place or planned to reduce the likelihood of SG degradation.

Attachment D: System Operational Measures/Defense In Depth

In Attachment D, Byron addresses information on various equipment, programs, and procedures to allow a thorough review of the IPC request. System Operational Measures discussed include the ability to detect and monitor SG tube leakage, the procedural guidance available to operators in the event that leakage is known to exist, and operator training relative to SG leakage identification and response. These measures ensure that Byron Station has the ability to detect SG leakage and take appropriate actions in response to any indicated leakage. Eddy Current Inspection Program enhancements are also discussed in this section. These enhancements have been implemented to ensure reliable and consistent detection and analysis of ODSCC.

As a result of a few large indications observed at the end of Braidwood Unit 1 Cycle 4, two additional analysis methodologies, listed below, have been pursued by ComEd for both Byron and Braidwood Stations to provide a "defense in depth" approach to IPC. Although these methodologies are not used as the bases for the Byron Unit 1 IPC, they provide added assurance of SG tube integrity beyond that relied upon for acceptance of the IPC approach.

- Evaluation of the limited TSP displacement during a main steamline break event (MSLB).
- Evaluation of a Probability of Detection (POD) greater than 0.6 for larger voltage amplitude indications.

As further assurance of the low impact of IPC on risk of core damage, an evaluation was performed to compare containment bypass core damage frequency with and without application of IPC at Byron Unit 1.

IPC Analysis and Technical Support-Proprietary Information

Attachment H provides information in support of the implementation of a proven bobbin coil voltage-based interim plugging criterion. This report contains information which is proprietary to Westinghouse Electric Corporation. Accordingly, we request that this information be withheld from public disclosure. In compliance with the requirements of 10 CFR 2.790, Byron is providing proprietary and non-proprietary versions of the material together with an affidavit with this submittal.

Additional Analysis and Documentation to Follow

In addition to the technical information provided in the attachments, ComEd will submit to the NRC the below documents in a timely manner. We are unable to provide this information at this time as these items warrant additional testing, analysis and review which will not be available until after the start of the B1R06 outage scheduled for September 1994. We are confident that the results of this technical information will strengthen our submittal and will in no way adversely impact the safety significance.

- The Byron End of Cycle 6 eddy current inspection results and the Cycle 7 leak and burst assessment:

  The inspection results through Cycle 5 have been consistent with axially oriented ODSCC found at other sites with approved IPCs. The Cycle 6 inspection results are expected to continue to demonstrate that the Byron degradation is axially oriented ODSCC and will support the application of an IPC. The leak and burst assessment will demonstrate that the structural adequacy of tubes to which IPC is applied meets the requirements of Regulatory Guide 1.21 and that the expected leak rate remains below the site allowable leak rate limit throughout the next operating period for normal operating and transient conditions.

- Tube Pull Analysis:

  A sample of tubes will be pulled from the Byron Unit 1 Steam Generators during B1R06. These tubes will be selected based on the indication size and voltage distribution. ComEd is developing a program, consistent with that being developed by the Nuclear Industry, to analyze these tubes to achieve the following objectives:

- Gain additional insight into the root cause of degradation of the SG tubing.
- Determine the free span leak rate at MSLB conditions of the larger indications.
- Determine the burst capability of the larger indications.
- Assure the microstructure of degradation conforms to that required for application of alternate repair criteria for ODSCC in the support plate region.

- Alternate repair criteria (ARC) risk evaluation:

  Using the inspection data from B1R06, an evaluation will be done to compare core damage frequency with containment bypass, with and without the interim plugging criteria applied at Byron Unit 1 for Cycle 7.

- IPC Supplement:

  Following release of the draft Generic Letter on Interim Plugging Criteria, a comparison with this submittal will be conducted and any discrepancies resolved in a supplement to this proposed amendment.

Summary Of Results

The proposed license amendment demonstrates that the implementation of this voltage-based plugging criteria ensures that the steam generator tubes qualified for continued service will retain adequate structural and leakage integrity with margins of safety in accordance with Regulatory Guide 1.121 during normal operating, transient, and postulated accident conditions. In addition, various inspection and operational issues have been addressed to ensure reliable and consistent flaw detection, and timely operator diagnosis and response to tube leakage. Recommendations and conclusions from SOER 93-01, IEN-94-43, draft NUREG-1477, the EPRI Technical Support document for ODSCC, and other approved IPC SERs have been incorporated into the Byron IPC methodology. The RCS flow margin and heat removal capacity are maximized and RCS loop asymmetries and loss of rated thermal power are minimized by leaving SG tubes in service which would otherwise require plugging or sleeving under the depth-based repair criteria.

Schedule Request

ComEd requests that this proposed license amendment be approved to permit IPC application during B1R06. Approval of the proposed license amendment is required in order to declare the Byron Unit 1 SGs operable prior to entering Mode 4, (Hot Shutdown) during the Cycle 7 startup. Based on the current outage schedule, Byron Unit 1 anticipates entering Mode 4 on Wednesday, October 26, 1994. It is worthy to note that the steam generator manways are scheduled for reinstallation on Saturday, October 15, 1994. In order to minimize potential rework and scheduling impact, ComEd respectfully requests that this amendment be approved on or before October 15, 1994.

This request for a Technical Specification Amendment has been reviewed and approved by the Onsite and Offsite Review Committees in accordance with ComEd procedures. ComEd has reviewed this proposed amendment in accordance with 10 CFR 50.92(c) and has determined that no significant hazards consideration exists as documented in Attachment F. An Environmental Assessment has also been completed and is contained in Attachment G.

Pursuant to 10 CFR 50.91(b)(1) a copy of this request has been forwarded to the designated State of Illinois Official.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other ComEd employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any comments or questions regarding this matter to this office.

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

Respectfully,

Joseph A. Bauer

Joseph A. Bauer
Nuclear Licensing Administrator

Mary Jo Yack 8-1-94

Attachments

cc: G. Dick, Byron Project Manager - NRR
H. Peterson, Senior Resident Inspector - Byron
B. Clayton, Branch Chief - Region III
Office of Nuclear Facility Safety - IDNS

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSES NPF-37 AND NPF-66

DESCRIPTION OF THE PROPOSED CHANGE

Commonwealth Edison Company (ComEd) proposes to amend the following Technical Specification:

Specification 3/4.4.5 REACTOR COOLANT SYSTEM - STEAM GENERATORS

This proposed license amendment request will modify Specification 3/4.4.5 to allow an eddy current bobbin coil probe voltage-based steam generator (SG) tube support plate (TSP) interim plugging criteria (IPC) to be applied for Byron Unit 1 beginning with Cycle 7.

Technical Specification Bases Section 3/4.4.5, STEAM GENERATORS, will also be modified to reflect these changes.

DESCRIPTION OF THE CURRENT REQUIREMENT

Specification 3/4.4.5

The Technical Specification Surveillance Requirements (TSSRs) associated with Specification 3.4.5 currently require that any SG tube with an imperfection depth at or exceeding the plugging or repair limit of 40% of the nominal wall thickness be removed from service by plugging or repaired by sleeving in the affected area.

BASES OF THE CURRENT REQUIREMENT

Specification 3/4.4.5

The TSSRs for inspection of the SG tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of SG tubes is based on a modification of Regulatory Guide (RG) 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975. Inservice inspection of SG tubing is essential in order to maintain surveillance of the condition of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of SG tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

NEED FOR REVISION OF THE REQUIREMENT

At both Byron and Braidwood, Unit 1 has four Westinghouse Model D-4 SGs and Unit 2 has four Westinghouse Model D-5 SGs. The significant differences between the SG models are in the tube material and tube support materials and design. The D-4s have 0.75" thick carbon steel tube support plates with drilled hole tube supports. The D-5s have 1.125" thick stainless steel support plates with Quatrefoil tube supports. The D-4 SG tubes are mill annealed Inconel 600 which were hard rolled into the tubesheet during initial assembly. Subsequently, the D-4 tubes were shot peened in the tubesheet area and stress relieved in the U-bend area. The D-5 tubes are heat treated Inconel 600 which were hydraulically expanded into the tube sheet during initial assembly. Over the past several refueling outages, the number of SG tubes plugged per outage has been increasing. At each site, Unit 1 has had more defective tubes than Unit 2 primarily due to the design differences between the D-4 and D-5 SGs as mentioned above.

In the most recent Byron Unit 1 Refueling Outage (B1R05), conducted in the spring of 1993, a SG tube inservice inspection was performed in accordance with the current TSSR 4.4.5.0. The results of this inspection identified a total of 1105 bobbin coil indications at the tube support plate locations. Using a rotating pancake coil to confirm these indications, 556 indications were flawed due to ODSCC at the TSPs in 530 SG tubes. The 530 tubes were removed from service by plugging. This increased the overall plugging total for Byron Unit 1 to 847 tubes or 4.6% of the tubes. Of the 847 tubes plugged to date, 671 were plugged due to ODSCC at the tube support plate locations.

For the upcoming Byron Unit 1 Refueling Outage (B1R06), predictions on the number of pluggable indications using the current TSSR 4.4.5 acceptance criteria are approximately 1950 tubes. With the approval to use the Interim Plugging Criteria proposed, the predicted number of tubes requiring removal from service by plugging or repair by sleeving would be reduced to 600.

DESCRIPTION OF THE REQUESTED REVISION

The changes proposed in the amendment are contained in seven inserts to the surveillance requirements for the Byron Technical Specifications and Bases. The inserts are applicable to Unit 1 but not Unit 2. The inserts reflect the option to allow tubes to remain in service using a voltage-based IPC for ODSCC indications in the tube support plate region. Using IPC also results in changes to the sample selection, inspection criteria, and reporting requirements. A new term, "Tube Support Plate Interim Plugging Criteria Limit", is defined to identify the acceptance criteria to be used during the SG inservice inspections to allow a tube to remain in service. Clarifications are made to existing definitions to reference IPC, as appropriate.

Specification 4.4.5.2. Steam Generator Tube Sample Selection and Inspection

Changes to this section of the surveillance requirements will require that all tubes remaining in service due to application of IPC shall be included as part of the tubes to be inspected as an addition to the sample selection made in accordance with existing criteria. Also, the surveillance requirements will specify how IPC will be implemented.

Insert "A" adds a section to Specification 4.4.5.2.b, requiring all tubes in which the tube support plate IPC plugging limit is applied be inspected in each scheduled refueling outage. Insert "A" reads as follows:

> "For Unit 1, tubes in which the tube support plate IPC plugging limit has been applied."

Insert "B" adds section 4.4.5.2.d to describe the inspections associated with the implementation of IPC. Insert "B" reads as follows:

> "For Unit 1, implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. An inspection using a rotating pancake coil (RPC) probe is required in order to show OPERABILITY of tubes with flaw-like bobbin coil signal amplitudes greater than 1.0 volt but less than or equal to 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as ODSCC."

Specification 4.4.5.4, Acceptance Criteria

Insert "C" adds to the definition of "Plugging or Repair Limit", Specification 4.4.5.4.a.6, to identify that this definition does not apply for Unit 1 in the region of the tube subject to the TSP IPC limit, i.e. the TSP intersections, and that Specification 4.4.5.4.a.11 describes the repair limit for use within the TSP intersection of the tube. Insert "C" reads as follows:

> "For Unit 1, this definition does not apply to the region of the tube subject to the tube support plate interim plugging criteria limit, i.e., the tube support plate intersections. Specification 4.4.5.4.a.11 describes the repair limit for use within the tube support plate intersection;"

Insert "D" adds to the definition of "Tube Inspection", Specification 4.4.5.4.a.8, to clarify which portions of the tube subject to IPC are to be inspected. Insert "D" reads as follows:

> "For Unit 1, for a tube in which the tube support plate interim plugging criteria limit has been applied, the inspection will include all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications."

Insert "E" adds Specification 4.4.5.4.a.11 to define the TSP IPC limit. Insert "E" reads as follows:

"11) For Unit 1, the Tube Support Plate Interim Plugging Criteria Limit is used for the disposition of a steam generator tube for continued service that is experiencing outer diameter stress corrosion cracking confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria limit, the tube's disposition for continued service will be based upon standard bobbin coil probe signal amplitude of flaw-like indications. The plant specific guidelines used for all inspections shall be amended, as appropriate, with respect to the voltage/depth parameters specified in Specification 4.4.5.2. An ASME standard calibrated against the laboratory standard will be utilized in Unit 1 steam generator inspections for consistent voltage normalization, pending incorporation of the voltage verification requirements in ASME standard verifications. The acceptance criteria are as follows:

1. A tube can remain in service with a flaw-like bobbin coil signal amplitude of less than or equal to 1.0 volt, regardless of the depth of the tube wall penetration.

2. A tube can remain in service with a flaw-like bobbin coil signal amplitude greater than 1.0 volt but less than or equal to 2.7 volts provided an RPC inspection does not detect degradation.

3. A tube with a flaw-like bobbin coil signal amplitude of greater than 2.7 volts shall be plugged or repaired. Detection of a tube or tubes with flaw-like bobbin coil signal amplitude of greater than 4.54 volts shall require an inspection report, as described in Specification 4.4.5.5.d.2, to be submitted to the Commission prior to plant startup.

4. Plant startup and operation is allowed following application of the above acceptance criteria if, as a result, the projected distribution of crack indications over the next operating period is verified to result in total primary-to-secondary leakage less than or equal to 12.8 gpm (including operational and accident leakage). If the operating period is less than a full cycle, at the end of that operating period, the plant shall be shutdown and an inspection performed in accordance with Specification 4.4.5.2.d. Continued operation beyond that operating, without a shutdown, may be permitted based on additional licensee justification and subsequent approval by the Commission.

Certain tubes identified in WCAP-14046, Section 4.7, shall be excluded from application of the tube support interim plugging criteria limit. It has been determined that these tubes may collapse or deform following a postulated LOCA + SSE."

Specification 4.4.5.5, Reports

Insert "F" adds reporting requirement 4.4.5.5.d to identify the reports, including content and time period, to be submitted to the Commission associated with the implementation of IPC. Insert "F" reads as follows:

"d.1 For Unit 1, if the flaw-like bobbin coil signal amplitudes detected during the inspection of tubes in which the tube support plate interim plugging criteria limit has been applied are less than or equal to 4.54 volts and if the leakage projected at end of the next cycle based on inspection results is less than or equal to 12.8 gpm, preliminary results of the inspection shall be reported to the Commission pursuant to Specification 6.9.2 prior to plant startup. The preliminary results to be reported include maximum indication voltage observed and predicted end of cycle leakage. The final results of inspection for all tubes in which the tube support plate interim plugging criteria limit has been applied shall be reported to the Commission pursuant to Specification 6.9.2 within 90 days following completion of the steam generator tube inservice inspection. This report shall include:

1. Listing of applicable tubes,

2. Location (applicable intersections per tube) and extent of degradation (voltage), and

3. Projected Steam Line (MSLB) Leakage

d.2 For Unit 1, if a flaw-like bobbin coil signal amplitude of greater than 4.54 volts was detected during the inspection of tubes in which the tube support plate interim plugging criteria limit has been applied or if greater than 12.8 gpm leakage is projected at end of the next cycle based on inspection results, both the preliminary and final reports described in 4.4.5.5.d.1 shall be submitted to the Commission for approval prior to plant startup. In addition to the information identified above, the final report shall include justification to permit startup, definition of the operating period, and a description of the actions to be taken at the end of the operating period, if less than a full cycle."

Bases 3/4.4.5, Steam Generators

Insert "G" adds a discussion to the Bases section of Technical Specifications to refer to the dispositioning of tubes in accordance with IPC. Also, a discussion of adjustment of the operating period to meet projected MSLB leakage limitations is included. Insert "G" reads as follows:

> " For Unit 1, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11. The operating period may be adjusted to less than the full operating cycle to meet the 12.8 gpm projected leakage limit. However, if greater than 12.8 gpm leakage is projected for end of cycle, the shortened period must be reported to the Commission for approval prior to plant startup in accordance with Specification 4.4.5.5.d.2. The maximum site allowable primary-to-secondary leakage limit, 12.8 gpm, includes the accident leakage from a faulted steam generator and the operational leakage of the three remaining intact steam generators equal to the Specification 3.4.6.2.c leakage limit."

The specific changes to these Technical Specifications and associated bases are included in Attachment E.

BASES FOR THE REVISED REQUIREMENT

Byron is requesting this revision based on the following considerations:

- The May 1994 approval of the Braidwood request of a 1.0 volt Interim Plugging Criteria for 3/4" diameter SG tubing.

- The approval of similar requests for IPC for other plants with 3/4" and 7/8" diameter SG tubing.

- The NRC's ongoing review of Electric Power Research Institute (EPRI) Draft Report TR-100407, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates," Revision 1, August 1993, and EPRI Draft Report NP-6864-L, "PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions - Rev. 2," Revision 2, August 1993.

- An understanding of the NRC and industry desire to improve the basis for steam generator tube repair.

- The completion of a satisfactory review assuring the structural integrity of Byron SG tubing during the next cycle operation.

To support this request for amendment, Byron will remove tubes, as appropriate, from Unit 1 SGs for laboratory examination, leak, and burst testing. For scheduling and planning purposes, three tubes are expected to be removed during B1R06. It is the intent that each tube removed will include three support plate intersections plus the flow distribution baffle intersection. The tubes will be selected for removal based on the size and distribution of indications which they contain. The results of the Braidwood Unit 1 tube pulls will be incorporated into the Byron evaluation upon completion of Braidwood tube pull analyses.

Analysis required by the IPC methodology will be completed to demonstrate leak and burst capabilities using B1R06 inspection results and Cycle 6 growth rates. The bases of the IPC approach includes, in part:

- Determination of a beginning of cycle (BOC) voltage distribution for Cycle 7 with application of a POD of 0.6 in accordance with draft NUREG-1477.

- Prediction of an end of cycle (EOC) voltage distribution by applying Cycle 6 growth rates to the BOC distribution through Monte Carlo simulations.

- Application of a log-logistic probability of leakage (POL) function.

- Application of the EPRI leak rate versus voltage correlation (conditional leak rate model).

- Calculation of the EOC leak rate and comparison with the site allowable leak rate for off-site dose consideration.

- Calculation of the EOC tube burst probability and comparison with the allowable burst probability per NUREG-0800.

IMPACT OF THE PROPOSED CHANGE

With the implementation of this proposed license amendment request, the Byron Unit 1 SGs will continue to satisfy the requirements of Regulatory Guide 1.121. There will be no significant reduction in the margin of safety to protect the health and safety of the public. Based on current projections, approximately 1950 tubes with ODSCC would require repair under current repair criteria during B1R06. Implementation of a 1.0 volt IPC at Byron Unit 1 will save approximately 1350 tubes from repair. This represents a savings of approximately $5.2M in plugging and sleeving repair costs alone. In addition, IPC implementation saves a minimum

of 24 days in critical path outage time and eliminates the associated replacement power costs. RCS loop asymmetries and the loss of rated thermal power due to excessive plugging and sleeving are minimized through IPC application and RCS flow and available heat transfer area are maximized.

With the issuance of the draft Generic Letter on Interim Plugging Criteria, ComEd will review the draft against this submittal. Any discrepancies will be resolved in a supplement to the submittal.

H. SCHEDULE REQUIREMENTS

ComEd requests that this proposed license amendment request be approved to permit IPC application during B1R06. Approval of this proposed license amendment request is required in order to declare the Byron Unit 1 SGs operable prior to entering Mode 4, Hot Shutdown. Based on the current outage schedule, Byron Unit 1 is predicted to be ready to enter Mode 4 on Wednesday, October 26, 1994. It is worthy to note that the steam generator manways are scheduled for reinstallation on Saturday, October 15, 1994. In order to minimize potential rework and scheduling impact, ComEd respectfully requests that this amendment be approved on or before October 15, 1994.

ATTACHMENT B

BYRON UNIT 1 STEAM GENERATOR INTERIM PLUGGING CRITERIA METHODOLOGY

INTRODUCTION

Byron Unit 1 contains four Westinghouse Model D-4 steam generators. Each generator has 4578 mill annealed Inconel 600 U-tubes that are 3/4" diameter with 0.043" nominal wall thickness. Following the Cycle 5 eddy current tube inspection, 1105 bobbin coil indications were identified at tube support plate locations. Of these, 556 indications in 530 tubes were characteristic of outer diameter stress corrosion cracking (ODSCC) as confirmed by rotating pancake coil (RPC) inspections. Subsequently, all 530 tubes were removed from service by plugging, thus increasing the overall plugging total to 847 tubes or 4.6% of the tubes. Of the 847 tubes plugged to date, 671 were plugged due to ODSCC at the tube support plate locations. The number of repairs are projected to be significant in future outages should the current Technical Specification plugging limit of 40% through-wall be applied. Byron Station is therefore requesting Technical Specification changes to implement a voltage-based interim plugging criteria (IPC) for ODSCC at tube support plate intersections for Unit 1. The requested repair limits and inspection requirements are based on the Braidwood Unit 1 and Catawba Unit 1 NRC Safety Evaluation Reports (SERs) which approved a 1.0 volt repair limit for 3/4" tubing. Additional considerations from the Braidwood SER for IPC, the Braidwood technical support document for IPC (WCAP-14046), public comment resolutions on draft NUREG-1477 from the February 8, 1994 NRC/industry meeting, the EPRI technical support documents for ODSCC at support plates, and draft NUREG-1477 have been incorporated into the Byron Unit 1 IPC methodology.

The design, operation, and licensing basis of Byron Unit 1 and Braidwood Unit 1 are very similar and both contain Model D-4 steam generators. The major mode of degradation experienced by both Byron Unit 1 and Braidwood Unit 1 has been ODSCC. However, through Cycle 5, Byron Unit 1 has not experienced the large voltage and growth rate indications as seen at Braidwood Unit 1. The largest indication at Byron Unit 1 during the previous Cycle was 2.6 volts, as sized during a reanalysis of the plugged tubes using the IPC guidelines for the proposed license amendment. ComEd concludes that the IPC methodology described in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria", May 1994, is applicable to Byron Unit 1 and is used as the basis for the Byron Unit 1 IPC methodology, with the following exceptions:

- The tube support plate limited displacement analysis will not be used.

- The reactor coolant dose equivalent iodine-131 limit is not reduced.

- The probability of detection (POD) used for the beginning of cycle (BOC) voltage distribution is 0.6, in accordance with draft NUREG-1477.

Some of the features of the Byron and Braidwood IPC methodology are:

- A 1.0 volt IPC limit.

- Calculation of the site specific maximum allowable primary-to-secondary leakage during a Main Steam Line Break (MSLB) event based on a small fraction of 10 CFR 100 limits at the site boundary.

- Calculation of the tube structural limit is identical to the method applied during the Braidwood Unit 1 IPC implementation and is based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation.

- Enhanced eddy current inspection guidelines have been implemented to increase detectability and minimize voltage variability.

- The methodology for calculating primary-to-secondary leakage from the steam generator tubes during a postulated MSLB is identical to the method used during the Braidwood Unit 1 IPC implementation and is described in the Braidwood SER issued on May 7, 1994.

- The beginning of cycle (BOC) distribution is determined by applying a POD correction of 0.6 to the voltage distribution obtained from the inspection and adjusting for tubes repaired. The end of cycle (EOC) voltage distribution is predicted by applying a Monte Carlo simulation to the BOC distribution to account for voltage growth during the next cycle and eddy current uncertainty. The EOC distribution is used to evaluate tube burst and leakage during a postulated MSLB event.

- Implement a shutdown primary-to-secondary leak rate limit of 150 gpd per steam generator and a total of 600 gpd limit for all four steam generators. These limits have been administratively placed pending approval of the Byron 06/03/94 sleeving submittal incorporating these reduced limits.

- Remove a sample of steam generator tubes to evaluate the mode of degradation and provide additional data points for leak rate and burst correlations.

INSPECTION CRITERIA

Inservice inspection of the steam generator tubing is an essential element in maintaining the reliability and structural integrity of the tubing. Inservice inspection also provides a means to characterize the nature and cause of any tube degradation so that corrective measures can be implemented. In the past, Byron Station has performed inspection on 100% of the tubes each refueling outage since plant startup with inspection and analysis guidelines consistent with EPRI Steam Generator Inspection Guidelines and Regulatory Guide 1.83. A common Byron/Braidwood eddy current guideline document was created and implemented. Prior to the Braidwood Unit 1 Cycle 4 refueling inspection, the Byron/Braidwood guidelines were updated and used in support of the Braidwood Unit 1 IPC effort. The update incorporated inspection requirements from draft NUREG-1477 and is consistent with those contained in Appendix A of WCAP-13854, referenced in Amendment Numbers 111 and 105 for Catawba Units 1 and 2, respectively. The enhanced inspection guidelines are intended to increase flaw detectability and reduce voltage variability such that consistent and accurate voltages and growth rates are obtained. Enclosed in Attachment I, for reference, is the current revision of the Byron/Braidwood Steam Generator Inspection Guidelines to illustrate the inspection methods used at Byron and Braidwood.

