October 16, 2000

MEMORANDUM TO: Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: M. Christopher Nolan, Project Manager, Section 1 /RA/
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF THE JULY 26, 2000, MEETING HELD WITH THE NUCLEAR ENERGY INSTITUTE (NEI) REGARDING THE STATUS OF NEI 97-06, "STEAM GENERATOR PROGRAM GUIDELINES"

On July 26, 2000, members of the Nuclear Regulatory Commission (NRC) staff met with NEI and representatives of Westinghouse, the Electric Power Research Institute (EPRI), and individual licensees to discuss the current status of NEI 97-06 and proposed changes to their generic license change package regarding steam generators (SGs). In addition, NEI sponsored some technical discussions to address previously identified NRC concerns regarding eddy current (EC) noise, EC data quality, inspection of U-bends, use of high frequency EC probes, and SG tube burst testing. For example, data was presented to demonstrate that the level of EC noise in the Indian Point Nuclear Generating Station, Unit 2 (IP-2), U-bends is not representative of conditions at certain other plants of similar design and vintage. Industry representatives also indicated that EPRI SG inspection guidelines will be strengthened with respect to site validation requirements, including noise considerations. The NRC and NEI further discussed the time dependency of tube burst pressure data and its implications regarding the accuracy of SG tube burst models. NEI indicated that the industry was aware that resolution of these technical issues would facilitate the NRC staff's review of the industry's program for managing SG tube degradation. NEI then stated its intention to provide feedback to the industry regarding lessons learned from recent industry experience at IP-2, and Arkansas Nuclear One, Unit 2. NEI has not determined the specific method or timing for providing this feedback. This meeting was held at the NRC Headquarters in Rockville, Maryland. Enclosure 1 is a list of meeting attendees. The presentations used during this meeting are also included as enclosures to this meeting summary and are listed below.

Project No. 689

Attachments:
1) List of Meeting Attendees
2) NEI SG Program Status
3) NEI's Draft Changes to the Generic License Change Package
4) EPRI's SG Program Generic License Change Package Status
5) NEI's U-Bend Noise Study
6) Westinghouse's SG Tube Burst Testing Status

cc w/atts: See next page

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OFFICE	PDIV-1/PM	PDIV-1/LA	EMCB/SC*	PDIV-1/SC
NAME	C.Nolan	D.Johnson	T.Sullivan	R.Gramm for
DATE	10/10/00	10/6/00	09/28/00	10/11/00

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ATTENDANCE LIST

PUBLIC MEETING HELD ON July 26, 2000

Name	Organization	Telephone No.
Chris Nolan	NRC/NRR	301-415-1320
Louise Lund	NRC/NRR	301-415-2786
Steve Long	NRC/NRR	301-415-1077
Joe Donoghue	NRC/NRR	301-415-1131
Jack Goldberg	NRC/OGC	301-415-1681
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Jim Riccio	Public Citizen	202-546-4996
Dave Lochbaum	Union of Concerned Scientists	202-332-0900
Deann Ralegh	Search, Bechtel Power	301-228-6312
Forrest Hundley	Southern Company	205-992-6998
Rick Mullins	Southern Company	205-992-5502
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Ken Karwoski	NRC/RES	301-415-5356
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Scott A. Redner	Northern States Power	612-630-4259
Jim Riley	Nuclear Energy Institute	202-739-8137

DISTRIBUTION FOR MEETING SUMMARY

Dated: October 16, 2000

PUBLIC
PD# IV-1 Reading
RidsOgcRp
RidsAcrsAcnwMailCenter
RidsNrrDlpm (J. Zwolinski/S. Black)
RidsNrrDe (J. Strosnider)
RidsNrrDeEmcb (W. Bateman)
RidsNrrPMTAlexion
RidsNrrPMMNolan
RidsNrrLADJohnson
E. Sullivan
E. Murphy
L. Lund
S. Long
J. Donoghue
J. Goldberg
J. Shea
R. Ennis
J. Yerokun, R-I
A. Rubin
K. Karwoski
T. Madden
D. Lang, EDO
RidsRgn4MailCenter (K. Brockman)
C. Beardslee Khan

Steam Generator Program Status

SGTF / NRC Meeting

July26, 2000

Jim Riley, NEI

202-739-8137, jhr@nei.org

Agenda

- Generic License Change Package Status
- Tech Spec and TRM Changes
- NEI 97-06 Changes
- ANO Scenario
  - Status of Pressure Test Results Study
  - Implications on EPRI Guidelines
- NRC Questions Regarding IP2
- Remaining Activities

Attachment 2

Generic License Change Package Status

- Generic TS template submitted to NRC on 2/4/00
- Requested a commitment from each PWR to implement proposed TS changes.
  - 69 / 69 PWR sites committed to implement TS changes as originally proposed after NRC endorsement
  - Revised package will be reviewed
- Evaluating the impact of recent industry events on the package

NEI

Generic License Change Package Status

- Completion of NRC review delayed to at best 12/31/00
- Revision to package created because of:
  - EPRI PSL G/L Rev 2
  - TSTF comments
- Submittal to NRC in September after industry review

NEI

Tech Spec Changes

- Changes to Administrative section on Steam Generator Program
  - Addresses TSTF comments - uses standard wording (avoids the use of the term "licensee")
  - Provisions of SR 3.0.2 are applicable to the primary-to-secondary leakage test frequencies

NEI

Tech Spec Changes

- Operational leakage LCO:
  - Action statement clarified to remove 4 hour delay to S/D for primary-to-secondary leakage
  - Bases changed to clarify relationship of 150 gpd per SG to safety analysis
  - Bases changed to express difficulty in measuring low primary-to-secondary leakage rates when shutdown

NEI

TRM Changes

- Major reasons for changes:
  - Wording standardized and clarified
  - LCO and actions did not match
  - Reporting requirements not clear
- Resulting changes:
  - "Repair Criteria" added to LCO
  - Modes 5 and 6 added to applicability
  - SG Operability clarified

NEI

TRM Changes

- Resulting changes (continued):
  - Reporting requirements clarified in accordance with the proposed version of 50.72 / 73
  - Surveillance for tube repair added and Note allowing entry into Mode 5 and 6
  - TRM Bases added to document reasons for requirements

NEI

NEI 97-06 Changes

- Incorporates improvements identified during the development of the Generic License Change Package without tie to proposed TS
- Draft rev 1 distributed for industry comment on 6/26 via an APC letter
- No substantive changes to the requirements as compared to the 2/4/00 draft rev 1
- 2/4/00 revision will be issued as rev 2 after NRC endorses generic / plant specific TS changes

NEI

NEI 97-06 Changes

- Clarifies the relationship between the safety analysis assumption and the Operational Leakage Criterion limit
- Clarified the Accident Induced Leakage criterion with respect to severe accident considerations

NEI

Pressurization Rate Study

- Project status - Mati Merilo/ EPRI
- Implications on EPRI Guidelines?

NEI

IP2 Event

- Industry actions on NRC NDE questions - Dan Mayes
- Industry Noise Study - Scott Redner
- Implications on EPRI Guidelines?
- Disseminating lessons learned / interim guidance

NEI

Remaining Activities

- Issue Rev 1 of NEI 97-06
- Complete industry review of Generic License Change Package and submit
- Complete Pressurization Rate Study
- SGTF / NRC meeting
- Senior management meeting

NEI

Revision to the Template for an Administrative Section Technical Specification for a Steam Generator Program

Draft 2

Attachment 3

5.5.9 Steam Generator Program

A. General Requirements

A Steam Generator Program shall be established and implemented to ensure that steam generator tube integrity is maintained. Steam generator tube integrity is maintained by meeting the performance criteria as defined in the Steam Generator Program.

The provisions of SR 3.0.2 are applicable to the primary-to-secondary leakage test frequencies specified in the Steam Generator Program.

B. Condition Monitoring Assessment

Condition Monitoring Assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural and accident leakage integrity. The "as found" condition refers to the condition of the tubing during a steam generator inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition Monitoring Assessments shall be conducted during each outage during which the steam generator tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met. Requirements for condition monitoring are defined in the Steam Generator Program.

C. Performance Criteria

The steam generator performance criteria are defined in the Steam Generator Program. **Revisions** to performance criteria (and their associated definitions as used in the Steam Generator Program) **require** review and approval by the NRC. **The performance criteria** (and their associated definitions as used in the Steam Generator Program) **may be revised** to incorporate changes approved generically by the NRC subject to the limitations and conditions set forth in the staff's approving document. **Demonstration of** satisfaction of the generic limitations and conditions must be documented in a safety evaluation prepared in accordance with 10 CFR 50.59.