Historically, eddy current inspection requirements have been fairly consistent and have been routinely updated to include the latest industry techniques and analysis methods. During every refueling outage since startup, Byron has performed 100% bobbin coil inspections on the hot leg tubing through the U-bend and inspections of approximately 50% of the cold leg tubing. RPC inspections have been performed on each indication that could not be sized with the bobbin coil to assure proper repair dispositions. Use of the 3-coil RPC probe has been incorporated into the RPC inspection program. For the application of IPC, the Byron/Braidwood inspection guidelines have been updated to assure consistent test results and to reduce voltage variability. The following requirements have been incorporated into the Byron/Braidwood IPC inspection program:

- Use of ASME calibration standards cross-calibrated to a reference IPC laboratory standard.

- Use of probe wear standards to assess bobbin coil probe change outs at 15% or greater deviation from initial calibration amplitudes.

- Bobbin coil probe diameter is to be at least 0.610 inches.

- Data analysts are trained and tested to evaluate ODSCC defect sizing.

- Data analysts must pass a site specific performance demonstration. Whenever possible, Qualified Data Analysts (QDAs) examined by the industry EPRI performance demonstration program will be used. The data analyst pool of the current inspection vendor at Byron contains approximately 75% QDAs.

Eddy current bobbin coil inspections will be performed on 100% of the hot leg tubes through the U-bend and down to the lowest cold leg support plate intersection that has exhibited ODSCC in support of IPC. Rotating pancake coil (RPC) examinations will be performed on

- All indications greater than 1.0 volt and less than or equal to 2.7 volts.

- A minimum of 100 support plate intersections below 1.0 volt.

- All dented support plate intersections greater than 5.0 volts.

- Intersections with artifact indications, i.e., indications with unusual phase angles or intersections with large interference signals that could mask a 1.0 volt defect.

Whenever possible, a 3-coil RPC probe will be used to determine degradation orientation. Notification to the Staff is required prior to startup of any unexpected RPC results relative to the assumed characteristics of flaws at tube support plates. This includes reporting any detectable circumferential indications or detectable indications extending beyond the thickness of the tube support plate. Indications with bobbin coil voltages greater than 1.0 volt but less than or equal to 2.7 volts may remain inservice if degradation not found with RPC. Indications of ODSCC degradation greater than 2.7 volts will be plugged or repaired.

Safety-related work associated with steam generator repair at Byron Station is conducted in accordance with the Quality Assurance Program requirements identified in Appendix B to 10 CFR Part 50. Byron Station's Site Quality Verification (SQV) group typically provides an independent oversight of steam generator inspection and repair activities during refueling outages. Field Monitoring Reviews are conducted on steam generator activities, including eddy current testing and tube repairs activities.

INSPECTION RESULTS

The eddy current data from the 530 tubes that were plugged due to ODSCC from the previous outage (Cycle 5 refueling outage in the Spring of 1993) were reanalyzed using the IPC guidelines described in **INSPECTION CRITERIA** above to determine a voltage distribution. The indications found during the Cycle 5 refueling outage were traced back to the Cycle 4 refueling outage and reanalyzed to obtain a growth rate for Cycle 5. Figures B-1 and B-2 show the voltage and growth rate results for Byron Unit 1 Cycle 5. The average growth rate of the plugged tubes for Cycle 5 was 0.31 volts over the entire cycle, or 0.28 volts/EFPY, with the largest single growth rate being 2.4 volts over the cycle (2.13 volts/EFPY) and all other growth rates being less than 1.8 volts over the cycle. Likewise, the largest single voltage amplitude detected was 2.6 volts. All other indications were less than 2.2 volts. These results are summarized in Table B-1 below:

Parameter	Voltage for Cycle 5	Voltage per EFPY
Average Growth Rate	0.31	0.28
Maximum Growth Rate	2.4	2.13
Average Voltage	0.72	n/a
Maximum Voltage	2.6	n/a

Table B-1
Byron Unit 1 Cycle 5 Plugged Tube Data

The Byron Unit 1 Cycle 5 growth rates and amplitudes for plugged tubes are consistent with industry data described in the EPRI PWR Steam Generator Tube Repair Limits - Technical Support Document for ODSCC at Tube Support Plates, TR-100407 Revision 1. This data indicates that Byron Unit 1 ODSCC tube condition is supportive of an IPC since the data is comparable to other plants with an approved IPC.

Figures B-3A through B-9B of this Attachment show examples of ODSCC bobbin and RPC results from the Byron Unit 1 Cycle 5 inspection. WCAP-14046, Section 3.0, Figures 3-1 through 3-5 and Figures 7-30 through 7-48, show examples of confirmed industry ODSCC and Braidwood Unit 1 inspection results. Comparison of typical Byron Unit 1 inspection results with the results from other plants that have axial ODSCC degradation confirmed through tube pulls, indicates that the Byron Unit 1 degradation is consistent with known ODSCC.

A sample of degraded tube intersections and a sample of non-degraded tube intersections from Byron Unit 1 will be removed during the next refueling outage scheduled for September of 1994. The tube pull results will confirm the crack morphology for degradation at the tube support plates and provide additional data points for leak and burst correlations.

STEAM GENERATOR TUBE INTEGRITY

The purpose of the Technical Specification repair limit is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions, consistent with General Design Criteria 14, 15, 31, and 32 of 10 CFR Part 50, Appendix A. Structural integrity is defined as maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes" requires a structural margin of safety of 1.43 against tube failure under postulated accident conditions and a margin of safety of 3 against burst during normal operation. The proposed repair limit for Byron Unit 1 involves the implementation of the voltage based eddy current signal amplitude of 1.0 volt for ODSCC occurring within the thickness of the tube support plates. This is a change from the traditional depth-based criteria of 40% which is currently required by Byron Technical Specifications. The proposed IPC for Byron Unit 1 meets the requirements of Regulatory Guide 1.121 by demonstrating that tube leakage is acceptably low (resulting in offsite doses that are a small fraction of 10 CFR 100 limits) and tube burst is a highly improbable event during normal operation as well as during a postulated MSLB event.

The Byron Unit 1 methods for determining the structural and leakage tube integrity for normal operating and MSLB conditions is described in WCAP-14046, with the following exceptions:

- Tube burst evaluations for MSLB conditions assume that the degradation is free span and the constraining effects of the tube support plates are not considered. Therefore, the tube support plate limited displacement analysis is not used for Byron Unit 1.

- The reactor coolant dose equivalent iodine-131 limit is not reduced.

- A POD of 0.6, in accordance with draft NUREG-1477, is factored into the determination of the beginning of cycle (BOC) and end of cycle (EOC) distributions for tube burst and leakage assessments.

Sections 5.0 and 6.0 in WCAP-14046 describe in detail the methodology of assessing tube structural and leakage integrity. Sections 7.0 and 8.0 of the WCAP demonstrate implementation of the methodology by assessing the Braidwood Unit 1 EOC-4 results. A brief summary of the contents of Sections 5.0 and 6.0 of WCAP-14046 and the Byron approach follows.

The approach described in WCAP-14046 addresses tube structural and leakage integrity at normal operating and MSLB conditions by correlating bobbin coil voltage amplitude data to 1) tube burst pressure and burst probabilities and 2) leakage rates and leakage probabilities at normal and MSLB conditions for the BOC and EOC voltage distributions. The database supporting the alternate repair criteria burst and leak rate correlations for 3/4" tubing is described in EPRI Report NP-7480-L, Volume 2. Section 5.0 of WCAP-14046 describes the database and application of the guideline for removal of data outliers from the burst and leak correlations used in support of the IPC.

Application of IPC begins with acquisition of eddy current data using the IPC guidelines described in **INSPECTION CRITERIA** above and generating a voltage distribution. Growth rates are determined by comparing voltage amplitudes of the current and previous outage eddy current data. The BOC distribution is generated by applying a POD of 0.6 as described in draft NUREG-1477. The predicted EOC distribution is determined by increasing the BOC voltages by allowances for non-destructive examination (NDE) uncertainties and voltage growth through a Monte Carlo simulation. These BOC and EOC distributions are used to assess the probability of burst, probability of leakage and overall leak rates for normal operation and MSLB conditions. A degree of conservatism is inherent in these distributions due to the use of a constant POD of 0.6. Data collected by the EPRI Performance Demonstration Program indicates that the POD is larger than 0.6 for indications greater than 1.0 volt and increases with voltage amplitude.

For normal operating conditions, tube burst is precluded due to the constraining effects of the support plate in the area of the ODSCC affected portion of the tube. However, during a MSLB event, support plate displacement may expose a portion of a crack to free span conditions and increase the probability of burst. Accident burst assessments are conservatively determined assuming the cracks are located entirely in the free span. The cumulative probability of burst for all indications left inservice, considering a POD of 0.6, must demonstrate a probability of less than 2.5×10^{-2}/reactor year, in accordance with NUREG-0844, over the entire cycle of operation. The EPRI database in the report for 3/4" tubing serves as the basis for the log-linear relationship between burst pressure and bobbin voltage. Using the lower 95% confidence level of the burst pressure and bobbin voltage correlation, the bobbin voltage corresponding to the free span structural limit is 4.54 volts for a

burst pressure of 3657 psi. This burst pressure corresponds to the MSLB differential pressure with a 1.43 safety margin consistent with Regulatory Guide 1.121.

The steam generator MSLB accident tube leakage assessment consists of a probability of leakage model and a model predicting leak rate as a function of voltage, assuming a leak occurs. The probability of leakage (POL) model is based on a single function form, which is the log-logistic. The use of the log-logistic function form for IPCs was determined to be acceptable by the Staff in the May 7, 1994 SER for Braidwood Unit 1. Any non-conservatism which may be associated with the use of the log-logistic function as compared to other function forms is small in comparison to the conservatisms inherent in the existing methodology for estimating the radiological consequences of leakage induced by a postulated MSLB.

The leak rate versus voltage correlation is based on a linear regression fit of the logarithms of the corresponding leak rate and voltage data (known as the conditional leak rate model). The linear regression fit for the leak rate versus voltage correlation was deemed to be valid by the NRC in guidance from the February 8, 1994 NRC/industry meeting when a p-value test result of less than 5% can be demonstrated. WCAP-14046, Section 6.6 determined that the conditional leak rate model p-value test result was significantly less than 5%, thus demonstrating a valid correlation. The overall MSLB leak rate is obtained by applying the POL correlation and the conditional leak rate correlation to the POD adjusted EOC voltage distribution. An upper bound 95% confidence limit is factored into the final leak rate value. The final overall leak rate must be less than the maximum site allowable leak rate.

The normal operational tube leakage is implicitly assured by the allowable limits on the operational leak rate as specified in the Byron Technical Specifications. The primary-to-secondary leak rate limit through any one steam generator has been reduced from 500 gpd to 150 gpd with an overall limit of 600 gpd through all 4 steam generators. This reduction has been implemented administratively pending approval of the Byron 06/03/94 submittal incorporating these limits into the Technical Specifications in support of sleeving. The 150 gpd limit in support of IPC is based on the guidance provided in draft NUREG-1477 and the EPRI Technical Report for ODSCC (TR-100407 Rev 1). These limits are based on the ability to detect primary-to-secondary leakage at normal operating conditions which could potentially develop into a tube rupture during faulted plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of a leak from an unexpected single crack associated with the longest permissible free span crack length. The longest permissible crack is the length that provides a factor of safety of 1.43 against burst at faulted conditions. Alternate crack morphologies can correspond to the structural voltage limit of 4.54 volts, so a

unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, burst pressure versus through-wall crack length correlations are used to define the longest permissible crack for evaluating leakage limits. This evaluation, discussed in Section 8.3 of WCAP-14046, demonstrates that the 150 gpd limit provides capability of plant shutdown prior to reaching critical crack lengths for MSLB conditions at leakage below the 95% confidence level and for the more restrictive 3.0 times normal operating pressure differential at less than nominal leak rates. During normal operating conditions, tube burst is precluded due to the proximity of the support plate to the area of degradation. Therefore, with the reduced limit and the leak rate monitoring program described in Attachment D to this submittal, reasonable assurance that a significant leak could be detected and the appropriate operator actions would occur in a timely manner are provided.

TUBES NOT APPLICABLE TO IPC

For a combined seismic event (SSE) and loss of coolant accident (LOCA) condition, designated LOCA + SSE, the potential exists for yielding of the tube support plate in the vicinity of the tube support plate-to-wrapper wedge locations. This deformation may lead to opening of pre-existing tight through-wall cracks or propagation of pre-existing non-through-wall cracks that would result in primary-to-secondary leakage. Therefore, tubes located in these susceptible areas are excluded from IPC consideration. The tubes susceptible to collapse during a LOCA + SSE event are identical for Byron Unit 1 and Braidwood Unit 1 since both units contain Model D-4 steam generators with similar configurations. The tube lists and tube maps contained in Section 4.7 of WCAP-14046 are applicable to both Byron Unit 1 and Braidwood Unit 1.

SITE ALLOWABLE LEAKAGE LIMIT

Purpose And Scope

The purpose of this calculation is to generate the maximum allowable end of cycle primary-to-secondary steam generator leak rate and to maintain the offsite dose within a small fraction of 10CFR100 design criteria during a postulated main steam line break. The methodology used in the evaluation is consistent with WCAP-14046, "Braidwood 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May, 1994. ICRP 30 thyroid inhalation dose conversion factors were used to determine Exclusion Area Boundary dose, consistent with WCAP-14046 and with the May 7, 1994, NRC SER approving IPC for Braidwood.

The evaluation was performed for both a pre-accident and accident initiated iodine spike. Only the release of iodine and the resulting thyroid dose at the Exclusion Area Boundary was considered in the leak rate determination. Whole body and skin doses due to noble gas immersion have been determined in other evaluations to be less limiting than the corresponding thyroid dose.

The evaluation was based on an acceptance criteria of 30 rem thyroid dose at the Exclusion Area Boundary per the Standard Review Plan (NUREG 0800) Section 15.1.5, Appendix A. For the preaccident iodine spike, the initial primary coolant activity was 60 μCi/gm dose equivalent Iodine 131 (I 131), and for the concurrent spike the activity was 1 μCi/gm. The secondary coolant activity was 0.1 μCi/gm I 131. The calculation of primary coolant dose equivalent I 131 is based on a mixture of 5 iodine nuclides (I 131 through I 135) and the dose conversion factors of TID-14844, consistent with the Byron Technical Specification definition of dose equivalent iodine. The leak rate in the three intact steam generators was assumed to be the proposed Technical Specification limit of 150 gallons per day (about 0.1 gpd) in each generator. The leak rate from the reactor coolant system was 1 gpm. The dose model, breathing rate, and dispersion factor were obtained from the Byron/Braidwood Updated Final Safety Analysis Report.

The activity released to the environment due to a main steam line break was analyzed in two distinct releases: 1) the release of the iodine activity that has been established in the secondary coolant prior to the accident, and 2) the release of the primary coolant iodine activity due to tube leakage. The release of the activity initially contained in the secondary coolant (4 SGs) results in a site boundary thyroid dose of approximately 1.13 rem. The dose contribution from primary to secondary tube leakage of all four SGs (1.0 gpm) is approximately 2.24 rem. Applying the 30 rem thyroid dose limit, the total allowable primary-to-secondary leak rate is (30 rem - 1.13 rem) / 2.24 rem/gpm = 12.8 gpm. Allowing 0.1 gpm (150 gpd) per each of the 3 intact SGs, results in 12.5 gpm for the steam generator on the faulted loop.

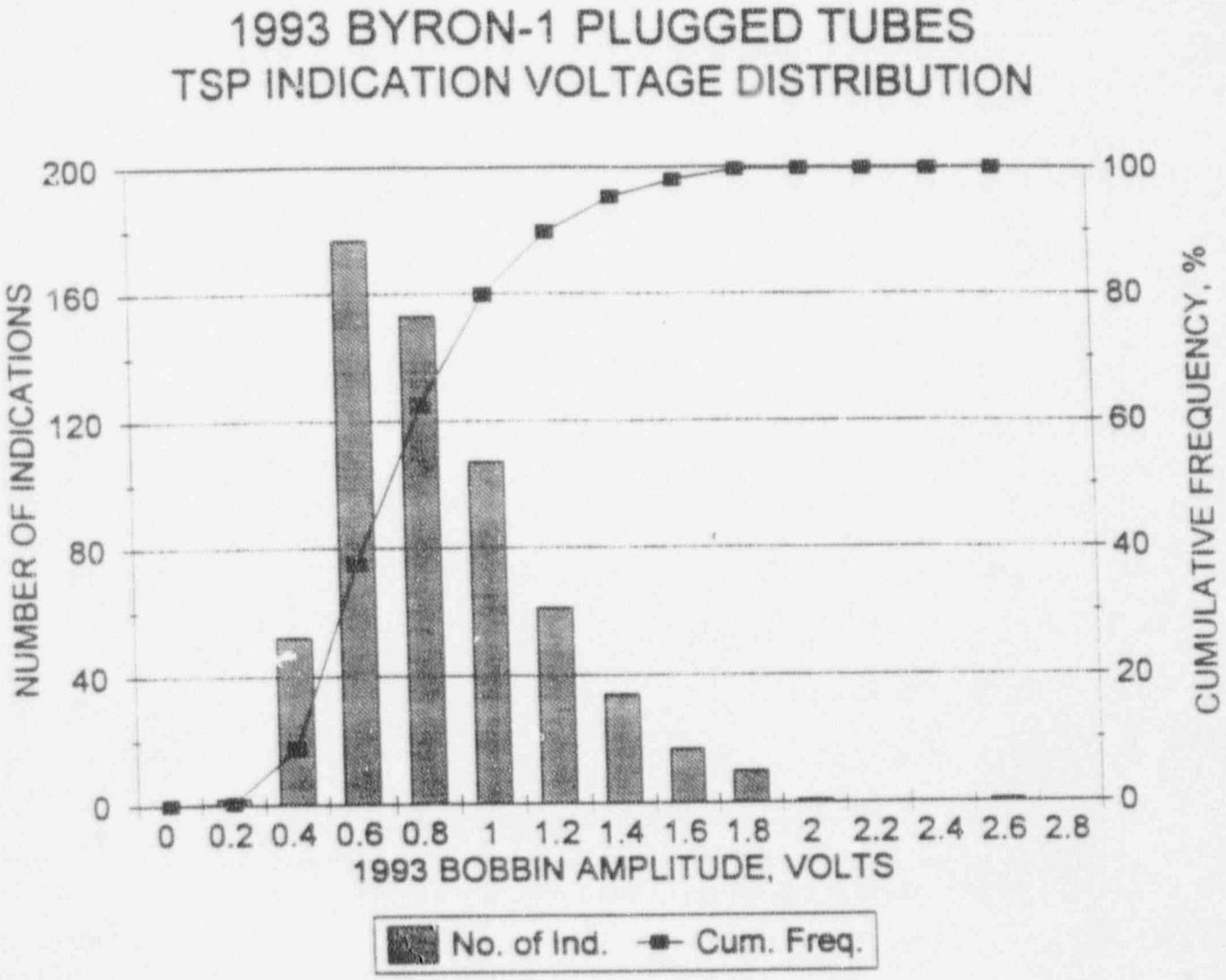


FIGURE B-1: Byron Unit 1 End of Cycle 5 Voltage Distribution (Plugged Tubes Only)

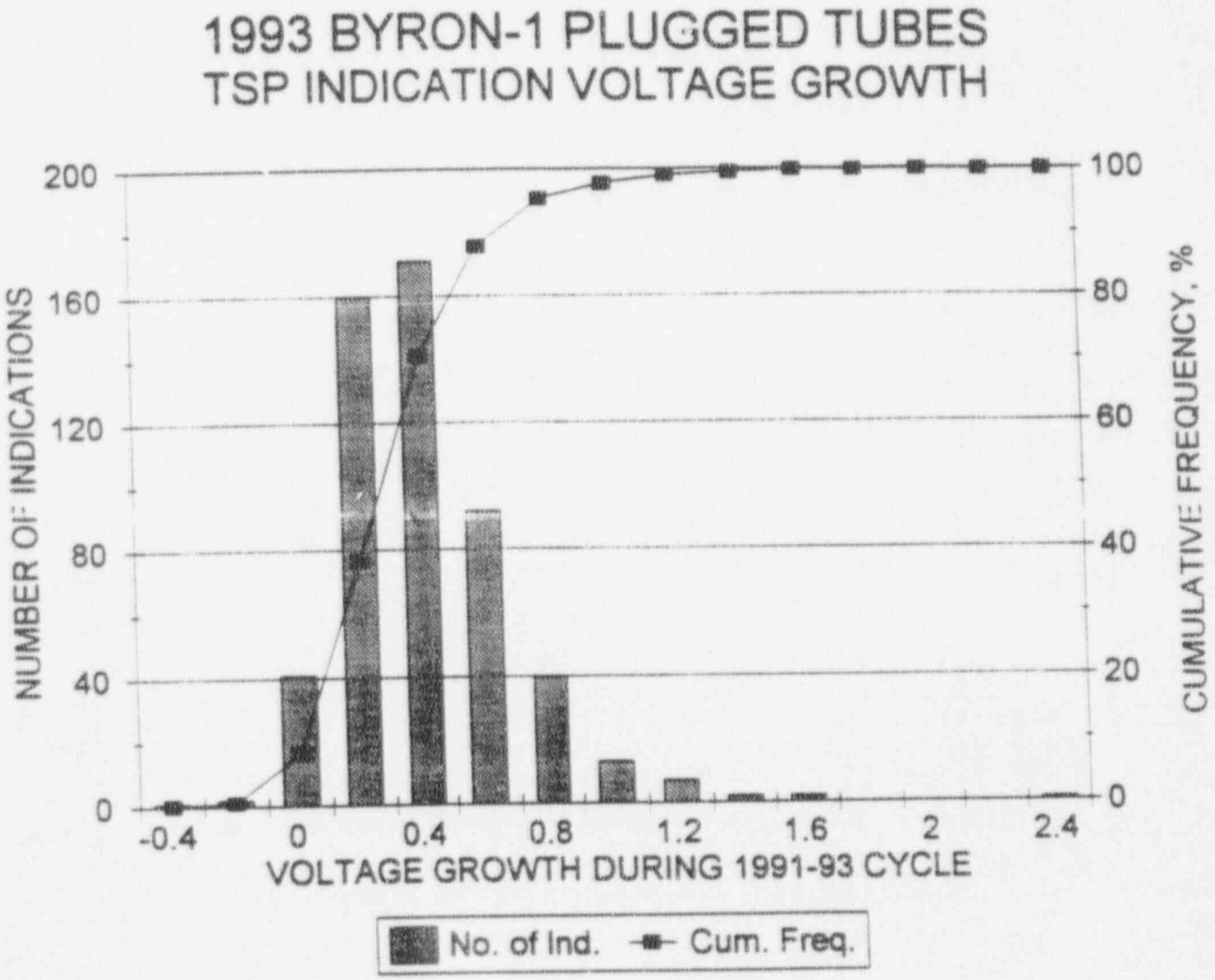


FIGURE B-2: Byron Unit 1 End of Cycle 5 Growth Rate Distribution (Plugged Tubes Only)

FIGURE B-3A: SG A, Row 24 Col 38, Bobbin Coil

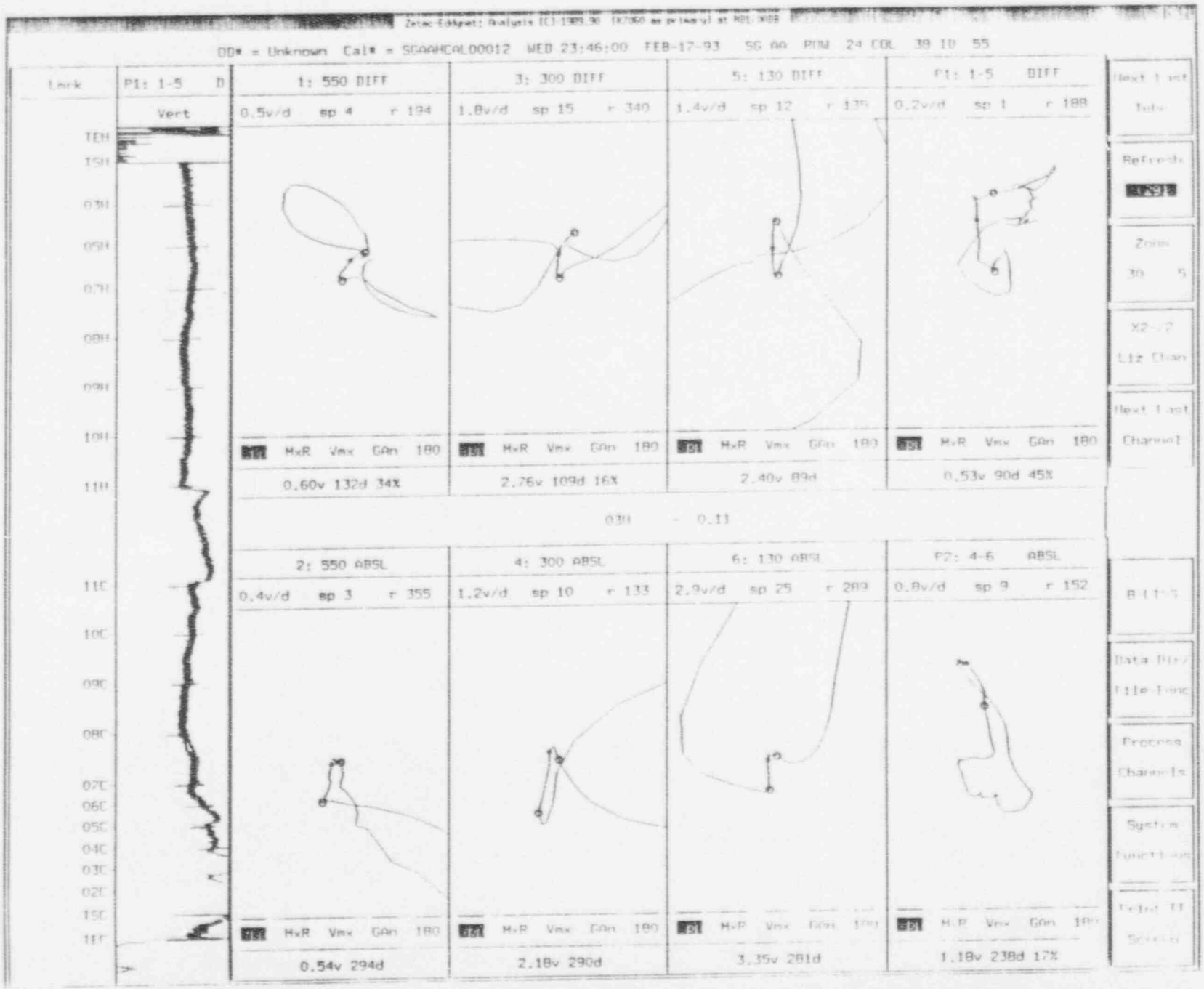


FIGURE B-3B: SG A, Row 24 Col 38, Axial Sensitive RPC Coil

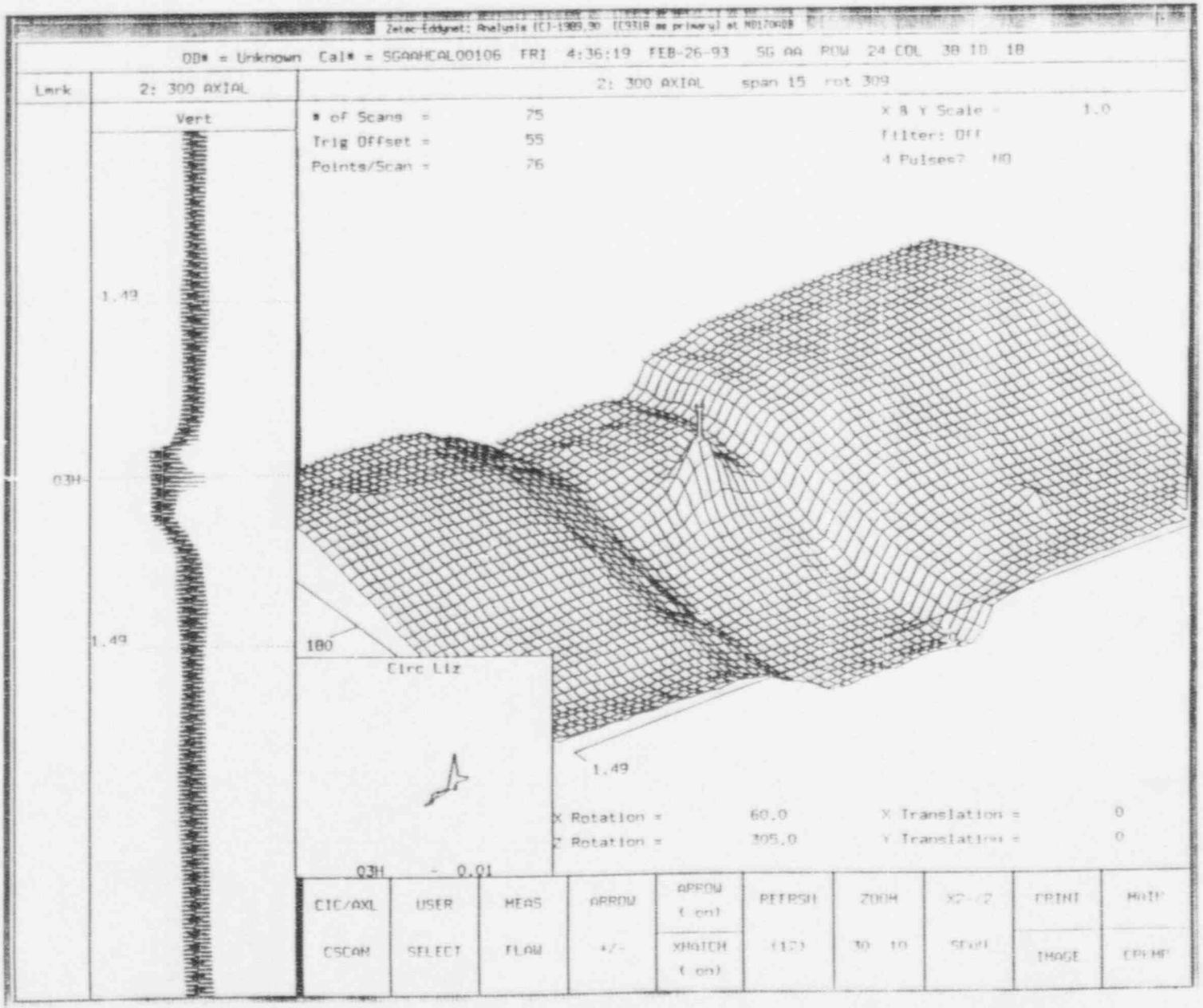


FIGURE B-4A: SG B, Row 9 Col 103, Bobbin Coil

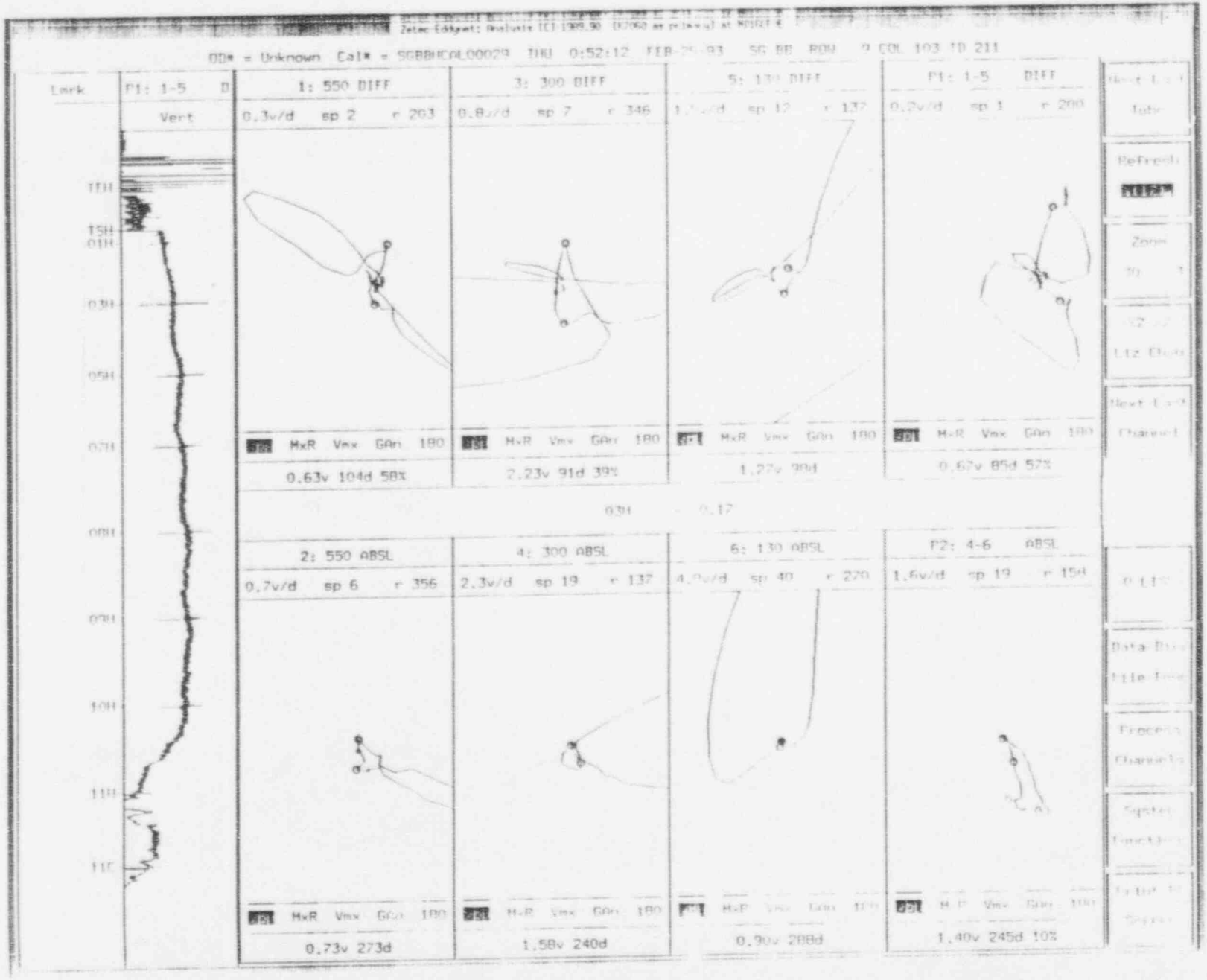


FIGURE B-4B: SG B, Row 9 Col 103, Axial Sensitive RPC Coil

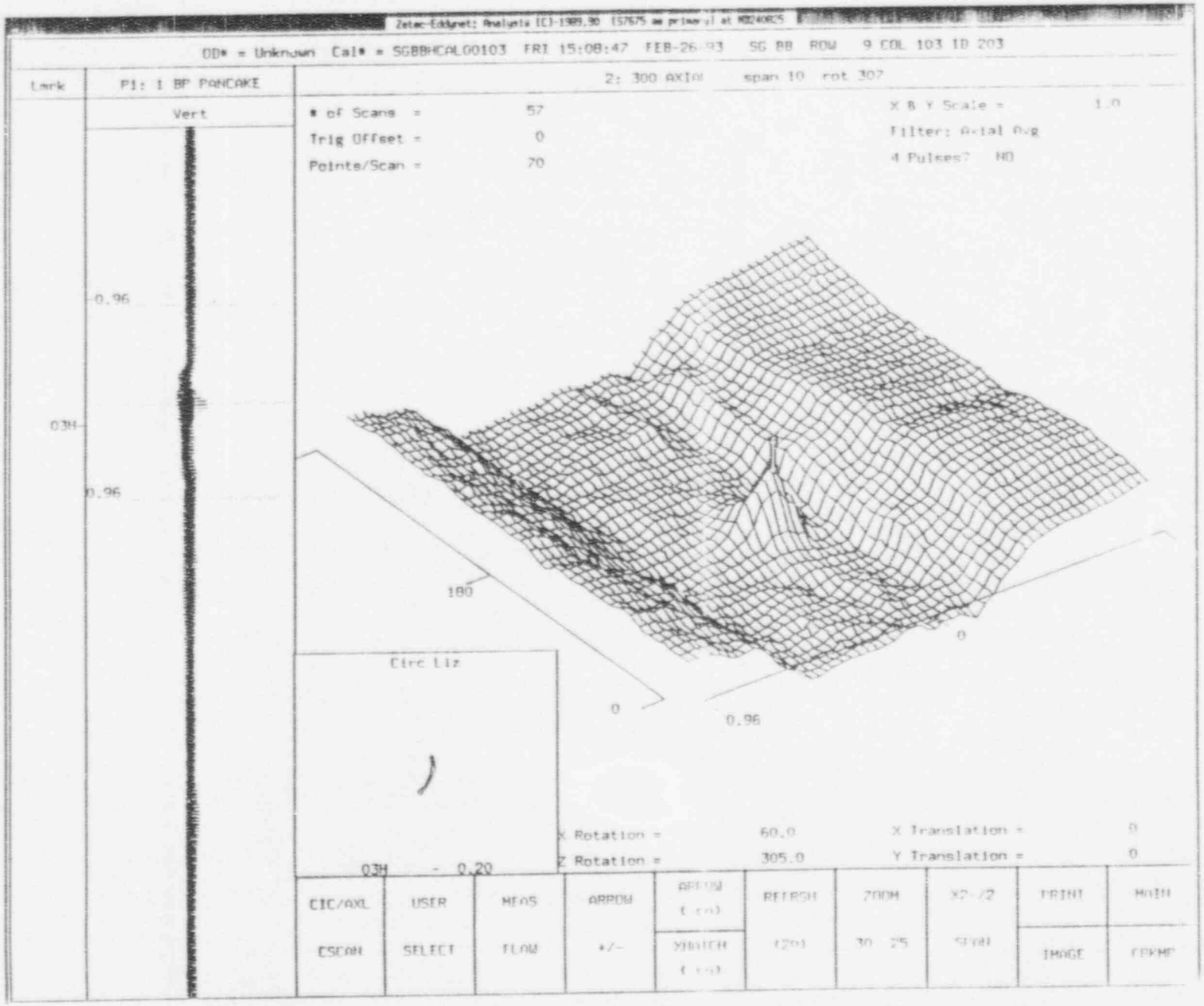


FIGURE B-5A: SG B, Row 1 Col 81, Bobbin Coil

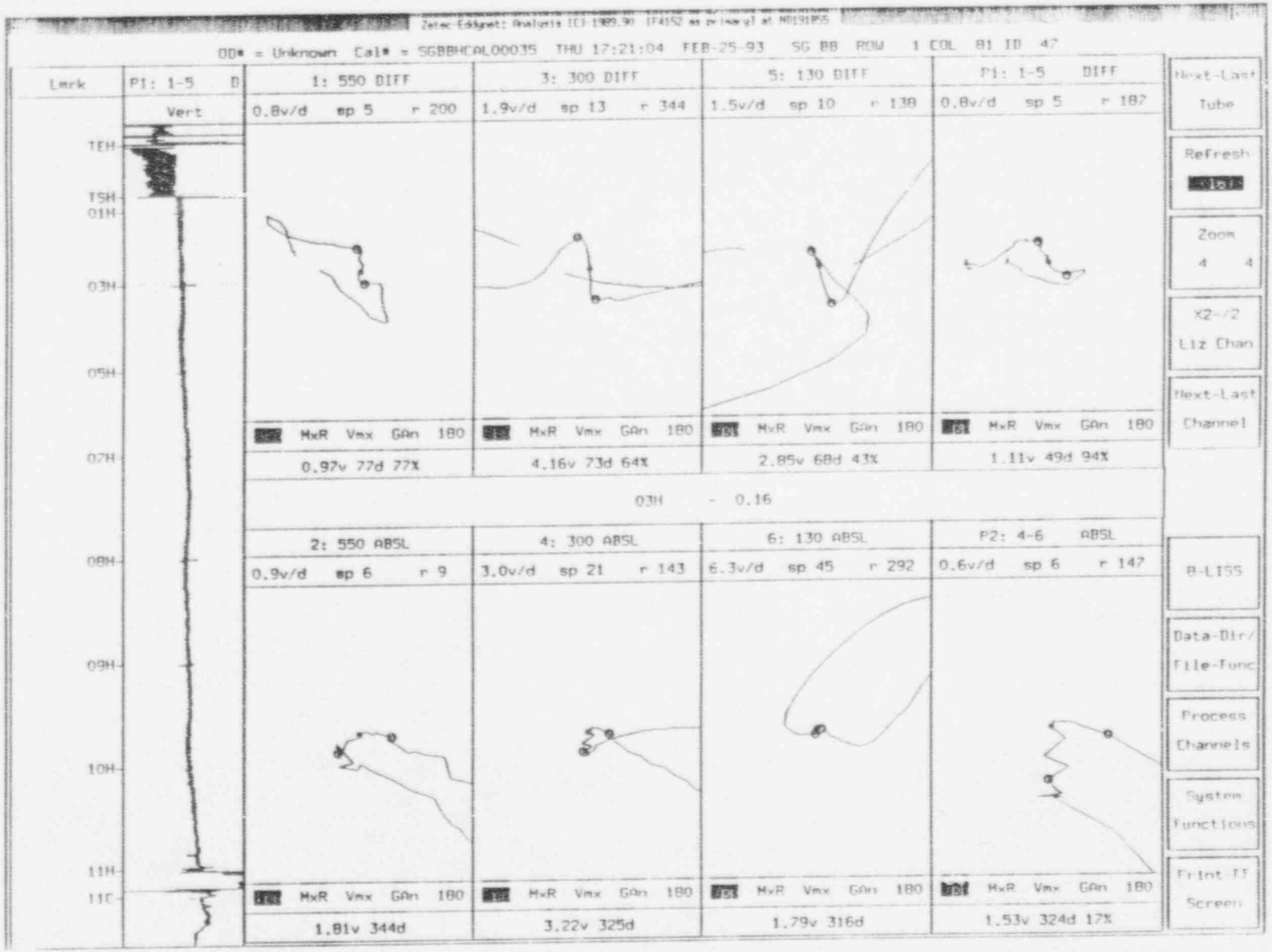


FIGURE B-5B: SG B, Row 1 Col 81, Axial Sensitive RPC Coil

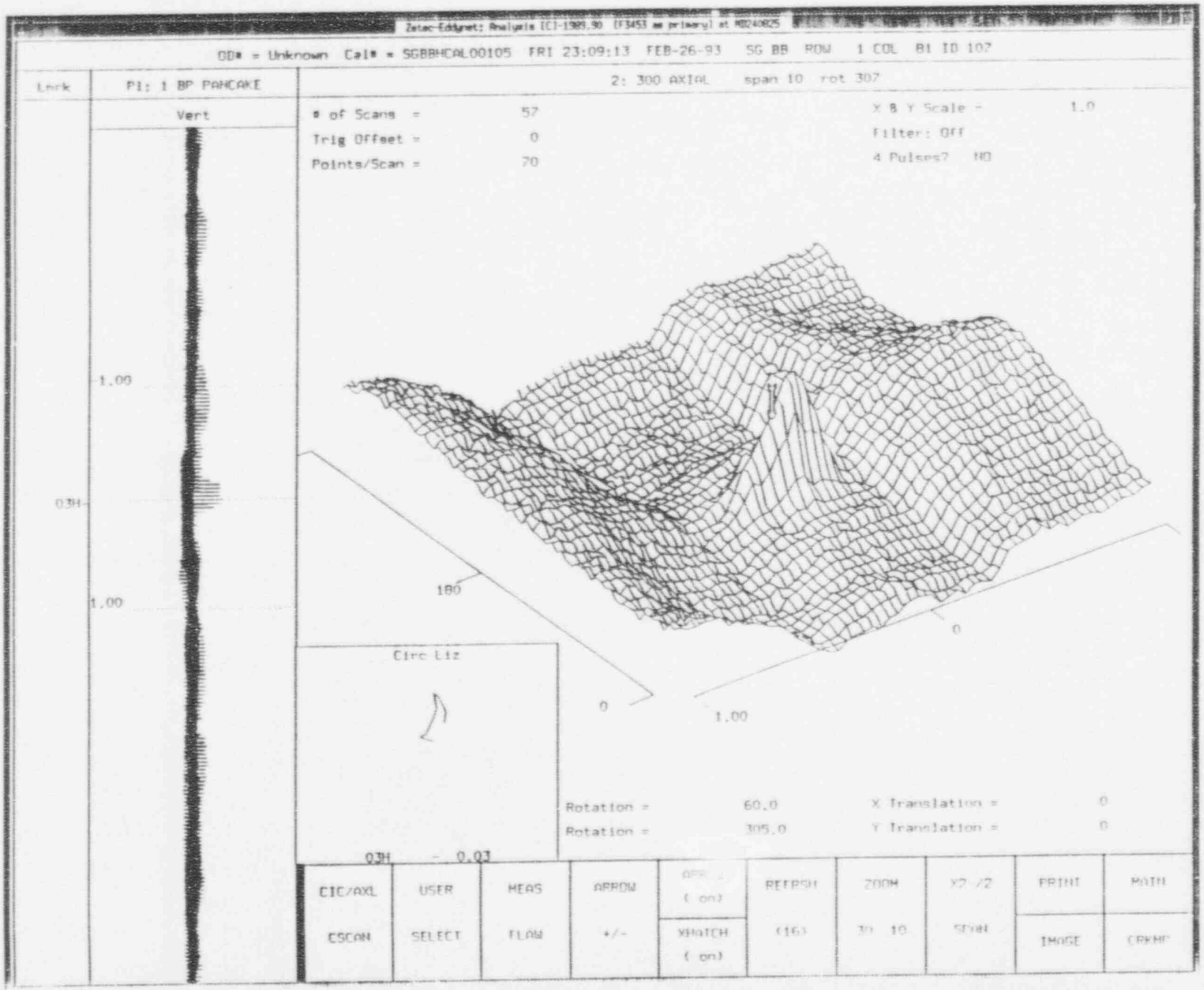


FIGURE B-6A: SG B, Row 42 Col 25, Bobbin Coil

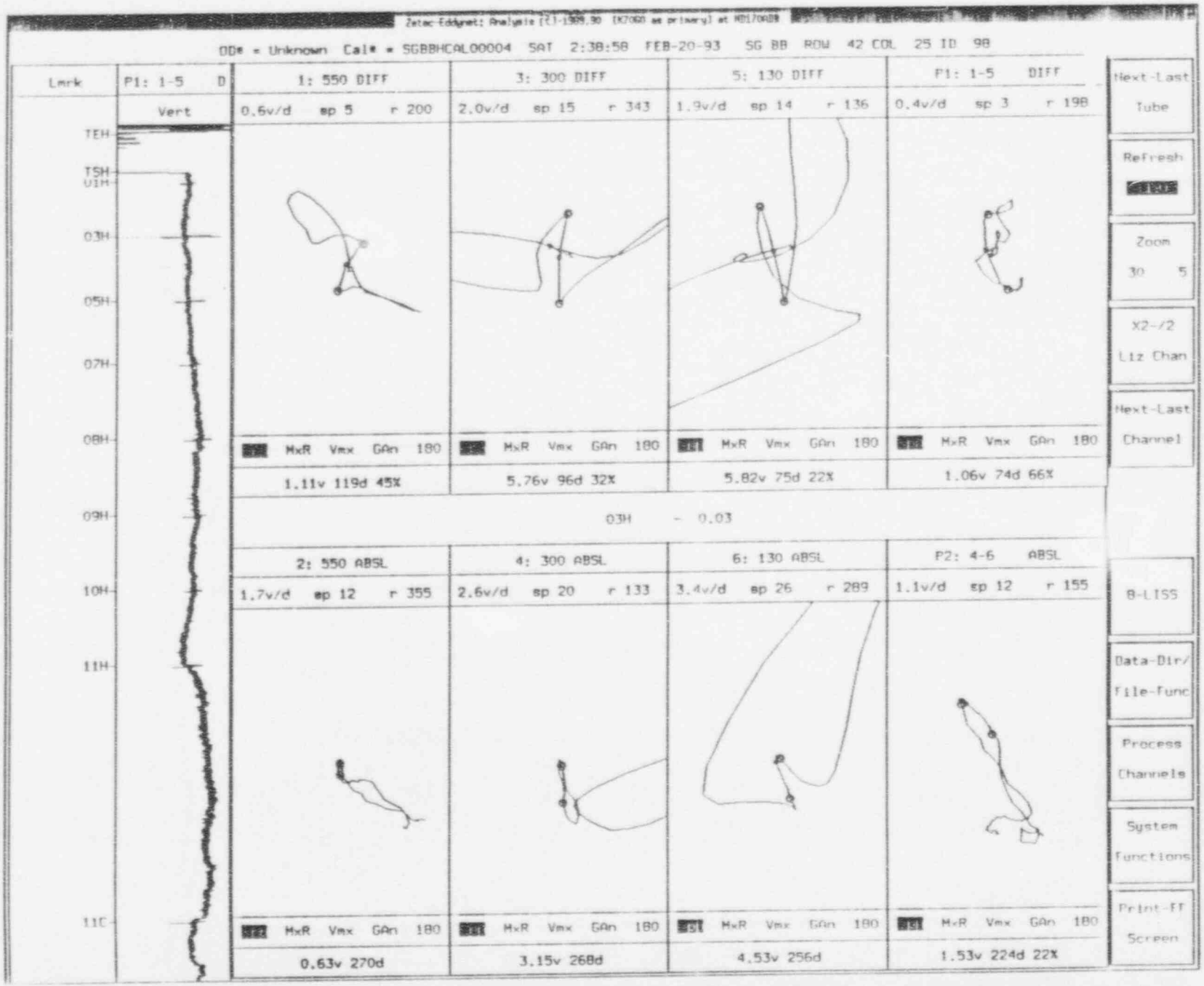


FIGURE B-6B: SG B, Row 42 Col 25, Axial Sensitive RPC Coil

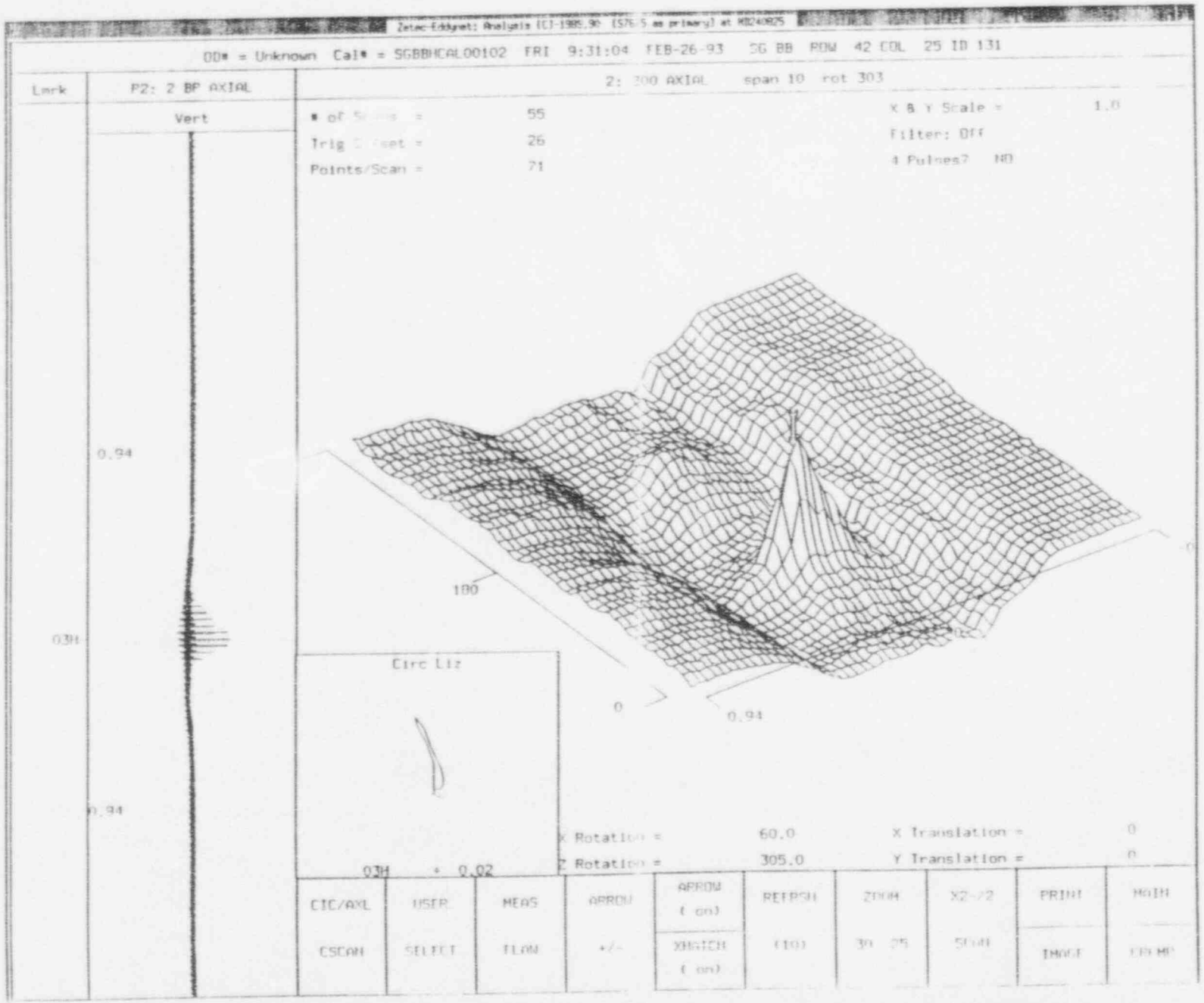


FIGURE B-7A: SG C, Row 19 Col 49, Bobbin Coil

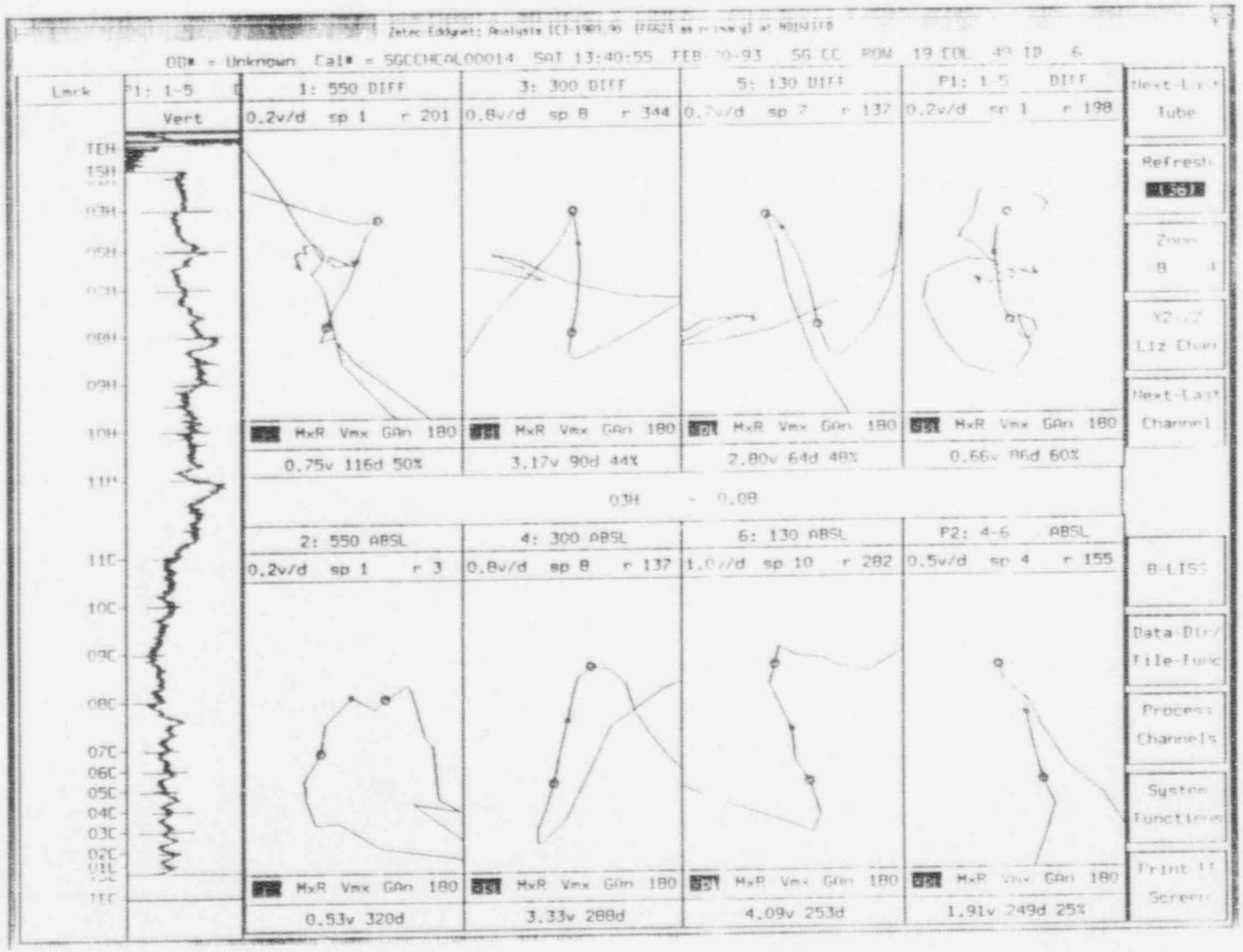


FIGURE B-7B: SG C, Row 19 Col 49, Axial Sensitive RPC Coil

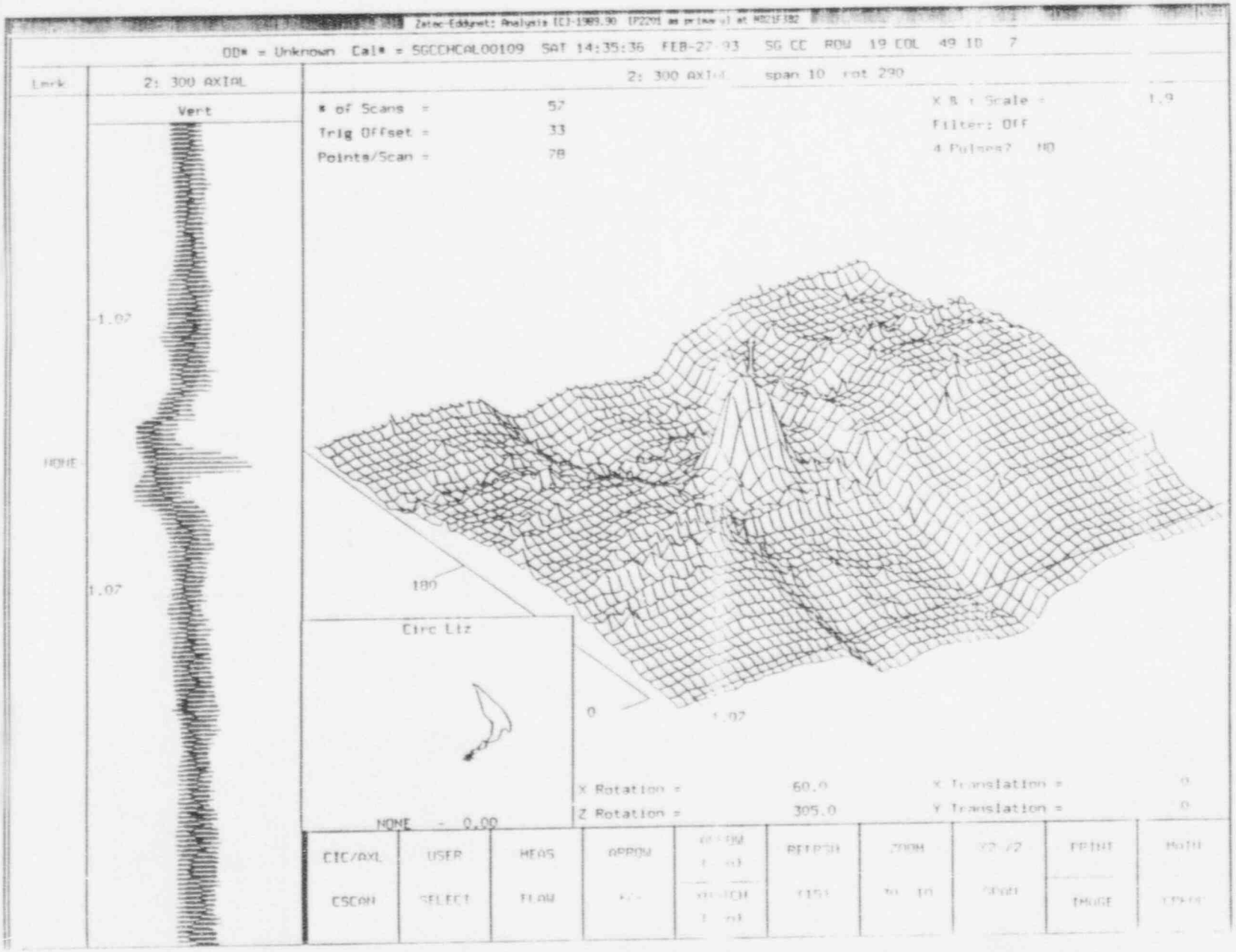


FIGURE B-8A: SG C, Row 2 Col 8, Bobbin Coil

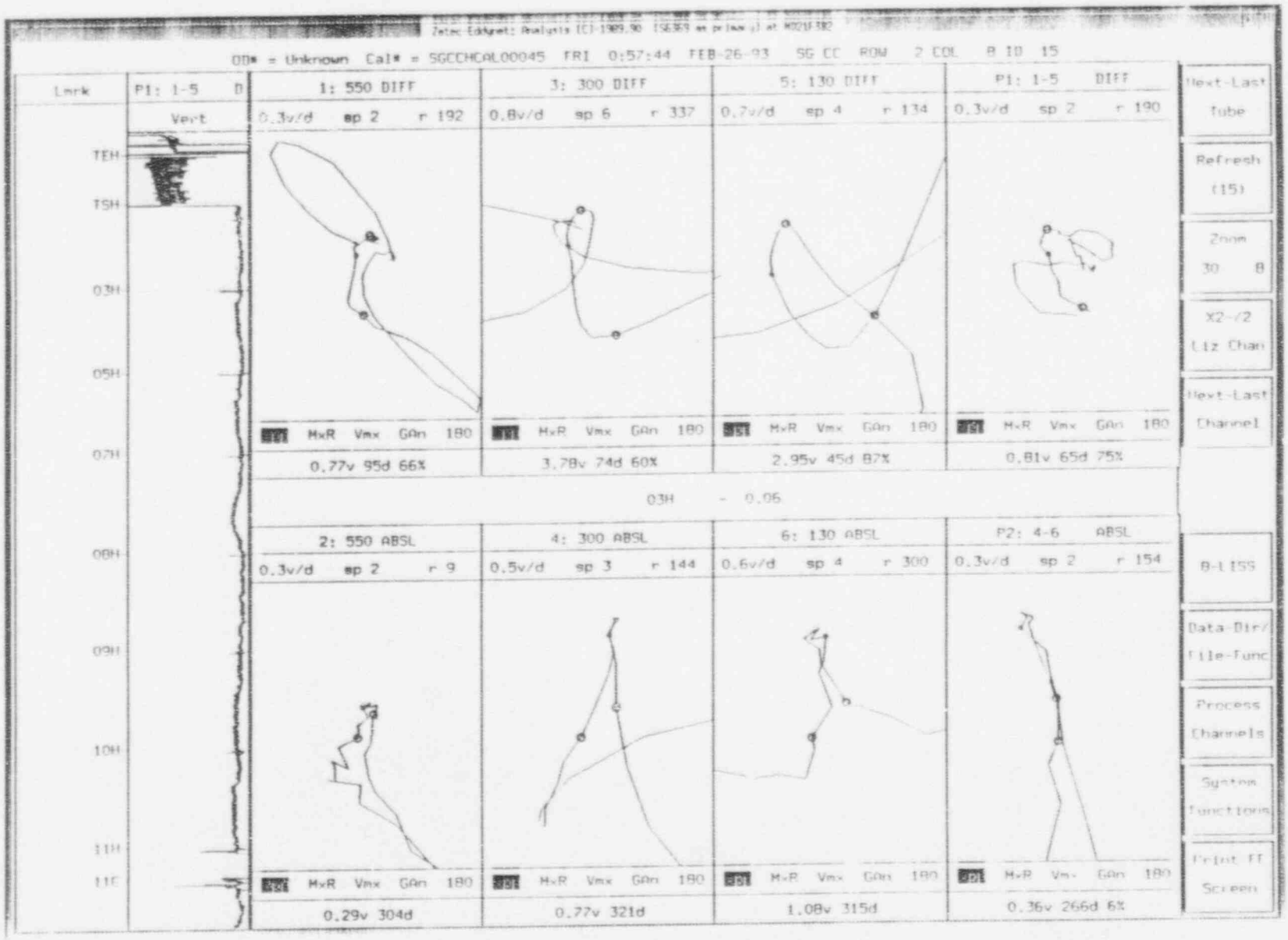


FIGURE B-8B: SG C, Row 2 Col 8, Axial Sensitive RPC Coil

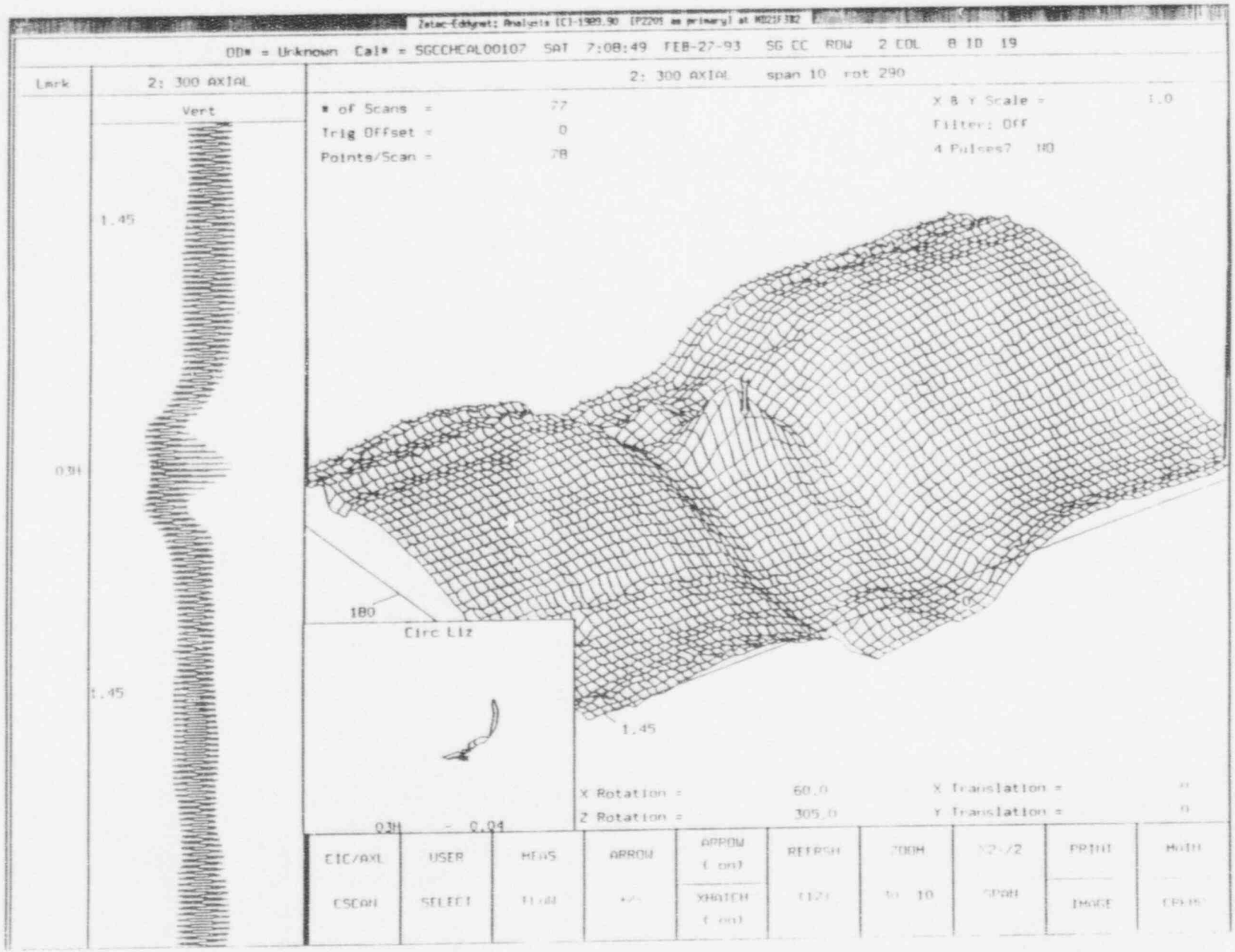


FIGURE B-9A: SG D, Row 5 Col 103, Bobbin Coil

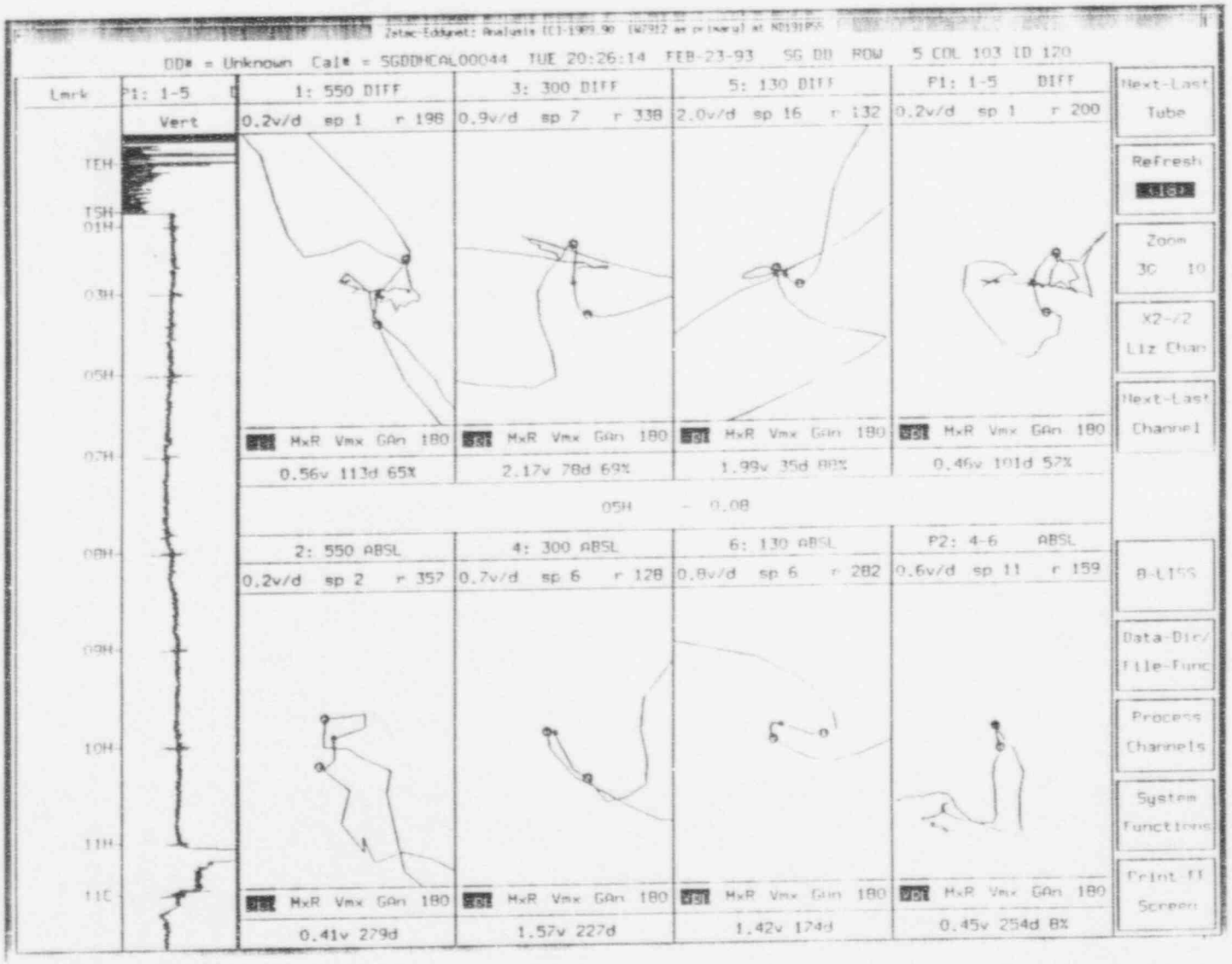
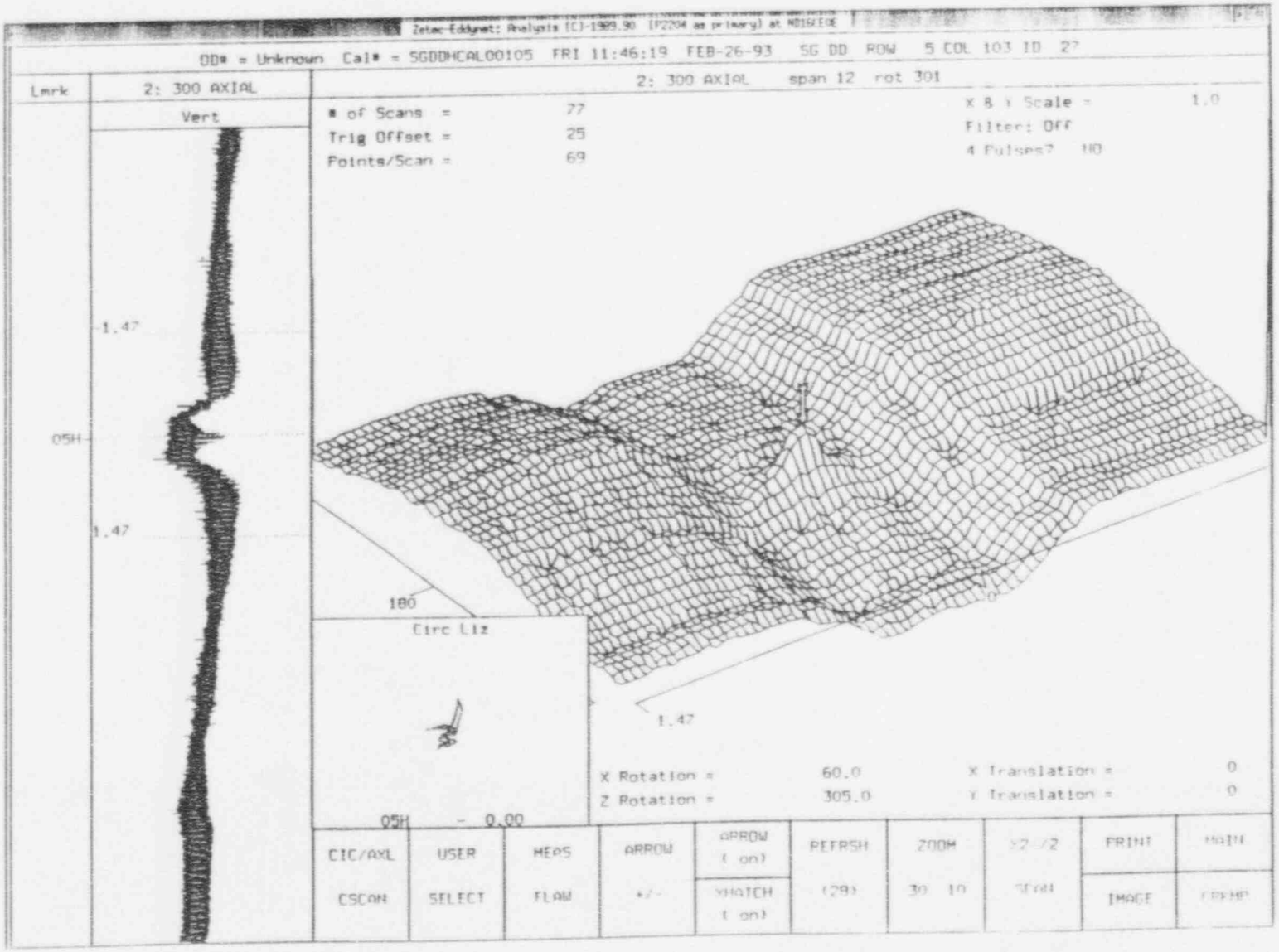


FIGURE B-9B: SG D, Row 5 Col 103, Axial Sensitive RPC Co''



ATTACHMENT C

BYRON STATION
CORRECTIVE ACTIONS TO ADDRESS
STEAM GENERATOR
DEGRADATION

OPERATING CHEMISTRY

Byron Station is currently in its sixth cycle and has operated in a load following capacity. These power changes have resulted in secondary chemistry conditions that change throughout the day. The steam generator bulk water contaminant concentrations will vary based on current power levels. The attachments included indicate reactor power (Figures C-1 to C-4), sodium and chloride concentrations (Figures C-5 to C-8), and sodium to chloride ratio (Figures C-9 to C-12) for Cycles 3 through 6.

During Cycle 3, the sodium concentration in the bulk water exceeded that of the chloride and remained elevated through Cycle 6 (See Figures C-5 to C-8). This was due in part to work performed in the moisture separator reheaters (MSR). Sandblasting was performed to prepare the inner surface during installation of baffle plates which left a residual source of sodium. It has also been observed that dry out areas are occuring within the MSRs. As the dry out areas become rewetted when flows are changed, the sodium concentration in the secondary increases. Other sources of sodium ingress include condenser leakage, demineralizer performance, and routine maintenance activities. The elevated sodium levels can lead to formation of caustic tube support plate (TSP) crevices which are detrimental to alloy 600 material. Laboratory tests have shown that highly caustic crevices can promote initiation and propagation of stress corrosion cracking (ODSCC). Therefore, to mitigate ODSCC, several of the sodium sources have been addressed to minimize the secondary contaminant ingress.

A full main condenser eddy current was performed on 100% of the tubes during refueling outage B1R05 to address the low level leakage problems. Based on the eddy current results, several condenser tubes were preventatively plugged to mitigate condenser leakage. Selected waterboxes will be examined during subsequent outages.

The makeup demineralizer operation was optimized with the use of modified regeneration and operating procedures and resulted in reduced contaminant ingress. After MSR and other plant equipment work, extensive inspections are performed for cleanliness to further limit contaminants. With these program enhancements, the sodium concentration remained slightly above the chloride concentration due to the low levels of chloride (typically below the detectable threshold) normally residing in the secondary plant.