D. Tube Repair Criteria and Repair Methods

Tube repair criteria and repair methods shall be described in and implemented by the Steam Generator Program. **Repair criteria and repair methods may be implemented after review and approval by the NRC. In addition, repair criteria and repair methods approved generically by the NRC may be used subject to the limitations and conditions set forth in the staff's approving document. Demonstration of** satisfaction of the generic limitations and conditions

must be documented in a safety evaluation prepared in accordance with 10 CFR 50.59. Note that tube plugging is not a repair and does not need to be reviewed or approved by the NRC.

5.6.10 Steam Generator Tube Inspection Report

If the results of the steam generator inspection indicate greater than 1% of the inspected tubes in any steam generator exceed the repair criteria in accordance with the requirements of the Steam Generator Program, a Special Report shall be submitted within 120 days after the initial entry into Mode 4 following completion of the inspection. The report shall summarize:

a. The scope of inspections performed on each steam generator inspected in the affected unit during the current outage,
b. Active degradation mechanisms found,
c. NDE techniques utilized for each degradation mechanism,
d. Location, orientation(if linear) and measured sizes of service induced indications,
e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,
f. Repair method utilized and the number of tubes repaired by each repair method,
g. Total number and percentage of tubes plugged and/or repaired to date,
h. The effective plugging percentage for all plugging and tube repairs in each steam generator, and
i. The results of condition monitoring including the results of tube pulls and in-situ testing.

Template for an Operational Leakage Technical Specification

Revision to NRC Submittal
Draft 4

Changes as of 7/25/00

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

~~d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and~~

d. ~~e.~~ [~~500~~] 150 gallons per day primary to secondary LEAKAGE through any one ~~SG.~~ Steam Generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS Operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE ~~.~~ or primary to Secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to Secondary LEAKAGE not within limits	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	--------------------NOTE-------------------- Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. ----------NOTE---------- Not applicable to primary to secondary LEAKAGE -------------------- Perform RCS water inventory balance.	-----NOTE------ Only required to be performed during steady state operation -------------- 72 hours
SR 3.4.13.2	Verify ~~steam generator tube integrity~~ primary to secondary LEAKAGE is in accordance with the operational leakage performance criterion described in the Steam Generator ~~Tube Surveillance~~ Program.	In accordance with the Steam Generator ~~Tube Surveillance~~ Program

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~a 1 gpm primary to secondary LEAKAGE as the initial condition~~.

Insert "A"

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE is relatively inconsequential.

assumption in the safety analysis

The [SLB] is more limiting ~~for~~ site radiation releases. The safety analysis for the [SLB] accident assumes [1 gpm] primary to secondary LEAKAGE in ~~one~~ generator as an ~~initial~~ condition. The dose consequences resulting from the [SLB] accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

[One gallon per minute (gpm)] of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

Delete

~~d.~~ Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

d ~~e~~. Primary to Secondary LEAKAGE through Any One SG

Insert B

~~The~~ [500] gallons per day limit on one SG is based ~~on~~ the assumption that a single crack leaking ~~this~~ amount would not propagate to a SGTR ~~under~~ the stress conditions of a LOCA ~~or a~~ main steam line rupture. If leaked ~~through~~ many cracks, ~~the~~ cracks are very small, ~~and~~ the above assumption is conservative.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE~~,~~ or identified LEAKAGE~~, or primary to secondary LEAKAGE~~ in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if primary to secondary LEAKAGE is not within limits, or if unidentified LEAKAGE, or identified LEAKAGE, ~~or primary to secondary LEAKAGE~~ cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

Insert "C"

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.13.2

Insert "D"

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [15].

Insert A

APPLICABLE SAFETY ANALYSIS:

that primary to secondary LEAKAGE from all steam generators is [one gallon per minute] or increases to [one gallon per minute] as a result of accident induced conditions. The Technical Specification LCO requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the condition assumed in the safety analysis.

Insert B

Primary to Secondary Leakage through any one SG

The limit of 150 gallons per day per steam generator is based on operating experience gained from SG tube degradation mechanisms which result in tube leakage. This leakage rate along with the other performance criteria in the Steam Generator Program provide reasonable assurance that a single flaw leaking this amount will not propagate to an SGTR under the stress conditions of a LOCA or a main steam line rupture prior to detection by leakage monitoring methods and commencement of plant shutdown. If leaked through many flaws, the flaws are very small and the above assumption is conservative.

The leakage rate limit applies to leakage in any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the leakage should be conservatively assumed to be from one steam generator. However, plants with N-16 monitors on individual steam lines can quantify leakage in any one steam generator.

The RCS Operational primary to secondary leakage is measured at standard temperature and pressure.

Insert C

SR 3.4.13.1:

A note under the surveillance column states that this SR is not applicable to primary-to-secondary leakage because leakage limits as low as 150 gallons per day cannot be measured accurately by an RCS water inventory balance. In addition, a surveillance frequency of less than 72 hours is important to ensure adequate protection against rapidly increasing SG tube leaks.

Insert D

SR3.4.13.2:

This SR requires that the Operational Leakage performance criterion in the Steam Generator Program be satisfied. Satisfying the Operational Leakage performance criterion ensures that the primary to secondary LEAKAGE limit is met. The Operational Leakage performance criterion along with the other performance criteria in the Steam Generator Program provide reasonable assurance against tube burst at normal and faulted conditions. The 150 gallons per day limit is measured at standard temperature and pressure.

Primary to secondary leakage is determined through the analysis of secondary coolant activity levels. At low power, primary and secondary coolant activity is sufficiently low that an accurate determination of primary to secondary leakage may be difficult. Immediately after shutdown, the short lived isotopes are usually at sufficient levels to monitor for leakage by normal power operational means as long as other plant conditions allow the measurement. During startup, especially after a long outage, there are no short lived isotopes in either the primary or secondary system. This limits measurement of the leakage to chemical or long lived radiochemical means. The Steam Generator Program provides guidance on leak rate monitoring during MODES 3 and 4.

Template for a Licensee Controlled Document
[Technical Requirements Manual]

Revision to NRC Submittal

Draft 7

The <u>content</u> of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

TRM x.y **<u>Steam Generators</u>**

LCO

Each steam generator shall meet primary to secondary pressure boundary integrity Performance Criteria and Repair Criteria as specified below.

A. <u>Performance Criteria</u>

(i) <u>Structural criterion</u>:
Steam Generator Tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a margin of 3.0 against Burst under Normal Steady State Full Power Operation and a margin of 1.4 against Burst under the Limiting Design Basis Accident, concurrent with a safe shutdown earthquake.

(ii) <u>Accident Induced Leakage criterion</u>:
The primary to secondary Accident Induced Leakage rate for the Limiting Design Basis Accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed [1 gpm per steam generator, except for specific types of degradation at specific locations where the tubes are confined. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria].

(iii) <u>Operational leakage criterion</u>:
Requirements related to the Operational Leakage criterion are delineated in the RCS Operational LEAKAGE Technical Specification.

The content of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

B. Repair Criteria

Repair Criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging. Repair Criteria approved for use at [Plant] are:

- [40%] nominal tube wall thickness
- [Other Repair Criteria that are currently approved for use – list.]

APPLICABILITY: MODE 1, 2, 3, 4, 5, and 6.

[CONTINGENCY MEASURES:]

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Failure to implement a required plugging or repair discovered in MODES 1, 2, 3, or 4.	A.1 Notify the NRC of the failure to implement required plugging or repair.	8 hours
	AND	
	A.2 Determine steam generators remain acceptable for continued operation based on meeting the Performance Criteria.	In accordance with [the licensee's corrective action program]
	AND	
	A.3 Submit a special report to the NRC providing the basis for the planned operating period.	60 days
B. Required Action A.2 and associated Completion Time not	Declare the SG inoperable and enter the appropriate Conditions and Required Actions in the TS.	Immediately