In December 1993, Byron Station implemented a molar ratio control program to correct the sodium to chloride imbalance. This program only addressed sodium and chloride. There is some degree of industry uncertainty as to which chemical species have a significant effect in this crevice chemistry. Calculations show that potassium and fluoride, among others, can possibly change the theoretical pH. This relationship is currently being evaluated at Byron Station. Molar ratio control involves injecting a dilute solution of ammonium chloride into the secondary side to maintain an operating concentration ratio of 0.4 - 0.6 sodium to chloride (Na/Cl) (see Figure C-13). This will theoretically lower the steam generator flow crevice Na/Cl molar ratio to 1 which is desired. Hideout return reports from past outages were used to determine the proper operating regime. The Multeq computer program was used to determine the theoretical crevice pH. A hideout return evaluation will be performed at the beginning of the upcoming refueling outage, B1R06, to determine the effectiveness of balancing the crevice pH. By also using the Multeq program, it was observed that crevices packed with contaminates created a more caustic environment (see Figure C-13). Figure C-13 also includes the Na/Cl molar ratio (above each point), which is based on hideout return data. It should be noted that the Na/Cl ratio varies significantly from shutdown to shutdown, and these data points only take into consideration sodium and chloride. A comparison of Eddy current data from B1R04 with the previous inspection, tends to support the theory that the steam generator support plate crevices are packed.

During B1R05, a video inspection was performed to determine the condition of the secondary side. It was observed that the crevice area was packed and that the tube surfaces were fouled. A high temperature chemical cleaning will be performed during B1R06 to remove fouling on the tube surfaces and return the crevices to their original condition.

The high temperature cleaning process to be deployed at Byron is being developed with the intent of removing crevice deposits. This process will include a 15% solution of EDTA with a corrosion inhibitor. The steam generators will be left at a temperature of 300°F and a pressure of 70 psig. Boiling will be caused by opening the power operated relief valves (PORVs) and depressurizing the steam generators to create an agitation effect at the TSPs. This step will be performed several times at different water levels. By creating this boiling action at several water levels, the solution will be allowed to interact at the support plates, thereby removing crevice deposits. A video inspection will be performed at the completion of the cleaning. This, as well as the eddy current analysis, will be used to determine the possible overall effectiveness of the chemical cleaning process.

The molar ratios over the past cycles were slightly elevated in the caustic region. Ammonium chloride injection was started to correct this problem. After completion of the chemical cleaning process, eddy current analysis, and evaluation of hideout return data, the molar ratio control program will be revaluated as to its effectiveness, and possible application in the future.

The addition of ethanolamine (ETA) was also started on both units (Unit 1 in July of 1993, and Unit 2 in November of 1993) to minimize the amount of iron transported to the steam generators. This program has been very effective on both units by reducing the feedwater iron concentrations from 10 - 12 ppb to 3 - 4 ppb. Further research is being conducted in the area of iron reduction by evaluating the use of alternate amines. Minimizing the iron transported to the steam generators, tube surface fouling and deposit formations in the crevice regions are reduced. The deposit formations within the crevice due to iron transport can contribute to the degradation of the steam generator tubes at the tube support plate.