The content of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

CONDITION	REQUIRED ACTION	COMPLETION TIME
met		
---------------NOTE------------- 1. Failure to meet the Performance Criteria in MODES 5 or 6 does not render the Steam Generator inoperable. 2. Required Actions C.2, C.3, and C.4 are required to be completed for every entry into Condition C. -------------------------------------- C. Performance Criteria A.(i) or A.(ii) not met as discovered in MODES 5 or 6.	C.1 Plug or repair the tubes in accordance with Repair Methods <u>AND</u> C.2 Notify the NRC of the failure to meet the Performance Criteria. <u>AND</u> C.3 Submit a special report to the NRC. <u>AND</u> C.4 Submit a special report to the NRC describing the basis for the planned operating period.	Prior to entering MODE 4 8 hours 60 days 120 days after entering MODE 4
---------------NOTE------------- Failure to meet the Repair Criteria in MODES 5 or 6 does not render the Steam Generator inoperable. -------------------------------------- D. Repair Criteria not met as discovered in MODES 5 or 6.	Plug or repair the tubes in accordance with Repair Methods	Prior to entering MODE 4

The content of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

[VERIFICATION REQUIREMENTS]

VERIFICATION		FREQUENCY
SR A.	Verify steam generator tube integrity is in accordance with the Performance Criteria described in the Steam Generator Program.	In accordance with the Steam Generator Program
-------------------------NOTE------------------------ Not required to be met in MODES 5 or 6 -- SR B.	Verify that SG tubes that exceed the Repair Criteria are plugged or repaired in accordance with Repair Methods.	In accordance with the Steam Generator Program

TRM Steam Generator Integrity

BASES

BACKGROUND

The purpose of the steam generator integrity LCO is to require compliance with the steam generator Performance Criteria and repair criteria. The steam generator Performance Criteria defines the basis for steam generator OPERABILITY. The Performance Criteria apply to steam generator tubes and associated appurtenances (e.g. plugs, sleeves, and other repairs). The repair criteria are intended to provide reasonable assurance that the Performance Criteria will not be exceeded. The Performance Criteria, Repair Criteria, and the processes required to meet them are addressed by the Steam Generator Program.

Satisfying the Performance Criteria provides reasonable assurance of tube integrity at normal and faulted conditions. Steam generator tube integrity means that the tubes are capable of performing their intended safety functions consistent with their licensing basis, including applicable regulatory requirements.

The steam generator (SG) tubes in pressurized water reactors have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. Finally, the SG tubes may be relied upon to maintain their integrity under conditions resulting from core damage severe accidents consistent with the containment objectives of preventing uncontrolled fission product release.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively.

BASES (continued)

The steam generator Performance Criteria identify the standards against which performance is to be measured. Meeting the Performance Criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity.

APPLICABLE SAFETY ANALYSIS

Satisfying the Performance Criteria provides reasonable assurance against tube Burst and the resulting primary to secondary leakage that might occur at normal and faulted conditions. The consequences of design basis accidents that include primary to secondary leakage are, in part, functions of the accident induced primary-to-secondary leakage rates and the dose equivalent I^{131} in the primary coolant.

The typical analysis for an event resulting in steam discharge to the atmosphere, except a steam generator tube rupture (SGTR), assumes that primary-to-secondary leakage for all steam generators is [1 gallon per minute] or increases to [1 gallon per minute] as a result of accident induced stresses. For accidents that do not involve fuel damage, the reactor coolant activity levels of dose equivalent I^{131} are at the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the accident conditions.

For most PWRs, the SGTR accident is the limiting design basis event that establishes limits for these parameters. In the analysis of a SGTR event, a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in the technical specifications plus the leakage rate associated with a double-ended rupture of a single tube is assumed. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is steamed to the main condenser.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The leakage is assumed to be at the design basis value, which is consistent with the accident induced leakage performance criterion. Due to the large amount of fluid

BASES (continued)

that is released during a SLB, it is usually more limiting for site radiation releases.

The steam generator Performance Criteria in this document and the limits included in the plant technical specifications for operational leakage and for dose equivalent I^{131} in primary coolant ensure the plant is operated within its analyzed condition. The dose consequences resulting from the most Limiting Design Basis Accident are within the limits defined in GDC 19 [1], 10 CFR 100 [2] or the NRC approved licensing basis (i.e., a small fraction of these limits).

LCO

The LCO requires that steam generator Performance Criteria and tube Repair Criteria be met.

The steam generator Performance Criteria include design basis parameters that define acceptable steam generator performance. Steam generator OPERABILITY is based on meeting the Performance Criteria.

Steam generator tube Repair Criteria are established to define the point at which tubes must be repaired or removed from service by plugging to ensure that the Performance Criteria are not exceeded. Exceeding the Repair Criteria does not necessarily mean that the steam generator is inoperable.