CHEMISTRY PROGRAM IMPROVEMENTS

The following additional administrative or procedural actions are in place or planned in support of this proposed license amendment request. The purpose of these actions is to mitigate the corrosive environment in the TSP crevices and to ensure that future growth rates and crack morphologies will be within expected bounds.

Byron has implemented the following actions:

- Full compliance with all EPRI Secondary Chemistry Guidelines, Revision 3.

- Reduction of SG Tubing Crevice Fouling

  - Use of advanced amines such as ethanolamine (ETA) for secondary pH control and iron transport reduction.

  - Secondary additives on startups to passivate and reduce the dissolved oxygen levels in the secondary system as well as maintaining a cleanup loop.

  - Perform a high temperature chemical cleaning process during B1R06.

- Improving SG Crevice pH

  - Performance of SG hideout return studies during shutdown to assess the impact of operating chemistry on SG crevice chemistry and potential formation of caustic crevices which can cause TSP ODSCC.

  - The molar ratio control program implemented during Cycle 6 will be evaluated. This program adjusts the sodium to chloride ratio in the steam generators by adding ammonium chloride to the condensate system.

  - The crevices will be chemically treated during B1R06. A high temperature chemical cleaning process is being designed with the target area of support plate crevices.

- SG Sodium Reduction

  - Modified the makeup water regeneration process to reduce overall impurity input to the secondary side, including sodium.

  - Evaluating the use of a reverse osmosis unit on the makeup water system to reduce impurities.

  - Performed 100% eddy current inspection of the 4 condenser water boxes during the last refueling outage.

- SG Electrochemical Potential (ECP) Reduction

  - Evaluating the use of secondary additives for maintaining reducing conditions in the SGs and passivation of piping systems and components. Currently, elevated hydrazine is used for this purpose.

  - Byron does not have any copper components in its feedwater or condensate systems.

- Increased sampling capabilities and steam generator corrosion monitoring.

  - Installation of a new steam generator sample panel. This will allow for constant chemistry and corrosion monitoring of individual steam generators.

  - Sodium analyzers will be installed on the steam generator blowdown demineralizers to monitor demineralizer performance.

Currently, Byron is pursuing several methods to further enhance the SG corrosion control program, in conjunction with the Corporate Chemistry Department. These methods are as follows:

- The addition of other amines either with ETA or in place of ETA will be evaluated to optimize the pH control and minimize iron transport.

- Chemical controls are being evaluated to improve iron transport out of the SGs. The goal is to increase the efficiency of iron removal via the blowdown system.

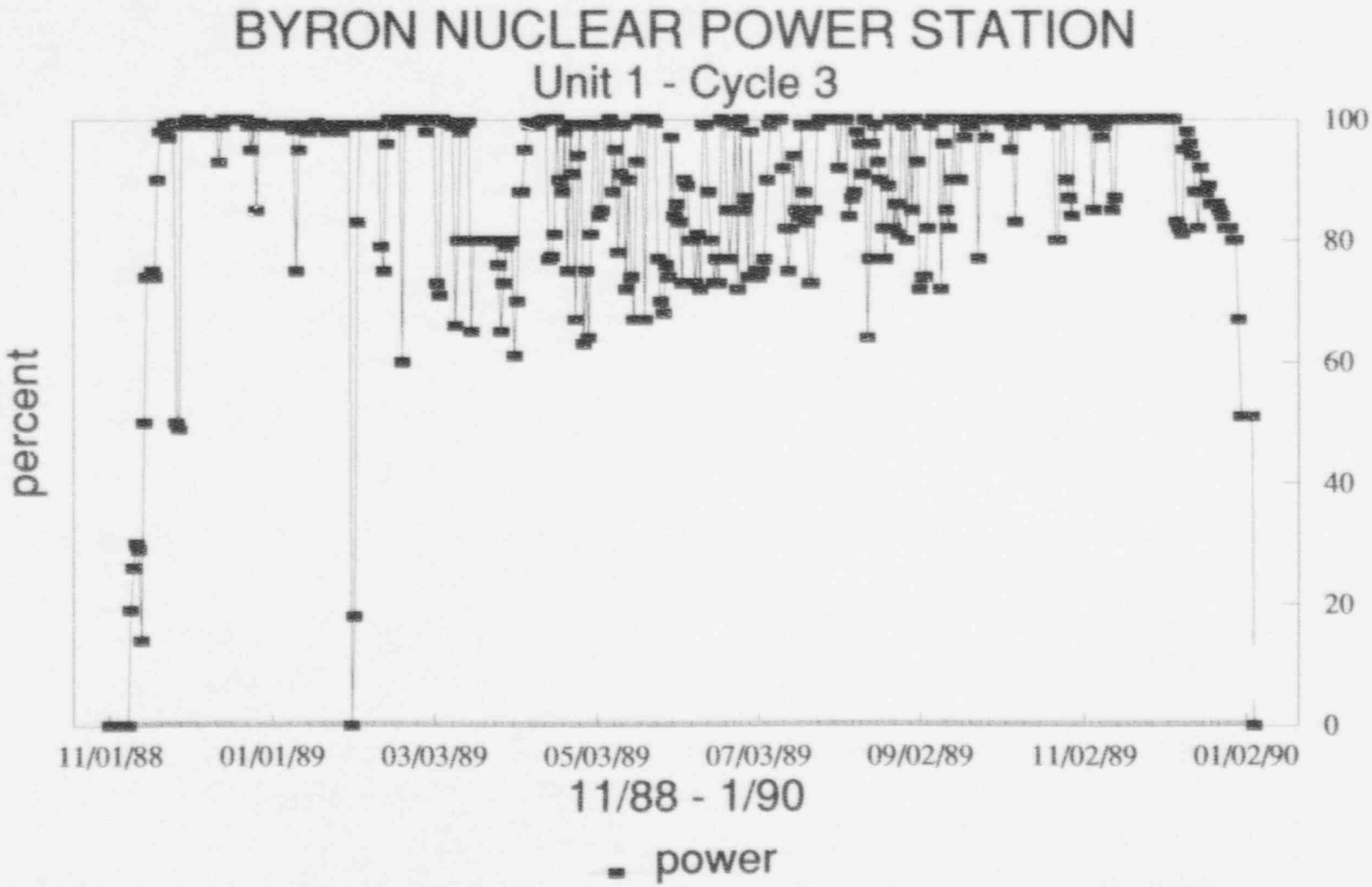

BYRON NUCLEAR POWER STATION
Unit 1 - Cycle 3
percent
100
80
60
40
20
0
11/01/88
01/01/89
03/03/89
05/03/89
07/03/89
09/02/89
11/02/89
01/02/90
11/88 - 1/90
power

Figure C - 1

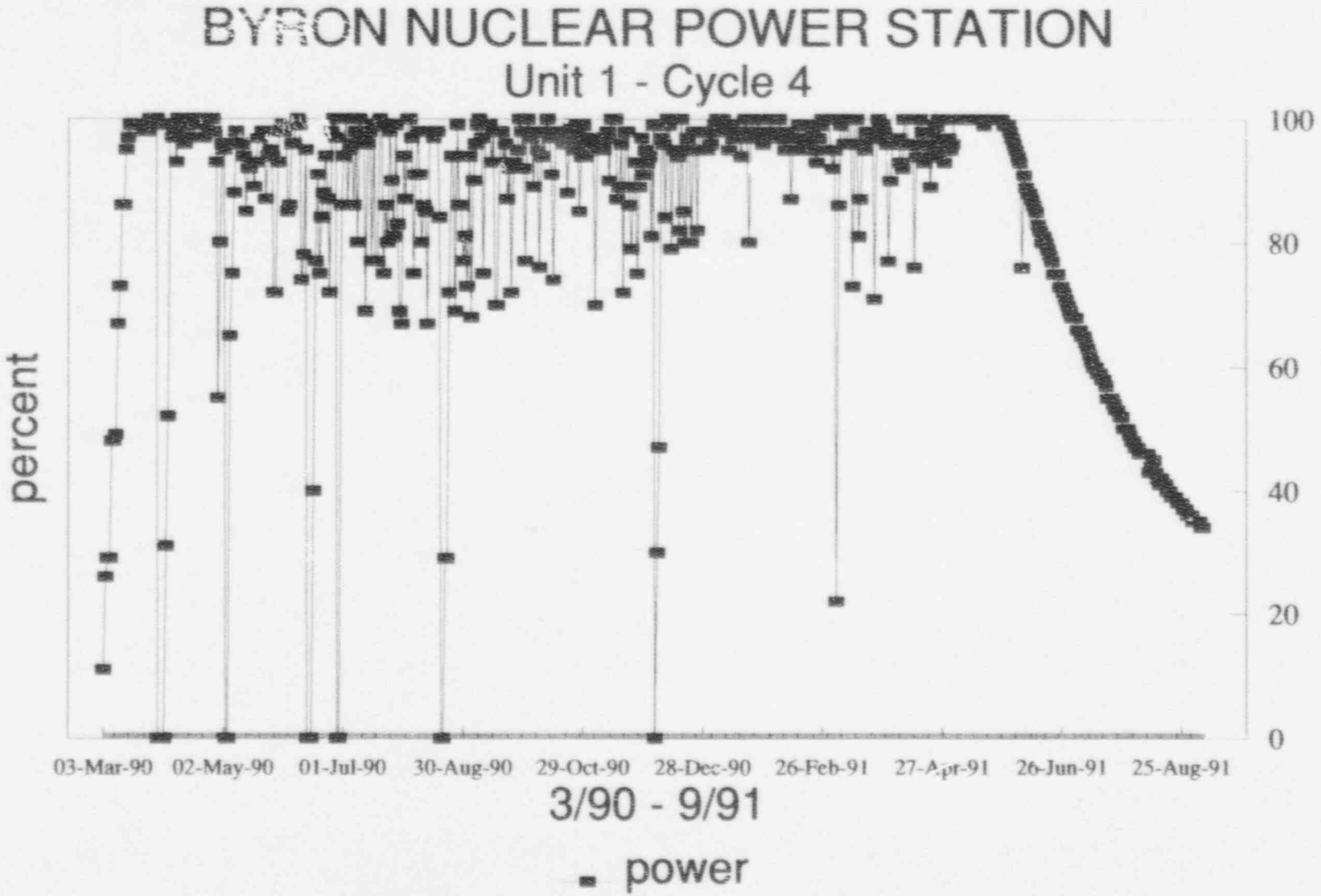

BYRON NUCLEAR POWER STATION
Unit 1 - Cycle 4
percent
100
80
60
40
20
0
03-Mar-90
02-May-90
01-Jul-90
30-Aug-90
29-Oct-90
28-Dec-90
26-Feb-91
27-Apr-91
26-Jun-91
25-Aug-91
3/90 - 9/91
power

Figure C - 2

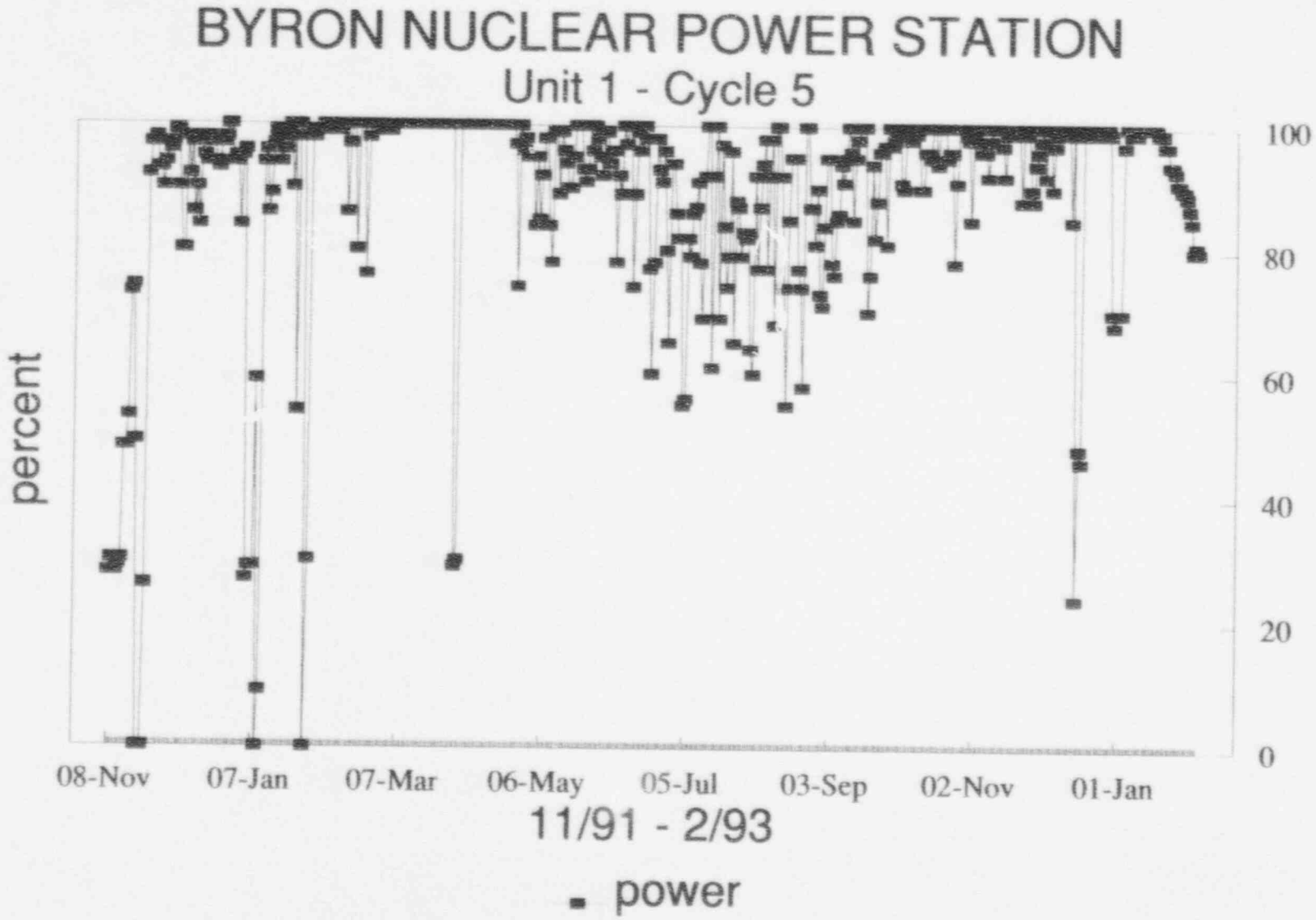

BYRON NUCLEAR POWER STATION
Unit 1 - Cycle 5
percent
100
80
60
40
20
0
08-Nov
07-Jan
07-Mar
06-May
05-Jul
03-Sep
02-Nov
01-Jan
11/91 - 2/93
power

Figure C - 3

Byron nuclear power station

Unit one cycle 6 power

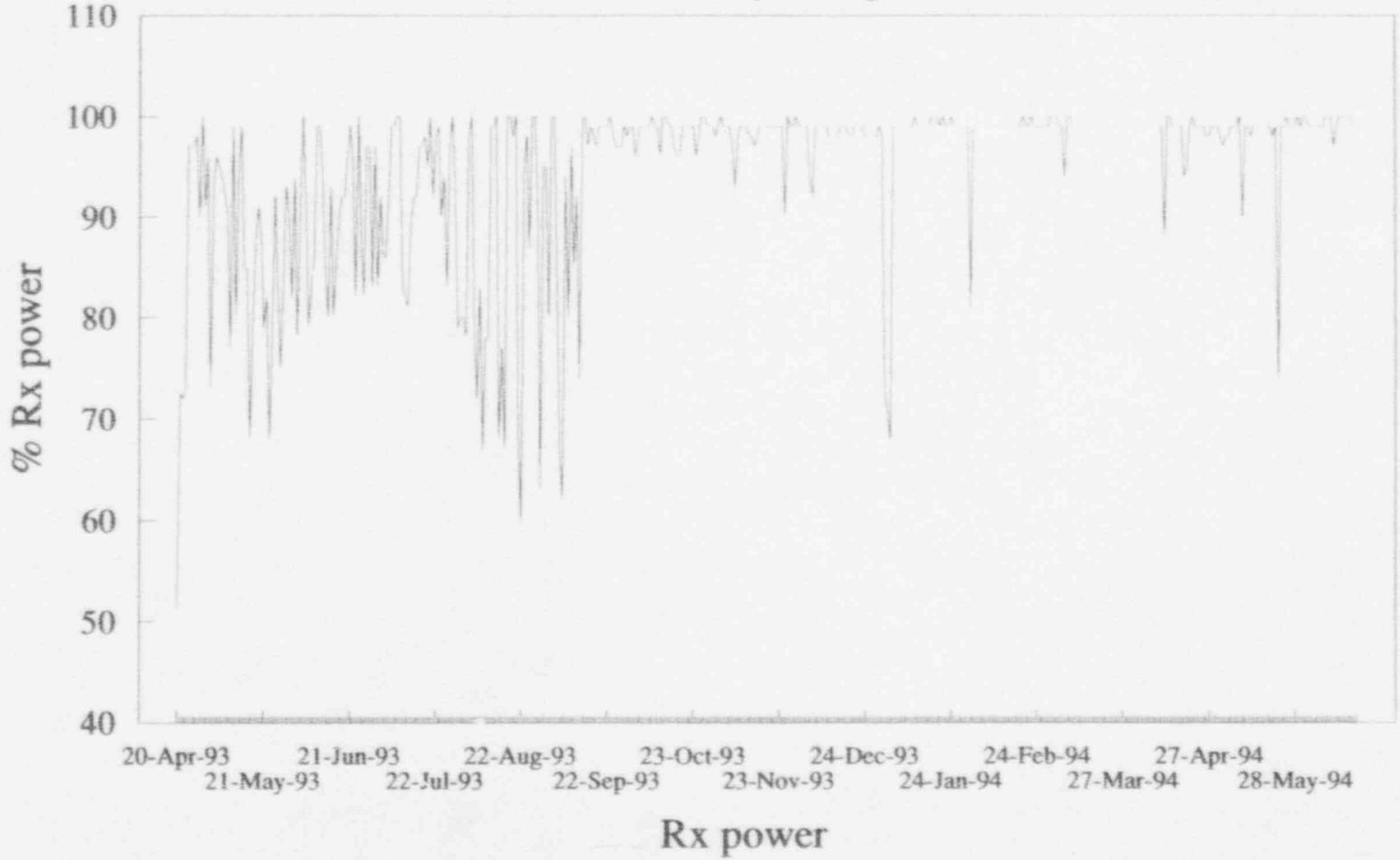


Figure C - 4

Byron nuclear power station

Unit one cycle 3 Na vs Cl

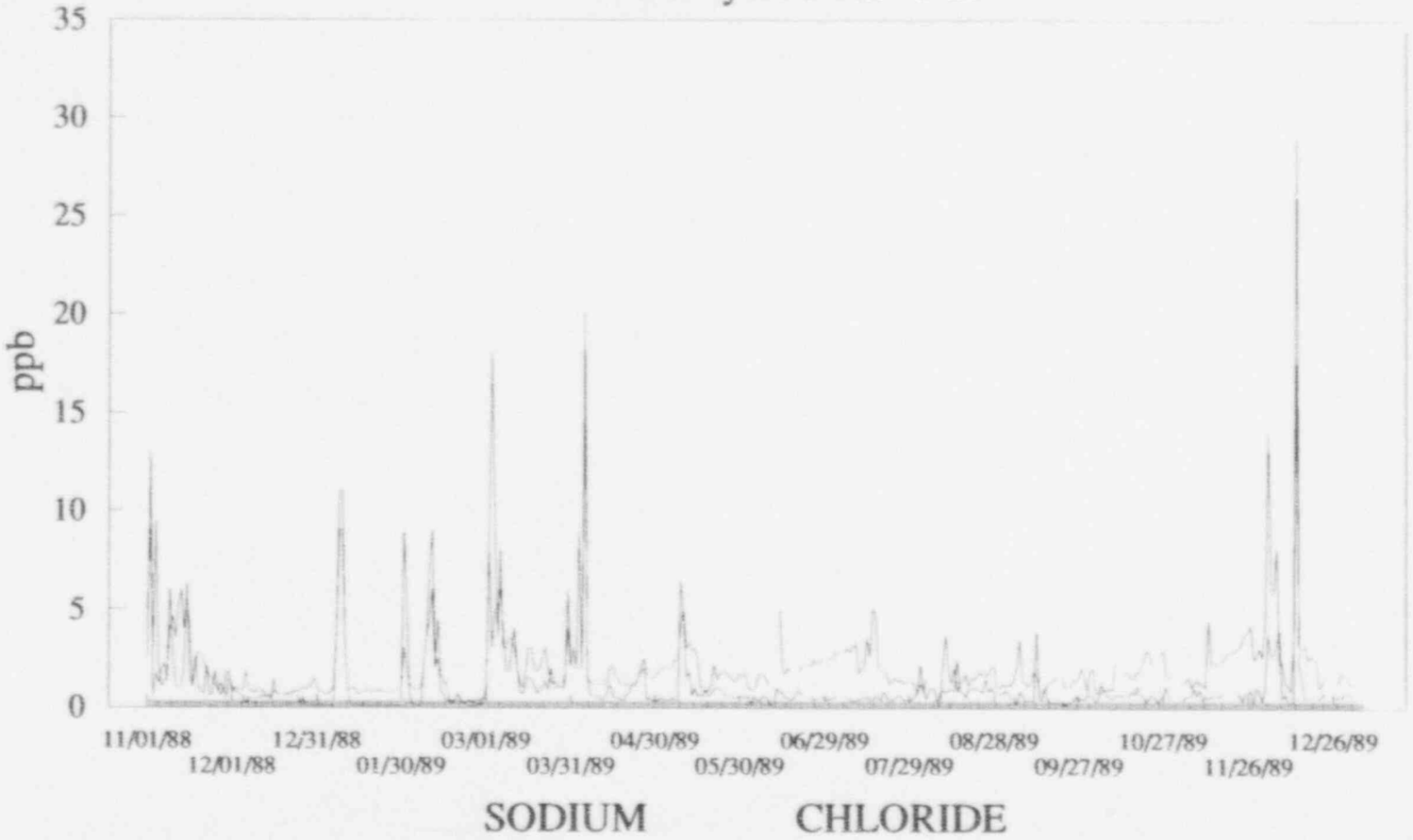


Figure C - 5

Byron nuclear power station

Unit one cycle 4 Na vs Cl

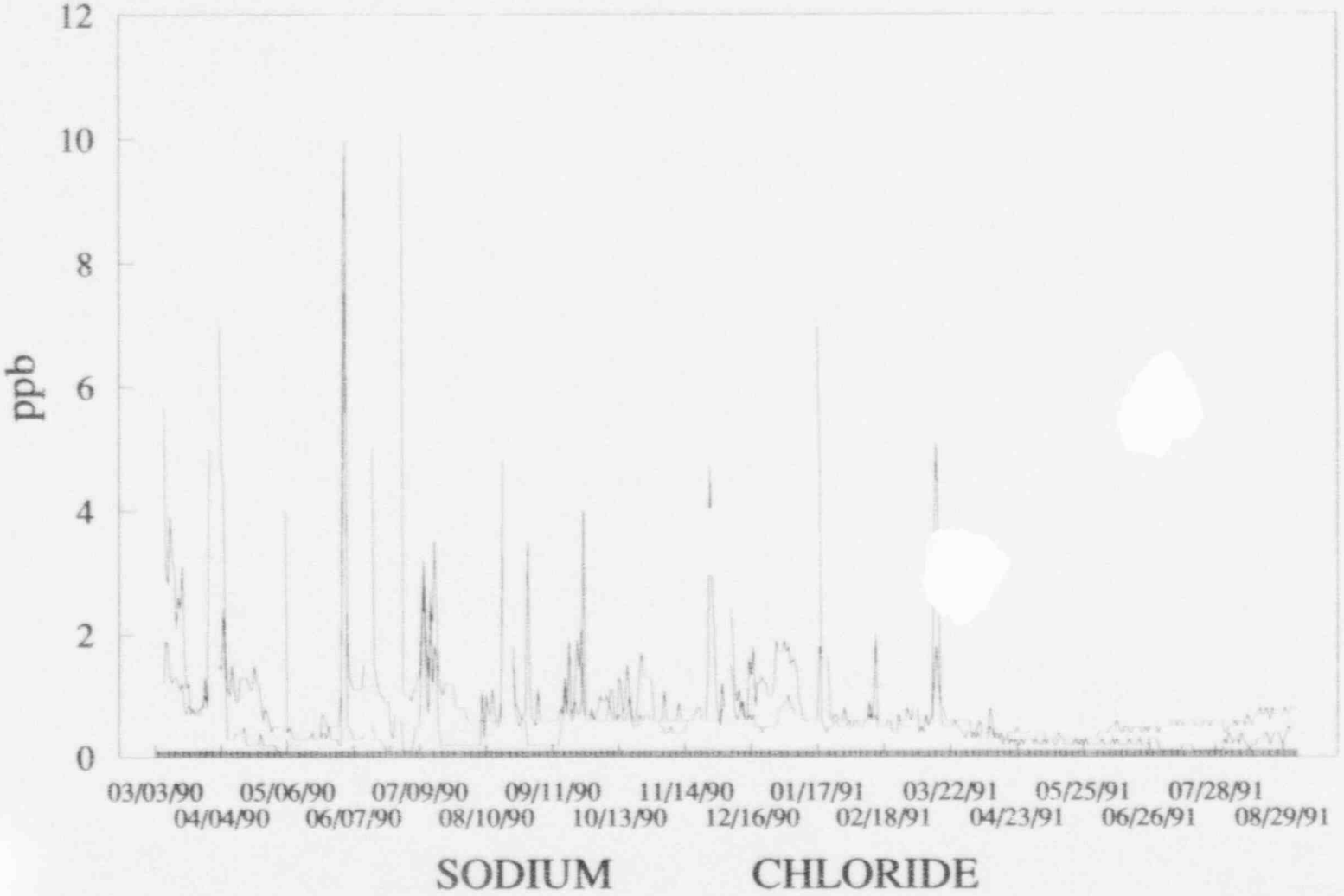


Figure C - 6

Byron nuclear power station

Unit one cycle 5 Na vs Cl

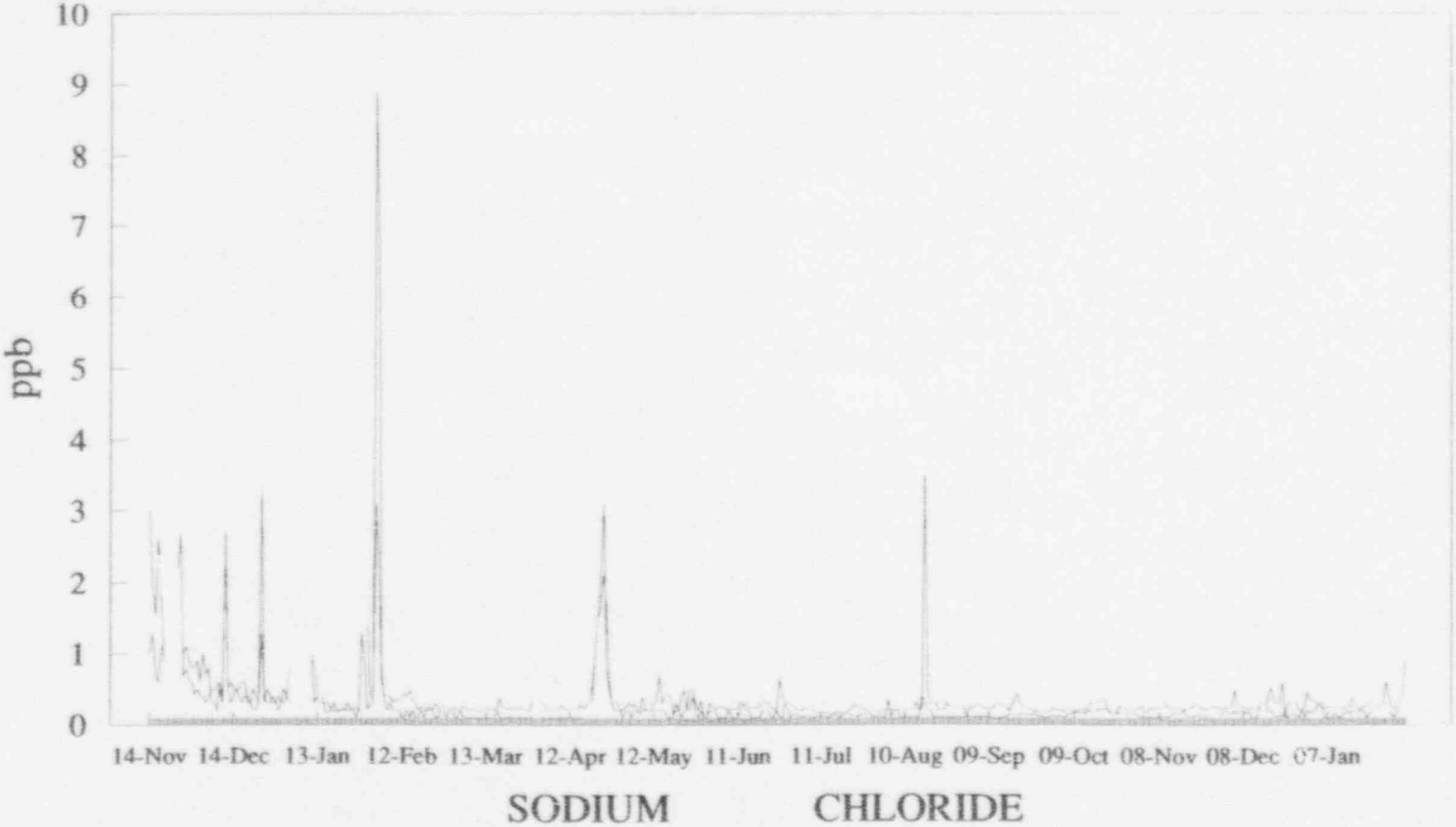


Figure C - 7

Byron nuclear power station

Unit one cycle 6 Na vs Cl

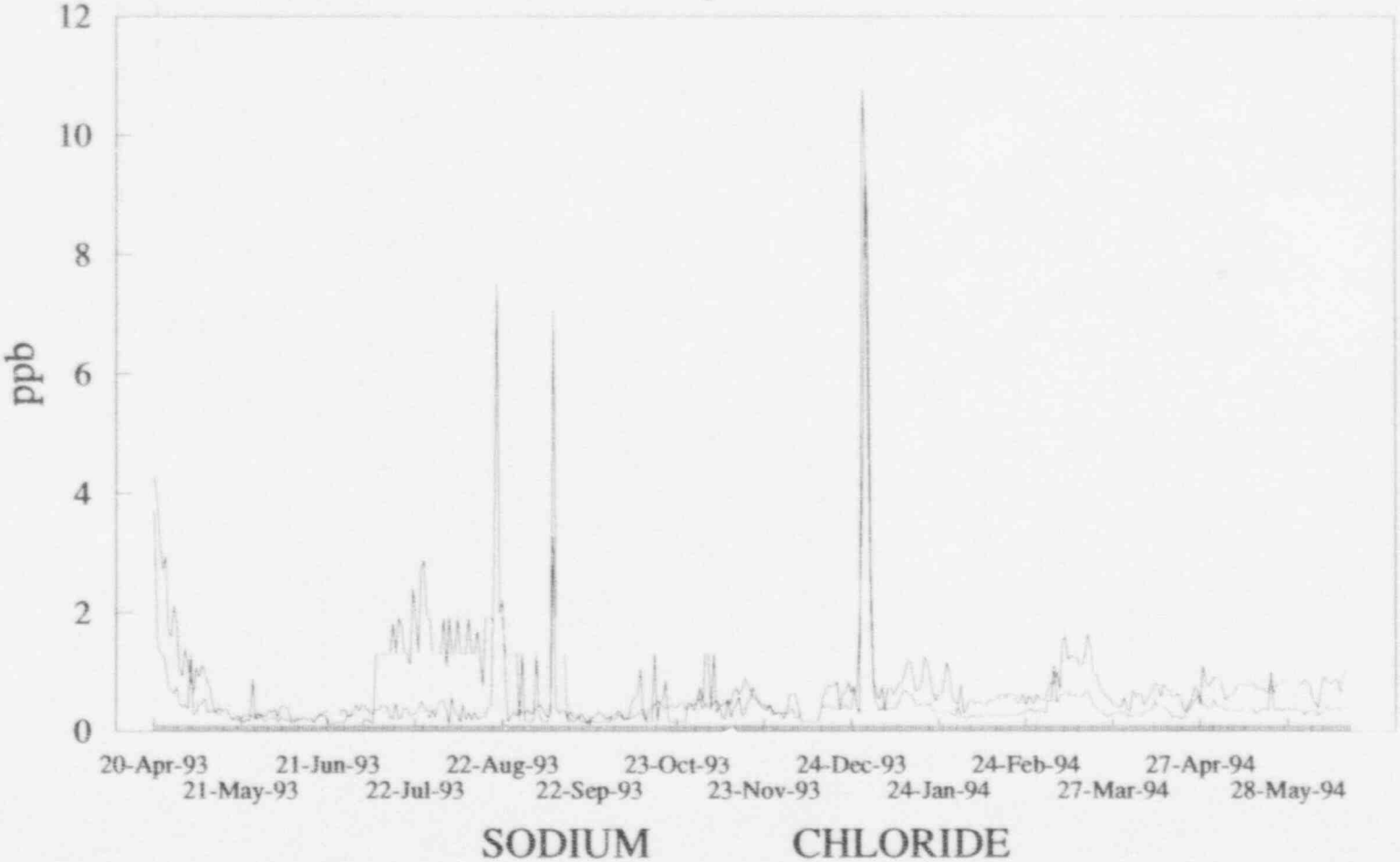


Figure C - 8

Byron nuclear power station

Unit one cycle 3 Na/Cl ratio

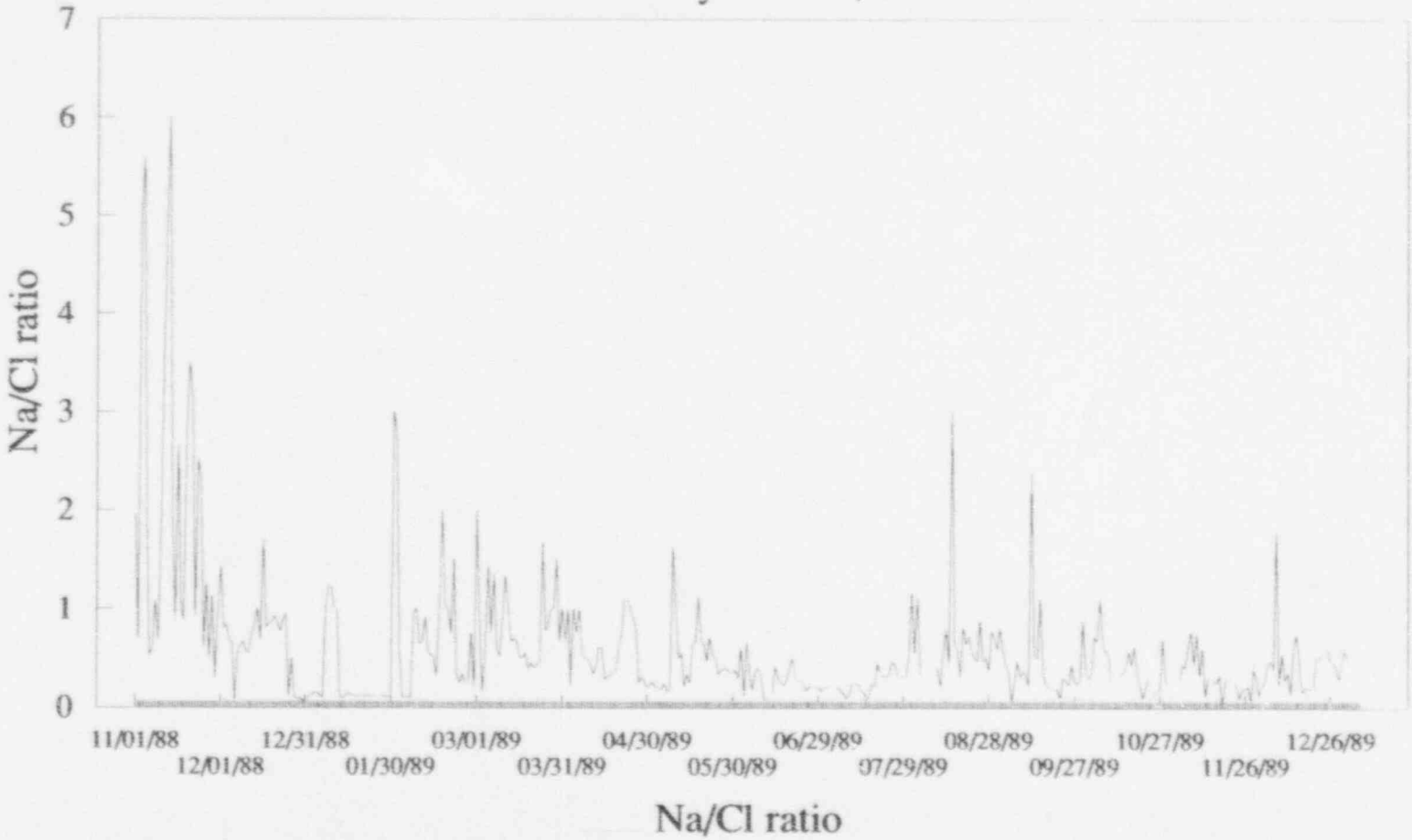


Figure C - 9

Byron nuclear power station

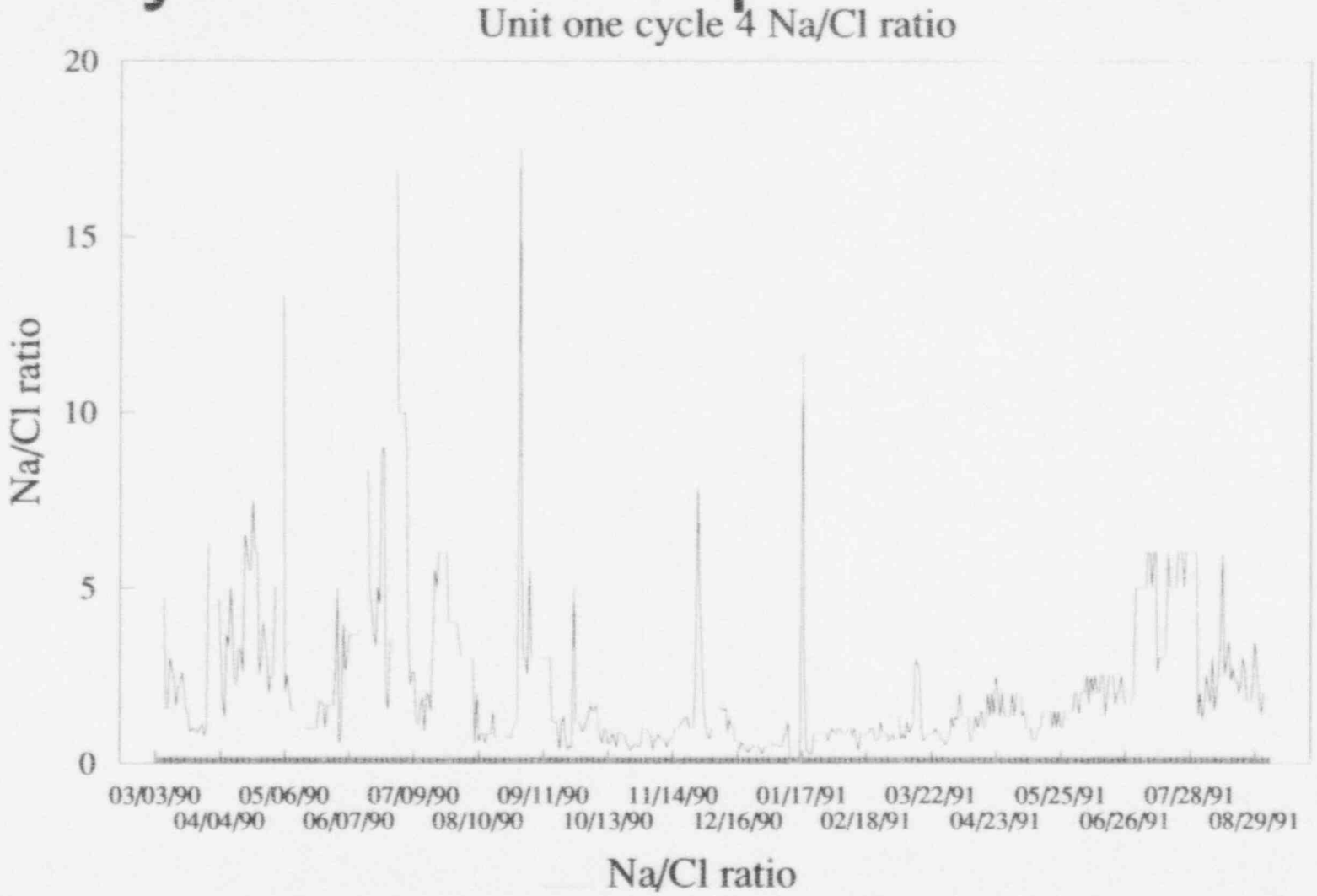
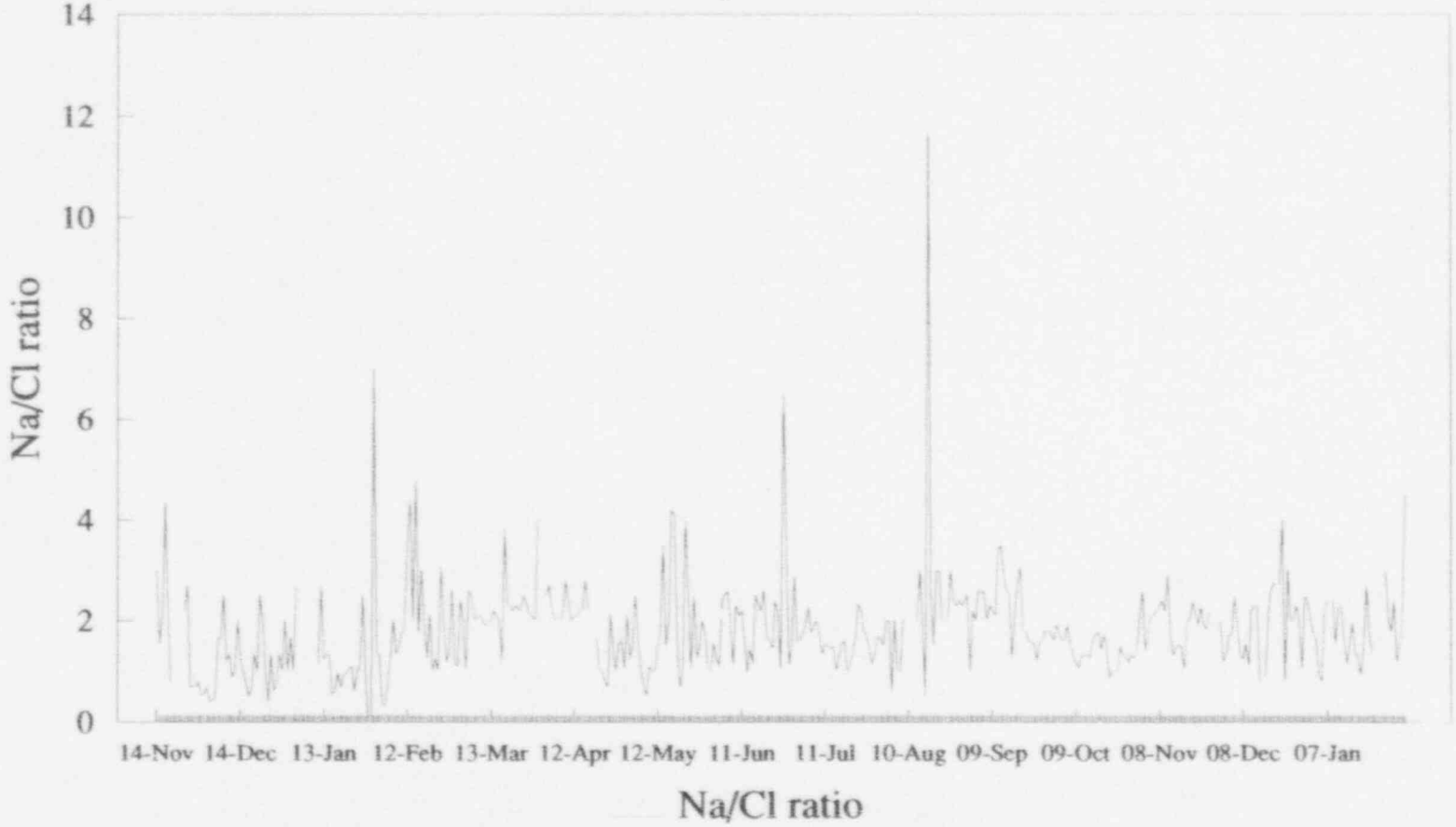


Figure C - 10

Byron nuclear power station

Unit one cycle 5 Na/Cl ratio



Byron nuclear power station

Unit one cycle 6 Na/Cl ratio

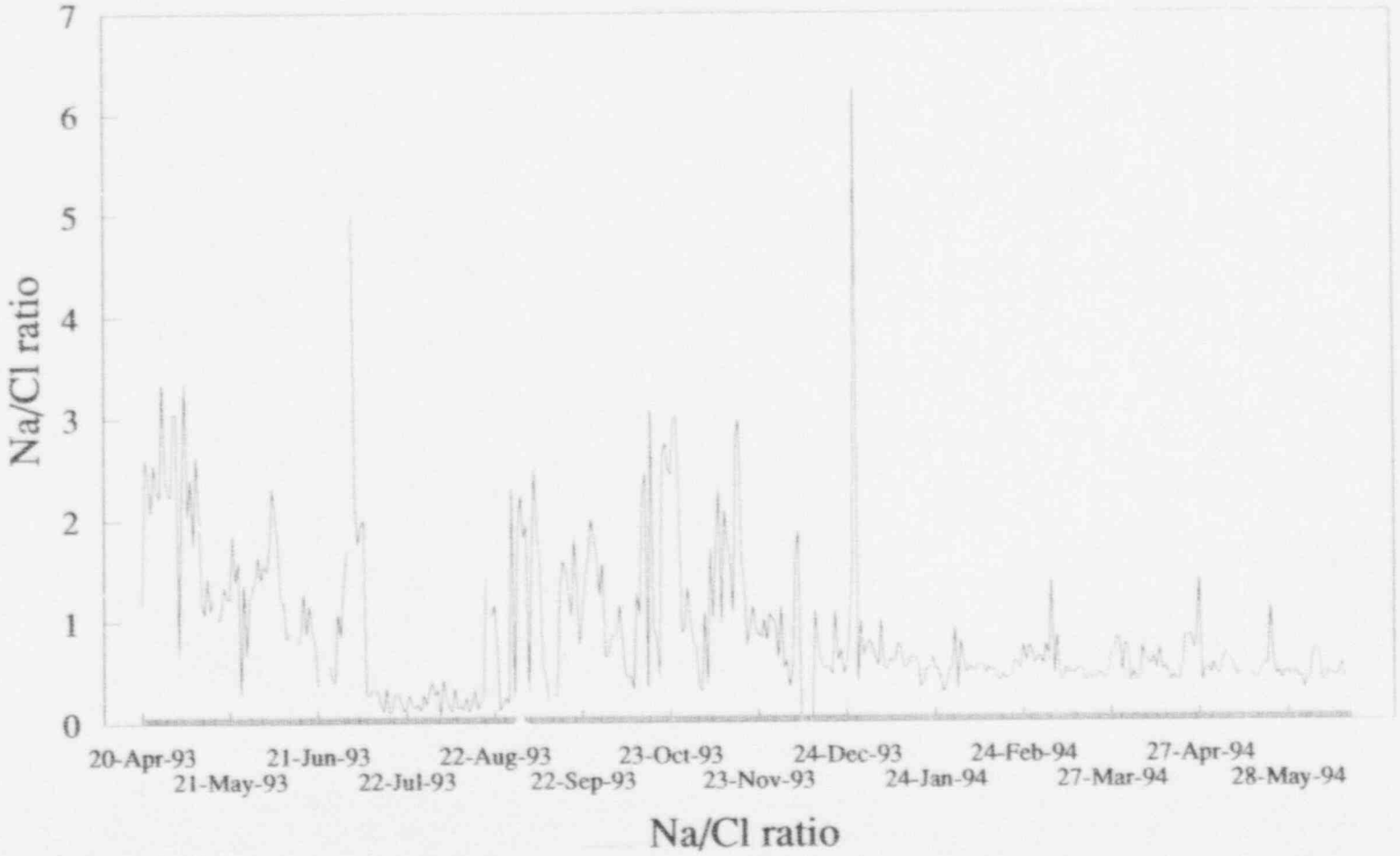


Figure C - 12

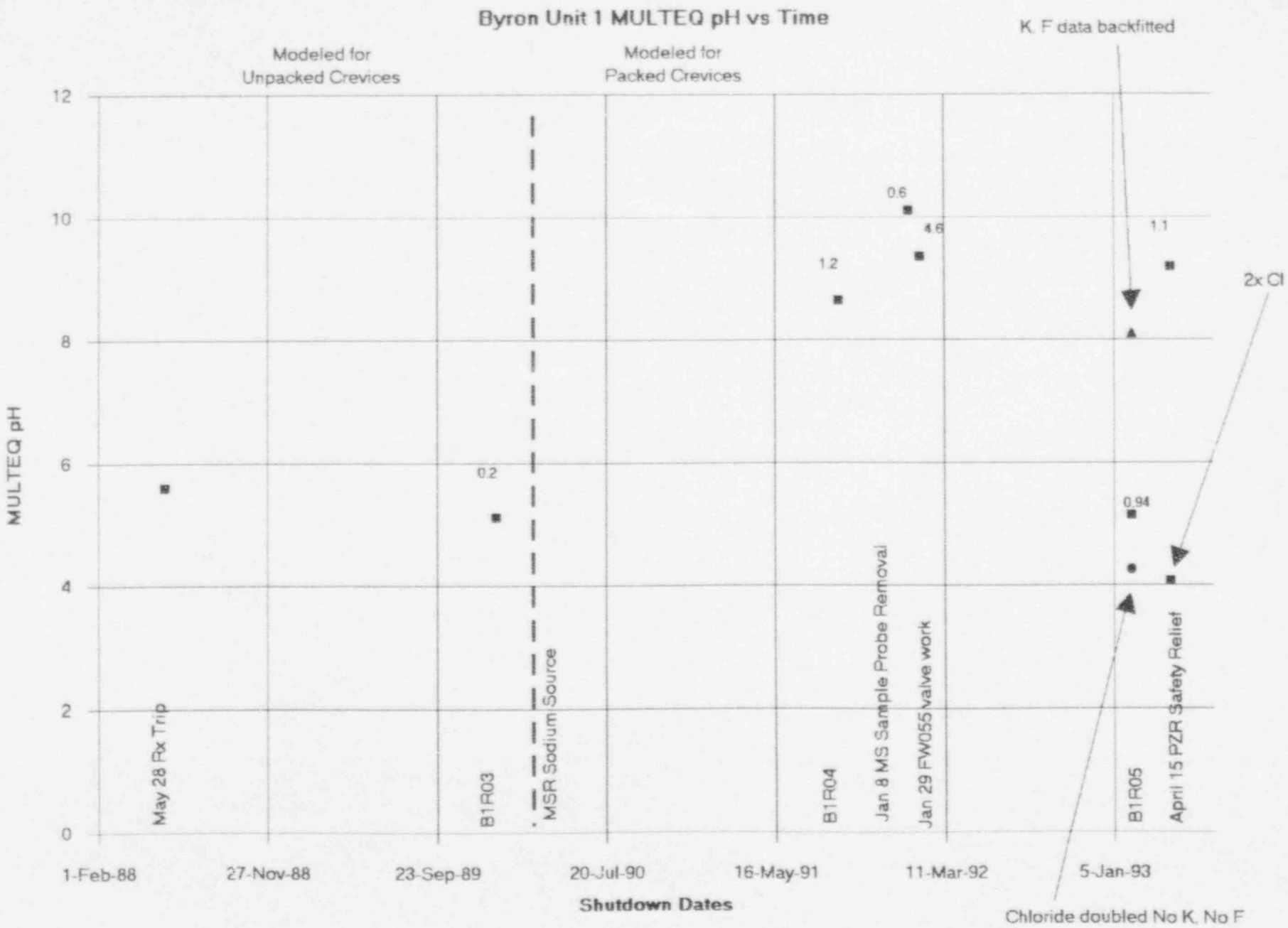

Byron Unit 1 MULTEQ pH vs Time
Modeled for Unpacked Crevices
Modeled for Packed Crevices
K. F data backfitted
2x Cl
MULTEQ pH
12
10
8
6
4
2
0
0.2
1.2
0.6
4.6
1.1
0.94
May 28 Rx Trip
B1R03
MSR Sodium Source
B1R04
Jan 8 MS Sample Probe Removal
Jan 29 FW055 valve work
B1R05
April 15 PZR Safety Relief
1-Feb-88
27-Nov-88
23-Sep-89
20-Jul-90
16-May-91
11-Mar-92
5-Jan-93
Shutdown Dates
Chloride doubled No K, No F

Figure C - 13

ATTACHMENT D

BYRON STATION
SYSTEM OPERATIONAL MEASURES/
DEFENSE IN DEPTH IN SUPPORT OF UNIT 1 IPC

Application of IPC at Byron Unit 1 is not expected to significantly increase the probability of primary-to-secondary SG tube leakage or the amount of leakage experienced, as discussed in Attachment F. However, it should be noted that system operational measures, such as equipment, procedures, and training, are in place to provide for monitoring and the ability to respond to SG tube leakage from any source. As a result of past industry leakage events, including SOER 93-01 and IEN 94-43, Byron has implemented and made plans to implement a number of program enhancements to improve the ability to detect leakage and respond to situations where leakage is detected. Data from the Braidwood primary-to-secondary leak in October 1993 was also specifically reviewed for applicability due to the similarity in design between Byron and Braidwood. The programs and equipment at Byron are revised, as appropriate, whenever significant industry input becomes available. The system operational measures available at Byron, including enhancements implemented and planned, are discussed below.

ABILITY TO DETECT

Byron Station uses a combination of different types of radiation monitors and a central processing unit (RM-11) in the Main Control Room to monitor for presence of radioactivity in the secondary system due to primary-to-secondary leakage. Isotopic analysis is used for verification, quantification and qualification of the leakage.

Permanently installed radiation monitors are on the main steam lines, steam generator blowdown lines, and at the Steam Jet Air Ejectors (SJAE). These monitors provide input to the RM-11 which provides trending capability in the Main Control Room. Byron also has two Merlin Gerin portable N-16 monitors capable of providing local measurement of steam line activity. Portable radiation detection instruments are also available for use in detecting or verifying primary-to-secondary leakage.

There are two redundant area radiation monitors on each of the four main steam lines. These monitors are externally mounted to the steam lines and are primarily used for the quantification of offsite releases in accordance with the Generating Station Emergency Plan (GSEP). They may also be used in the detection of primary-to-secondary leakage and identification of the leaking SG as directed by station procedures. In November 1993, NUMARC (NEI) issued new Emergency

Action Levels (EALs) for gaseous release rate classifications. The Main Steam Line monitor setpoints were revised due to the incorporation of the new EALs into the Byron GSEP procedures. The Alert Alarm setpoint was lowered to 0.55 mR/hr and the High Alarm setpoint was lowered to 5.5 mR/hr to ensure the radiation monitors would alarm prior to reaching the Site Emergency release rate EAL. In January 1994, the High Alarm setpoints were lowered again based on the recommendation of the ComEd Health Physics Support Department following a review of SOER 93-01 and data from the Braidwood October 1993 tube leak event. The revised setpoint is 1.1 mR/hr, which is currently in effect. The Alert Alarm setpoints were not revised further downward because the background levels range from 0.1 mR/hr to 0.3 mR/hr. Maintaining the Alert Alarm at 0.55 mR/hr prevents spurious alarms on the monitors.

The SJAE radiation monitor is a process monitor used to detect the presence of radioactivity in the off-gas system exhaust. The original monitor alarm setpoints were calculated using the realistic source terms listed in UFSAR Table 11.3-6, "Expected Annual Average Release of Airborne Radioactivity." In November 1993, the SJAE setpoints were lowered with the incorporation of the new NUMARC (NEI) EALs and a review of data from the Braidwood October 1993 tube leak event. The Alert Alarm setpoints at both Byron and Braidwood were reduced to three times background in accordance with industry practice. Background at Byron is 7.5E-6 μci/ml. At one point during the Braidwood tube leak, the SJAE monitor read 3.0E-5 μCi/ml for a calculated 300 gpd leak rate. Based on this reading, the High Alarm setpoint was lowered to 1.0E-5 μCi/ml at both Byron and Braidwood. Thus, a primary-to-secondary leak of 150 gpd or less would actuate the SJAE High Alarm assuming the RCS activity level is comparable to Braidwood's at the time of the leak. Currently, the RCS activity level at Byron is approximately one sixth the activity level of Braidwood's RCS in October 1993. Setpoints continue to be reviewed as leak rate data is obtained.

This monitor specifically detects the presence of noble gas in the off-gas system. Noble gases in the off-gas system are an indicator of primary-to-secondary leakage. This monitor cannot differentiate which generator contains the leak. Historically, the concentration of noble gas found in the SJAE has not correlatd to primary side noble gas concentrations. Potential contributors to this difference are the presence of additional lines feeding into the monitor, other than off-gas, and moisture intrusion in the process lines. The calculated leak rate using the SJAE monitor has also been compared to the SG blowdown monitor and this data has not correlated well. This difference may be caused by the liquid blowdown samples being subject to error due to iodine hideout and dilution effects.