Changes to the Performance Criteria or Repair Criteria are controlled by the license Technical Specifications.

A. Performance Criteria

The SG Performance Criteria are based on tube structural integrity, accident-induced leakage, and operational leakage. Compliance with the structural integrity and accident induced leakage performance criteria can only be determined during steam generator inspections. These inspections are performed when the reactor is shutdown and depressurized so that the primary side of the SG can be drained.

(i) Structural Criterion

BASES (continued)

The Structural Criterion is:

"Steam Generator Tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a margin of 3.0 against Burst under Normal Steady State Full Power Operation and a margin of 1.4 against Burst under the Limiting Design Basis Accident, concurrent with a safe shutdown earthquake."

Steam Generator Tubing refers to the entire length of the tube between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

In the context of the Structural Criterion, Limiting Design Basis Accident is defined as the accident that results in the largest differential pressure.

The Structural Criterion can be broken into two separate considerations:

- Providing a margin of safety against tube Burst under normal and accident conditions, and
- Ensuring structural integrity (preventing yield or Burst) of the SG tubes under all anticipated transients included in the design specification.

Tube Burst

Tube Burst is defined as the gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.

The Structural Criterion provides reasonable assurance that a steam generator tube will not Burst

during normal or postulated accident conditions. The Structural Criterion requires that the tubes not Burst when subjected to differential pressures equal to three (3) times those experienced during normal steady state operation and 1.4 times accident differential pressures concurrent with a safe shutdown earthquake. The safety factors of 3 and 1.4 are based on ASME Code Section III subsection NB [7] requirements and Draft Regulatory Guide 1.121 [8] guidance.

For most plants the Normal Steady State Full Power Operation condition defines the most limiting parameters under which the tubes are tested. In the context of the Structural Criterion, Normal Steady State Full Power Operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{hot} should be assessed and their effects on differential pressure should be accounted for if significant. Guidance on accounting for changes in these parameters is provided in the EPRI Integrity Assessment Guidelines [6].

<u>Tube Yield</u>

The Structural Criterion verifies that the primary pressure stresses do not exceed the yield strength for the full range of normal operating conditions including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification. All appropriate loads contributing to combined primary plus secondary stress are evaluated so as to ensure that these loads do not significantly reduce the Burst pressure for the full range of normal operating conditions including postulated accidents. For example, axial loads due to tube-to-shell temperature differences in once-through steam generator designs during postulated MSLB, or axial loading associated with locked tube supports in recirculating steam generator designs are addressed to

ensure that the test conditions are at least as severe as those expected during operating and accident events.

(ii) <u>Accident Induced Leakage Criterion</u>

The Accident Induced Leakage Criterion is:

"The primary to secondary Accident Induced Leakage Rate for the Limiting Design Basis Accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed [1 gpm per steam generator, except for specific types of degradation at specific locations where the tubes are confined. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria]."

In the context of the Accident Induced Leakage Criterion:

- Accident Induced Leakage Rate means the primary-to-secondary leakage occurring during postulated accidents other than a steam generator tube rupture. This includes the primary-to-secondary leakage rate existing immediately prior to the accident plus additional primary-to-secondary leakage induced during the accident.
- Limiting Design Basis Accident is defined as the accident that results in the largest dose.

The Accident Induced Leakage Criterion can be broken down into two separate considerations:

- Meeting design basis conditions, and
- Resisting severe accident conditions.

<u>Design Basis</u>

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a Limiting Design Basis Accident. The radiological dose consequences resulting from a potential primary-to-

secondary leak during postulated design basis accidents must not exceed the offsite dose limits required by 10 CFR Part 100 [2] or the control room personnel dose limits required by GDC-19 [1] or the NRC approved licensing basis.

In most cases when calculating offsite doses, the safety analysis for the Limiting Design Basis Accident, other than a steam generator tube rupture, assumes a total of [1 gpm] primary to secondary leakage as an initial condition. Plant specific assumptions for accident induced leakage are defined in each licensee's licensing basis. The leakage value used in the Accident Induced Leakage Criterion must be consistent with the licensing basis.