In response to IEN 94-43, a July EPRI workshop, and the July Sorrento Users Group meeting, the feasibility of SJAE monitor upgrades are being pursued by a Byron Station task force. This group is in contact with other utilities for

information on methods in use for leak rate calculations, such as the use of Xe-133 and Xe-133 equivalents. An action plan will be developed to coordinate the efforts of the technical groups to enhance the monitoring and quantifying of primary-to-secondary leaks.

The SG Blowdown Radiation Monitor is used to detect the presence of radioactivity in the secondary side of the SGs by sampling the SG blowdown lines. The monitor concurrently samples two SG blowdown streams, which are manually selected. With this alignment, the monitor cannot differentiate which of the selected SGs has the primary-to-secondary leak, but a grab sample from the SG sample panel can be obtained to quickly identify the specific generator. The Steam Generator Blowdown monitor setpoints were originally calculated using the realistic source term which is described in UFSAR Table 11.1-6, "Realistic Source Terms for Steam Generator Blowdown and Radioactive Source Streams." Three assumptions were used in this calculation:

(1) Presence of a one gallon per minute primary-to-secondary leak in one generator,
(2) Blowdown from the affected generator is 90 gpm and blowdown from the other three generators is 15 gpm per generator, for a total of 135 gpm, and
(3) A dilution factor of 1/135 is used.

The calculated setpoints are 1.27E-2 μCi/ml for the Alert Alarm and 1.54E-2 μCi/ml for the High Alarm. The Blowdown monitor setpoints were not affected by the new EALs or data from the Braidwood tube leak. A modification to install a new SG sample panel will provide additional capability to continuously monitor all four steam generators. A grab sample will still be required to identify the affected SG and quantify the leak. Sample collection time will be reduced since all four SG sample lines will be continuously purging. Following completion of the modification, the radiation monitor alarm setpoints will be appropriately revised to account for the increase in sample flow through the monitor. The installation of this modification is currently scheduled for completion in October 1994.

Point history data from the main steamline, SJAE, and SG blowdown radiation monitors were reviewed for three separate Byron Steam Generator Tube leaks. Three different leak rates were examined: 1 gpd, 130 gpd and 310 gpd. An attempt was made to correlate chemistry isotopic data to radiation monitor readings for given leak rate values. However, because the majority of point history data from these events was not recorded to tape, the correlation was not achievable. Due to this lack of Byron specific data, Braidwood October 1993 tube leak data has been used in determining main steamline and SJAE monitor setpoints as discussed above.

The two portable N-16 monitors are currently measuring the 1B and 1C Main Steam Lines. Past SG inspections at Byron have shown that these SGs have the most indications and are most likely to develop tube leakage. However, the monitors can be relocated to any loop's Main Steam Line and made operational within 15 minutes. The detectors have associated local strip chart recorders for obtaining trend information. The local trends are reviewed shiftly by the Operating Department. Currently, Byron is in the process of acquiring four more portable N-16 monitors. It is expected that the new monitors will be operational in mid-1995. Based on previous primary-to-secondary leak experience at Byron, the N-16 monitors can provide reliable indication at leak rates above 10 gpd, following calibration to chemistry calculations during an event, depending on the RCS activity level.

Primary-to-secondary leakage can also be detected through chemistry sampling of the secondary system gaseous and/or liquid effluents. Samples can be taken using installed sampling equipment from the SG blowdown system and the off-gas system. Routine samples are taken from the SG blowdown system. Quantification of leakage requires 1.5 to 2 hours. To determine the presence of radioactivity in a SG requires approximately one hour for a full count or 10 minutes for a quick count. The most accurate determination of primary-to-secondary leak rate requires isolation of blowdown and approximately 8 hours of steady power operation.

Depending on the level of RCS activity and the size of the primary-to-secondary leak, different iodine isotopes are measured to determine the leak rate. Most typically, I-131, I-133, and I-135 are detected in the SG secondary side and compared against the ratio of these isotopes found in the RCS. Based on past experience at Byron, it has been determined that detection of primary-to-secondary leakage is possible with leaks of approximately 20 gpd at current RCS activity levels. Accurate quantification of primary-to-secondary leakage is possible with leaks of approximately 50 gpd.

MONITORING PROGRAM

Routine SG leakage monitoring is accomplished by SG blowdown chemistry sampling every 72 hours and continuous radiation monitor alarm capability in the Main Control Room. The RM-11 panel, which provides alarm capability, also provides the operators the capability of trending monitor readings on a 10 minute, hourly, or 24 hour average basis. Chemistry sampling is being performed daily while the SG blowdown sample panel is being upgraded. When a leak is detected by a radiation monitor alarm or by evaluation of a secondary system chemistry sample, procedures are available to provide general actions to be taken to confirm leakage, to increase frequency of trending of the leakage via radiation monitors and sampling, to minimize the impact of secondary system contamination on plant

operation, and to provide guidelines for possible plant shutdown due to absolute value or rate of change of the leakage. Specific details of the available operating procedures are discussed below in the section **OPERATOR ACTIONS IN RESPONSE TO KNOWN LEAKAGE**.

When leakage is suspected, steam generator activity sampling frequency is increased, at the discretion of the Shift Engineer or Duty Chemist, to at least shiftly for increasing activity levels or daily for stable activity levels. An hourly chemistry sample may be requested to establish a trend or as required by BOA SEC-8. Permanently installed secondary system radiation monitors are trended on an hourly basis from the RM-11 in the Main Control Room. One of the portable N-16 monitors would be relocated to the suspect steam generator's main steam line for shiftly monitoring, as well. While other actions are being carried out, monitoring of the leak rate is continued.