Severe Accidents

Although severe accident conditions are not part of plant licensing basis, the NRC has stated that their probabilistic safety analysis sensitivity studies have shown that severe accident risk is sensitive to certain design basis parameters such as 1 gpm accident induced leakage per SG. As a result, leakage greater than the design basis or 1 gpm per steam generator is not allowed unless the NRC has approved greater leakage rates as part of an Alternate Repair Criterion.

(iii) Operational Leakage Criterion

The Operational Leakage Criterion and its associated action and surveillance requirements are contained in the RCS Operational Leakage Technical Specification. The Operational Leakage Criterion is not included in the TRM to avoid duplication with the license Technical Specifications. The following summary of the Operational Leakage requirements is provided to facilitate an understanding of all of the Performance Criteria since they act together to define SG OPERABILITY.

The Operational Leakage Criterion is:

BASES (continued)

> *"The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day."*

Plant shutdown should commence if primary-to-secondary leakage exceeds 150 gallons per day (GPD) from any one steam generator. Operational leakage is measured at standard temperature and pressure conditions.

The limit of 150 gallons per day per steam generator (the plant limit is 150 gpd times the number of SGs in the unit) is based on operating experience gained from SG tube degradation mechanisms which result in tube leakage. This leakage rate along with the other Performance Criteria in the Steam Generator Program provide reasonable assurance that a single flaw leaking this amount will not propagate to an SGTR under the stress conditions of a LOCA or a main steam line rupture prior to detection by leakage monitoring methods and commencement of plant shutdown. If leaked through many flaws, the flaws are very small and the above assumption is conservative.

B. Repair Criteria

Repair Criteria are those NDE measured parameters at or beyond which a tube must be repaired or removed from service by plugging. Tube Repair Criteria are established to prevent the SG tubes from exceeding the Performance Criteria.

The tube Repair Criteria establish limits for tube degradation that provide reasonable assurance that an affected tube will meet the Performance Criteria at the next scheduled inspection by allowing for anticipated growth during the intervening time interval. Because of this allowance for growth, exceeding a tube repair criterion does not necessarily render the steam generators inoperable.

Tube Repair Criteria are either the existing technical specification through-wall (TW) depth-based criterion (i.e., 40% TW for most plants), or other Alternate Repair Criteria

BASES (continued)

(ARC) approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05.

The depth base criterion, approved for use at all plants by the NRC, was established when the most frequent form of degradation was general wastage corrosion. This type of degradation is characterized by a volumetric loss of the tube wall. This criterion was established to allow for NDE uncertainties and growth and still provide a reasonable assurance that the affected tube would not fail in the event of an accident. Additional basis information is provided in Draft Regulatory Guide 1.121.

In recent years, improved inspection techniques, knowledge of corrosion mechanisms, and experience have revealed additional types of tube degradation in the form of cracks in the tube wall. In some instances, a reliable method of characterizing specific types of cracks at defined locations within certain steam generator designs has been developed. In these cases, the industry has developed, and the NRC has approved Alternate Repair Criteria (ARC) to permit leaving a tube in service (as opposed to plugging) when the tube has indications that fall within the limits established by the ARC. Plug or repair on detection is not an ARC.

The NRC must approve all Repair Criteria prior to use. Requirements for approval of changes to the Repair Criteria are contained in the license Technical Specifications.

The Repair Criteria approved for use at [Plant] are:

- [40%] nominal tube wall thickness
- [...]

APPLICABILITY

Steam generator tube integrity is required during all MODES; therefore the TRM requirements are applicable in MODES 1 through 6.

Steam generator tubes are designed to withstand the stresses due

BASES (continued)

to differential pressures as large as 3 times those experienced under normal full power operations or 1.4 times those experienced during a Limiting Design Basis Accident. This requirement is delineated in the Structural Criterion. This magnitude of differential pressure is only possible during MODES 1, 2, 3, and 4. When the reactor is in MODES 5 and 6, primary to secondary differential pressure and primary coolant activity are so low that primary to secondary leakage is not a significant safety concern. As a result, meeting the Performance Criteria is only required for SG OPERABILITY in MODES 1 through 4.

In MODES 5 and 6 reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for leakage. During these MODES the Performance Criteria are still required to be met but do not form a basis for SG OPERABILITY.

ACTIONS

A. Implementation of Repair Criteria can only be accomplished during SG inspections that are performed when the reactor plant is shutdown. If any plugging or repair of tubing is required, it is completed before the plant enters an operating MODE.