Byron experience has been that small leaks are detected initially by chemistry sample analysis before the leak size becomes large enough for detection on the radiation monitors. This is due, in part, to the low level of activity in the Reactor Coolant System (RCS). However, the alarm setpoints on the secondary system radiation monitors will alert the Control Room Operators to larger leaks, which can then be confirmed by chemistry sampling.

The installation of permanent main steam line N-16 monitors with direct readout and alarm capability in the Main Control Room was evaluated by Byron Station. Permanent N-16 monitors provide nearly instantaneous indication of primary-to-secondary leakage presence and leakage trends to the Control Room operators whenever the reactor is operating. An indication on the N-16 monitors must be calibrated to a chemistry grab sample in order to provide accurate leak rate information. Due to transport times and the short half-life of the N-16 isotope, an increased leak rate may not be readily detectable on the N-16 monitor depending on location of the leak. Periodic grab sample comparisons must be performed to ensure the initial correlation between monitor response and leak rate remain valid. The comprehensive Byron Station leakage monitoring program previously discussed, which includes the use of currently installed secondary system radiation monitors, chemistry sampling, and the use of portable N-16 monitors, provides comparable detection and trending capabilities to the Control Room operators allowing them to respond quickly to indications of SG tube leakage. Chemistry sample analysis is required to determine the actual leak rate prior to taking action to perform a plant shutdown, whether the initial leakage indication comes from permanent or portable radiation monitors. The existing monitoring program provides similar functions to those available by adding permanent N-16 monitors without incurring the expense of a modification. The cost to install permanent

N-16 monitors is approximately $400K per unit. Without a significant increase in detection or trending capabilities, Byron Station has determined that permanent N-16 monitors are not justified.

OPERATOR ACTIONS IN RESPONSE TO KNOWN LEAKAGE

Byron has created Byron Operating Procedure BOP MS-11, "Operation With Steam Generator Tube Leakage", in addition to Byron Abnormal Operating Procedure BOA SEC-8, "Steam Generator Tube Leak", to outline the activities to be addressed when continued plant operation is allowed following the identification and confirmation of a steam generator tube leak. The following actions are addressed by BOP MS-11:

- The Chemistry Department establishes an increased sampling frequency based on steam generator activity trends.

- Potentially contaminated secondary systems are identified as well as actions to minimize inputs into the affected unit's secondary systems.

- Radwaste processing systems are prepared for possible extended use.

- The Radiation Protection Department is notified to perform secondary system radiological surveys.

- Operating Department to conduct walkdowns of secondary systems to identify any leakage.

- Any secondary leaks identified during the walkdowns are reported to the Radiation Protection Department to address possible contamination concerns and a Health Physicist for verification that offsite releases are below 10CFR20 limits.

- System Engineering is contacted to install a portable N-16 monitor on the leaking steam generator's main steam line. The Operating Department initiates shiftly monitoring of the portable N-16 monitor.

Abnormal radiation in a steam generator indicates primary-to-secondary leakage. This can be shown by trends or alarms on main steamline, steam jet air ejector, or SG blowdown radiation monitors, or from chemistry samples. A large leak could also be indicated by feedwater flow being less than steam flow, decreasing feed flow, or decreasing feed regulating valve position in conjunction with a stable steam generator level, although these particular symptoms would more likely be noticed with a tube rupture. Any of these indications require entry into

BOA SEC-8. This procedure provides actions to mitigate a range of steam generator tube leaks from the smallest detectable up to those that can be controlled with a normal chemical and volume control system lineup (approximately 120 gallons per minute). The basic flow of this procedure is as follows:

- Enter the procedure based on any indication of primary-to-secondary leakage. This includes an increase in pre-existing leakage (i.e. increasing trends on main steamline or steam jet air ejector rad monitors, increased activity in chemistry samples, or an imbalance between RCS charging and letdown flows).

- Take actions to maintain RCS inventory. Trip the reactor and actuate safety injection if inventory can not be maintained.

- Minimize secondary system contamination by following the actions of BOP MS-11.

- Perform a gross leak rate check to determine if immediate plant shutdown should be commenced.

- Perform hourly trending of SG leak rate data using radiation monitor indications. The Chemistry Department is also contacted at this time to sample once per hour and perform leak rate determinations.

- Determine if shutdown is required based on either absolute leak rate or on a change in leak rate. Exit the procedure if shutdown is not required, but continue to monitor leak rate.

- Initiate a unit shutdown, within the time limit specified.

- Identify the leaking SG based on feed flow or SG level indications, high activity in any SG sample, or an increasing trend on any main steam line radiation monitor. (After this point the actions are very similar to Emergency Procedure BEP-3, "Steam Generator Tube Rupture", actions for a responding to a SG tube rupture.)

- Isolate flows from and to the leaking SG.

- Cooldown and depressurize the RCS to stop leakage.

- Cooldown the leaking SG using backfill, blowdown, or steam dump as determined by the Station Duty Officer.

BOA SEC-8 provides two basic methods of leak rate determination to the operators. Leak rates for large leaks (10 gpm or greater) can be determined by the operator from a charging-letdown flow balance on the RCS. Leak rates for smaller leaks use activity levels in the secondary system and are determined by the Chemistry Department. Main steam line and steam jet air ejector radiation monitor status is trended in the main control room on the RM-11. Stable radiation monitor indications are indicative of a stable leak rate. For stable radiation monitor readings, shutdown requirements are based on absolute leak rate values . Should radiation monitor readings show a rapid increase during a shutdown based on absolute leak rate values, the allowed time to complete the shutdown is reduced to the shortest time practical for a controlled shutdown (4 hours).

If secondary radiation monitor readings are increasing and the absolute leak rate is increasing, then the change in leak rate shutdown requirements are applied. The change in leak rate shutdown requirements are only applied if absolute leak rate exceeds 50 gpd.

Response to each of the three general categories of leaks is discussed below and summarized in Table D-1.

Leakage of less than 150 gpd that is changing less than 25 gpd per hour is the first category of leakage. For this leak size, the operator initiates actions to minimize contamination, monitor releases and leak size, and then exits the procedure. BOA SEC-8 is reentered if the leak size increases.

The second category of leakage consists of leaks that are less than 10 gpm and either 1) greater than or equal to 150 gpd, or 2) changing by greater than 25 gpd per hour but less than 100 gpd per hour when greater than 50 gpd. For this leak size, the operator again initiates actions to minimize contamination, monitor releases, and monitor leak size. The operator determines the time requirement for shutdown and leaves BOA SEC-8 to perform a shutdown and cooldown using the Byron General Procedures (BGPs). When the unit has reached MODE 5, the SG is isolated for repairs. During the shutdown, BOA SEC-8 is reentered if the leak size increases as indicated by a rapid increase in radiation monitor response or subsequent SG samples. Reentry into the BOA allows the operator to apply shorter shutdown time requirements.

The third category of leakage consists of leaks that are either larger than 10 gpm, or less than 10 gpm but increasing by greater than 100 gpd per hour. In this case, the operator performs a shutdown of the reactor within 4 hours following the BGPs, but remains in BOA SEC-8 to perform SG tube leak recovery in a manner similar to the Emergency Procedures.

A leak size of greater than 120 gpm requires a Reactor Trip and Safety Injection. This size leak is addressed as a tube rupture by the Byron Emergency Procedures (BEPs).

Procedural Adequacy

The Byron procedures have been reviewed in light of the Institute for Nuclear Power Operations (INPO) Significant Operating Experience Report (SOER) 93-01, "Diagnosis and Mitigation of Reactor Coolant System Leakage Including Steam Generator Tube Ruptures." The diagnostic process contained in Byron procedures is consistent with the requirement of SOER 93-01 and includes the use of radiation monitors and alternate indications. The diagnosis is treated as a continuous action step. The chemistry procedures for confirming and calculating leakage were also reviewed and found to be consistent with industry practices. Currently, EPRI is establishing Primary-to-Secondary Leak Guidelines to provide guidance to utilities for developing consistent, industry-wide methods for leak rate determinations. Upon issuance of the EPRI guidelines, Byron procedures will be reviewed and updated, as appropriate.

OPERATOR TRAINING

Simulator Training

The Byron Training Department routinely develops simulator scenarios for initial and continuing licensed operator and shift technical advisor training that includes steam generator tube leak and rupture scenarios with post tube leak SG cooldown actions. These scenarios present variations in radiation monitor response, including equipment failures that remove expected primary indications and conditions that complicate the mitigation of the accident. The training program ensures that steam generator tube rupture diagnosis avoids over reliance on radiation monitors by requiring the use of alternate indications. Training recommendations identified in SOER 93-01 have been incorporated into the Byron steam generator training program.

Since 1993, 11 scenarios have been used for training operators to diagnose and respond to steam generator tube leaks and ruptures. Leakage used in these scenarios range from a 50 gpm leak to a beyond design basis tube rupture of 2000 gpm. Sources for scenario development include the plant design bases as well as recent industry events (i.e., Palo Verde tube rupture event). One scenario supplements the control room crew with a Radwaste Foreman during the conduct of a post-SGTR cooldown of the plant to provide guidance in control, processing,

and disposal of the contaminated water generated from the accident. Feedback from crew responses to this scenario has been incorporated into station procedures including BOP MS-11.

Simulator Fidelity

Byron's Training Department has verified that the simulator capabilities and training scenarios accurately reflect the expected indications and plant responses during a steam generator tube rupture. This includes reviewing industry tube rupture events such as the North Anna, McGuire, and Palo Verde accidents to ensure the accuracy of the simulator model. In addition, following the 1993 Braidwood tube leak, actual plant data was evaluated against the simulator response.

Also, the Radiation Protection Department has done extensive verification of the simulator radiation monitor response to ensure that they are realistic and reflect actual plant response. This allowed the Station's Emergency Plan Exercise to be conducted on the simulator.

Classroom Training

The Licensed Operator and Continuing Training programs also include classroom training on various steam generator tube rupture and tube leak events. The training also reviews steam generator leak mitigation procedures.

Each year, the operators receive training on steam generator tube ruptures. The specific areas covered change from year to year, in part to incorporate industry events and experience. In 1993, the Training Department presented a detailed case study of the Palo Verde event that addressed all of the training recommendations identified in SOER 93-01. In 1994, the Training Department is covering the following related subjects as a part of the License Requalification Training program:

a. Steam Generators:
    1. Review of steam generator design and construction.
    2. Primary water stress corrosion cracking.
    3. Review of related Technical Specifications including the recently approved amendment concerning tube sleeving.
    4. Testing of steam generator tubes.
    5. Braidwood steam generator degradation.
    6. Zion's recent tube leak event.

b. Steam Generator Chemical Cleaning:
   1. Review of upcoming evolution during B1R06.
   2. Overview of operator's response in the event that a tube rupture occurs during the chemical cleaning process.

c. BOP MS-11 "Operation With Steam Generator Tube Leakage":
   1. Actions to minimize secondary contamination when a steam generator tube leak is detected.
   2. Methods for processing contaminated secondary water.

STEAM GENERATOR EDDY CURRENT TESTING

The objective of performing eddy current inspection of the steam generator tubing is, in part, to verify operability of the SGs in accordance with the Technical Specifications. This is done by identifying and repairing tubes that contain indications exceeding the repair limit. The inspection also provides information on the modes of tube degradation to assist in determining appropriate actions for mitigation. To assure consistent and reliable inspections, eddy current inspection guidelines and data analyst training programs have been implemented.

Guidelines

The guidelines for the eddy current test and data analysis procedures have been revised in support of IPC to ensure the consistent, reliable detection of low voltage signals due to ODSCC while minimizing voltage response and voltage measurement variability. The data analysis guidelines, common to both Byron and Braidwood, are consistent with the EPRI Steam Generator ODSCC Alternate Repair Criteria guidelines and the guidelines contained in WCAP-13854 submitted and approved for the Catawba plant. The updated Byron/Braidwood guidelines were used in support of the Braidwood A1R04 SG inspection. These guidelines contain, in part, bobbin and RPC specifications, calibration requirements, specific acquisition and analysis criteria, and flaw recording requirements. The Byron/Braidwood inspection guidelines are summarized in Attachment B. The current revision of the complete guidelines are also provided for information as Attachment I. These guidelines are routinely updated to address technology advancements and industry experience to assure the most up to date and appropriate inspection techniques are applied at ComEd stations.

Data Analyst Training and Qualification

The existing Byron data analyst training and qualification program has been revised to include enhanced ODSCC defect identification and sizing. The program includes a review and discussion of the Byron/Braidwood inspection guidelines,

familiarization with Byron SG configuration and degradation history, and experience with analyzing actual plant data from previous inspections. The data analysts must pass a site specific performance demonstration that contains samples of all damage mechanisms experienced at Byron, with special emphasis on ODSCC. Qualified Data Analysts (QDAs), examined by the EPRI performance demonstration program, will be used whenever possible. QDAs will also receive the Byron site-specific training and examination.

The enhanced Byron training program and use of QDAs provides added assurance that the eddy current inspection data is reliably and consistently analyzed.

B. DEFENSE IN DEPTH

As a result of a few large indications observed at the end of Braidwood Cycle 4, two additional analysis methodologies have been pursued by ComEd for both Byron and Braidwood Stations to provide a "defense in depth" approach to IPC. Although these methodologies are not used as the bases for the Byron Unit 1 IPC, they provide added assurance that margin is available beyond that relied upon for adequacy of the IPC approach.

- Evaluation of the limited TSP displacement during a main steamline break event (MSLB).
- Evaluation of a higher POD for larger voltage amplitude indications.

As further assurance of the low impact of IPC on the risk of core damage, an evaluation was performed to compare containment bypass core damage frequency with and without application of IPC at Byron Unit 1.

Limited Tube Support Plate Displacement

ODSCC degradation experienced at Byron Unit 1 is confined within the bounds of the tube support plate region. The support plate provides additional structural constraint to strengthen the tube to mitigate burst. However, during transient conditions such as a MSLB, the support plates displace due to thermal hydraulic loading and expose the crack to free span conditions, consequently increasing tube burst margins. For this reason, the Byron IPC approach conservatively assumes the freespan condition of ODSCC when assessing burst capabilities.

However, added assurance of increased tube burst margin is gained when considering the actual tube support plate (TSP) movement relative to ODSCC indications. With limited TSP displacement, portions of the ODSCC affected area of the tube remains within the confinement of the TSP and the freespan exposed crack lengths are reduced. Section 4.0 of WCAP-14046 describes the limited

displacement analysis method and the results for Model D-4 steam generators used for the Braidwood Unit 1 IPC submittal. This analysis and the Model D-3 analysis used in the Catawba submittal are currently under NRC review. Preliminary analysis results indicate that significant margin against tube burst is realized with implementation of the limited displacement analysis over the free span assumption.

Variable Probability of Detection

Probability of Detection (POD) of eddy current indications is an important consideration in the development and implementation of IPC. The POD is used to adjust the beginning of cycle (BOC) voltage distribution to account for indications not detected. The voltage distribution of detected indications is scaled up by a factor of 1/POD and tubes repaired are then subtracted to form the assumed population and voltage distribution of the next operating cycle. The adjusted BOC voltage distribution is used in tube leak and burst assessments in support of the IPC. Since the leakage and burst evaluations can be affected significantly by the assumed voltage distribution, the POD value is critical.

The Byron IPC approach uses a constant POD value of 0.6 for all indications, regardless of voltage amplitude. This is in accordance with the guidance given in draft NUREG-1477 and other approved IPCs. However, data collected as part of the EPRI Performance Demonstration Program indicates that the POD is actually greater than 0.6 for indications greater than 1.0 volt (refer to Figure D-1). This means that the use of a constant POD of 0.6 for all voltage amplitudes can over estimate the actual BOC distribution and consequently over estimate the leak rate and burst probability.

ComEd has a high level of confidence that larger amplitude indications be identified with a higher POD, as implied by data collected during the previous Byron and Braidwood inspections, as well as through the work by EPRI. Both Byron and Braidwood lack copper in the secondary system that could mask or hinder detection of flaws, therefore the POD should be higher. To support this contention, ComEd is evaluating the possibility of a "blind" re-analysis of Byron and Braidwood data to define a POD curve for a plant without copper in the secondary system. Until this is accomplished or other industry data is obtained, the Byron IPC approach will conservatively use the draft NUREG-1477 guidance on POD.

Risk Evaluation Of Core Damage

As part of ComEd's evaluation of the operability of Byron Unit 1 Cycle 7, an alternate repair criteria (ARC) risk evaluation will be completed. The objective of this evaluation will be to compare core damage frequency, with containment bypass, with and without the interim plugging criteria applied at Byron Unit 1.

As a preliminary assessment of the impact of ARC for ODSCC at Byron, a comparison was made to the Braidwood Unit 1 risk evaluation. The assumption was made that the beginning of cycle size distribution and the growth rates are those observed at Braidwood Unit 1 End of Cycle (EOC) 4. This is believed to be an extremely conservative assumption, the result of which would be to bound the potential impact on core damage frequency. This calculation was done assuming both a constant probability of detection of 0.6 and a variable probability of detection of 0.6 for indications smaller than 3 volts, and a POD of 1 for indications greater than or equal to 3 volts. It is important to remember that both the growth rate and the largest size of indications observed at Byron Unit 1 are substantially smaller than those observed in Braidwood Unit 1 EOC 4.

ComEd has evaluated the impact on operation of using the proposed interim plugging criteria against the results of insights from the Byron Individual Plant Examination (IPE). The ComEd evaluation parallels that described in the NRC Staff's Safety Evaluation Report for Palo Verde Unit 2, dated August 19, 1993.

The total core damage frequency for Byron is estimated to be 3.09E-5 per reactor year, with a total contribution from containment bypass sequences of 3.72E-8 per reactor year in the current IPE. Operation with the alternate repair criteria with a variable POD is expected to increase the MSLB with containment bypass sequence frequency contribution by a factor of only about 10%. An upper bound increase of a factor of two is derived when the fixed POD of 0.6 is employed in the calculation. Neither increase is significant from a risk perspective.

The reason for a reduced core damage frequency with a higher POD is that large voltage indications have a high assurance of being identified and removed from service during inspection. Therefore, the calculation of burst probability during MSLB changes because of differences in the assumed distribution of indications left in service at BOC. The EOC burst probability also changes because the growth distribution is added to the new BOC distribution of indications. The result of this change is a significant reduction in burst probability during MSLB.

Therefore, the operation of Byron Unit 1 Cycle 7 for a complete eighteen (18) month fuel cycle, with the application of the 1.0 volt IPC, will not significantly increase the core damage frequency, even with the conservative assumption of a POD of 0.6 and application of the full growth rate distribution observed during Cycle 6.

TABLE D-1

OPERATOR ACTIONS
PLANT SHUTDOWN REQUIREMENTS:

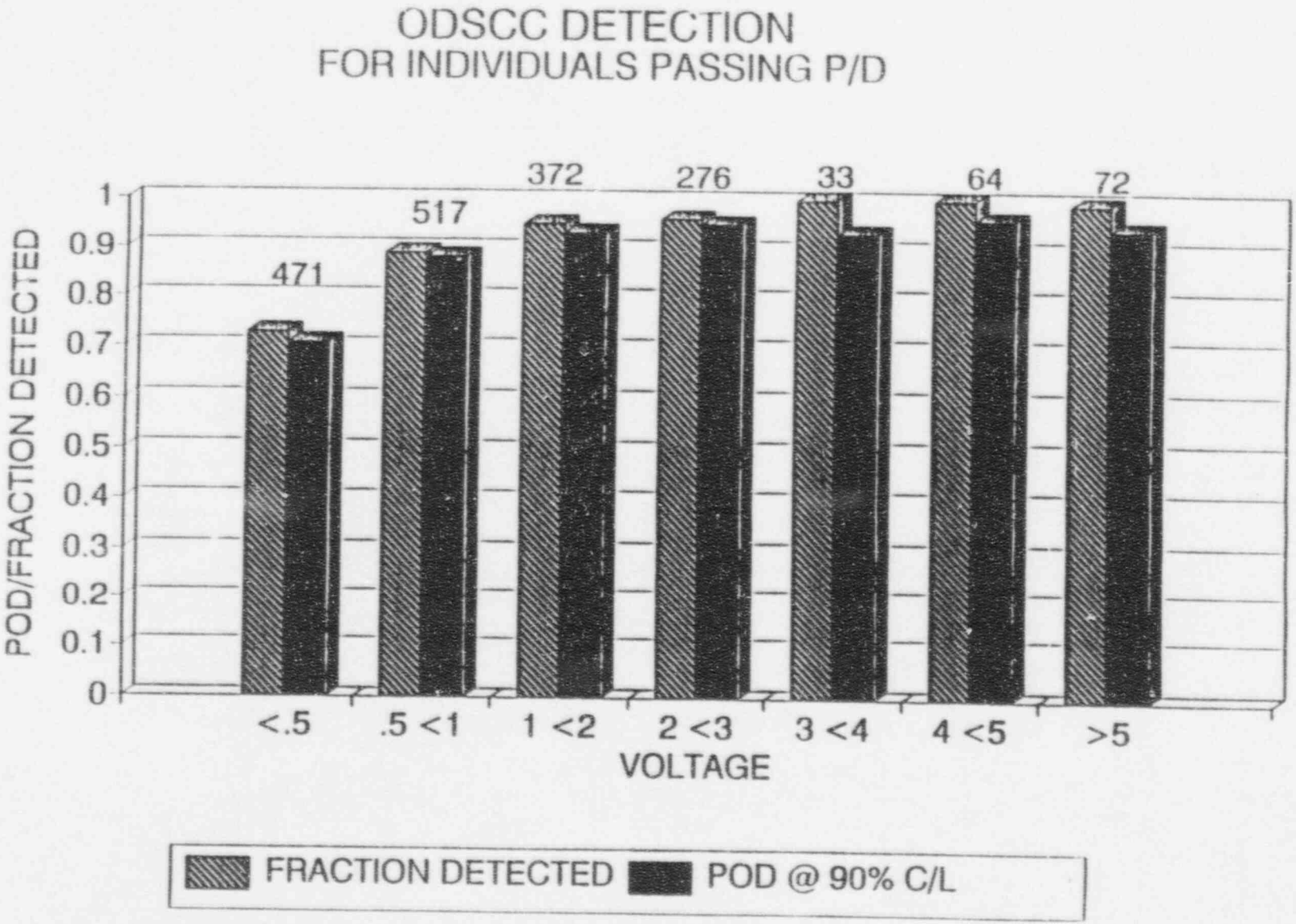
a. Leak rates:

Leak Rates	Time to perform plant shutdown
SG leakage greater than 10 gpm	Be shutdown within 4 hours GO TO BOA SEC-8 step 8 and continue in procedure. Refer to T.S. 3.4.6.2.
SG leakage greater than or equal to 150 gpd but less than 10 gpm	Be shutdown within 10 hours and perform normal cooldown per BGPs. Refer to T.S. 3.4.6.2.
Leak rate less than 150 gpd	Continue monitoring leak rate. Perform BOP MS-11, "Operation with Steam Generator Tube Leakage."

b. Change in leak rate:

Change in leak rate	Time to perform plant shutdown
Change in any SG leak rate of greater than 100 gpd in 1 hour and leak rate greater than 50 gpd	Be shutdown within 4 hours GO TO BOA SEC-8 step 8 and continue in procedure. Refer to T.S. 3.4.6.2.
Change in any SG leak rate of 25 to 100 gpd in 1 hour and leak rate greater than 50 gpd	Be shutdown within 5 hours and perform normal cooldown per BGPs. Refer to T.S. 3.4.6.2.
Change in any SG leak rate of less than 25 gpd in 1 hour AND leak rate less than 150 gpd	Continue monitoring leak rate. Perform BOP MS-11, "Operation with Steam Generator Tube Leakage."

FIGURE D-1:


ODSCC DETECTION
FOR INDIVIDUALS PASSING P/D
POD/FRACTION DETECTED
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
471
517
372
276
33
64
72
<.5
.5 <1
1 <2
2 <3
3 <4
4 <5
>5
VOLTAGE
FRACTION DETECTED
POD @ 90% C/L

ATTACHMENT E

PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37 AND NPF-66

BYRON STATION UNIT 1
REVISED PAGES:

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3/4 4-14
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3/4 4-18*
3/4 4-19*
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*THESE PAGES HAVE NO CHANGES BUT ARE INCLUDED FOR CONTINUITY.