If a required plugging or repair was not implemented during the last inspection, the SGs were returned to service with a tube already exceeding the Repair Criteria. A tube's failure to meet a Repair Criterion does not necessarily render the SG inoperable. Steam generator OPERABILITY is based on meeting the SG Performance Criteria since the Performance Criteria provide reasonable assurance that the SG tubing remains capable of meeting its safety function. The SG Repair Criteria define limits on SG tube degradation that allow for flaw growth between inspections and still provide assurance that the Performance Criteria will continue to be met.

If an operating plant discovers that required plugging or repair was not implemented during a previous steam generator inspection, the NRC must be notified within 8 hours. This notification is based on the requirement in 10 CFR 50.72 (b) (3) (ii) (A) [3] to notify the NRC within 8 hours of a serious degradation of a plant's principal safety barriers.

In order to determine SG OPERABILITY, an evaluation must be

completed that starts with the physical condition of the tube at the time of its last inspection and accounts for the time since the inspection, and the potential growth rate of the degradation. The OPERABILITY determination is based on the estimated condition of the tube at the time the situation is discovered. The plant's Corrective Action Program will establish the Completion Time requirement for the OPERABILITY determination.

A special report is required within 60 days of determining the failure to implement a required plugging or repair. This report is intended to be coincident with the report required by 10 CFR 50.73 (a) (2) (ii) (A) (serious degradation of the principal safety barriers) [4]. In addition to the content required by the regulation, the report must provide information on the specific Performance Criteria exceeded and the basis for the next operating period (Operational Assessment). This determination is important in order to provide reasonable assurance that the Performance Criteria are currently being met and will continue to be met at the time of the next SG inspection. The 60 day requirement is based on NEI 97-06 [5] that requires completion of a tube integrity assessment for the next operating cycle within 90 days after startup. The Action requirement for the report is more restrictive since this report is necessary to prove that the degraded SG tube(s) are currently meeting the Performance Criteria.

B. If the OPERABILITY evaluation required by Action A.2 determines that a SG tube does not meet a Performance Criteria in MODES 1 through 4, the steam generator must be declared inoperable and the actions required by the appropriate license Technical Specifications must be followed. The steam generator Performance Criteria include the design basis parameters that define acceptable steam generator performance. Meeting the Performance Criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity. Therefore, OPERABILITY of the steam generators is based on meeting the Performance Criteria.

C. During shutdown periods the steam generators will be inservice

BASES (continued)

inspected as required by the licensee's Steam Generator Program. The licensee will perform a condition monitoring assessment of the "as found" condition of the steam generator tubes. The structural and accident leakage Performance Criteria are then used to assess tube integrity and the effectiveness of the licensee's program. This assessment may be performed analytically or by test.

Note 1 in the CONDITION column states that failure to meet the Performance Criteria in MODES 5 or 6 does not render the steam generator inoperable. In MODES 5 and 6 reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for leakage. In addition, due to lower differential pressure driving forces and lower primary coolant activity, primary to secondary leakage in MODES 5 and 6 is not a significant safety concern. During these MODES the Performance Criteria are still required to be met but do not form a basis for SG OPERABILITY.

Note 2 in the Condition column states that NRC notification and reporting is required every time a Performance Criterion is not met. The intent of this note is to prevent a Technical Specification interpretation that makes the NRC report unnecessary based on the fact that any tube that fails to meet the Performance Criteria will be plugged or repaired before the report is required. Plugging or repairing the tube takes the plant out of Condition C and the standard Technical Specification interpretation does not require any subsequent actions once the Condition has been cleared. The note is not intended to require multiple reports. Only one set of reports is required for each SG inspection that identifies a tube that fails to meet a Performance Criterion.

If the Performance Criteria are not met in MODES 5 or 6, the affected tubes must be repaired by an approved Repair Method or removed from service by plugging.

Repair Methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a steam generator tube is not a repair. The Repair Methods approved for use at [Plant] are:

BASES (continued)

- [...]

The affected tubes must be repaired prior to entry into MODE 4 because the tubes must meet the Performance Criteria in MODE 4 in order for the steam generators to be OPERABLE.

The NRC must be notified of this condition within 8 hours. This notification is based on the requirement in 10 CFR 50.72 (b) (3) (ii) (A) [3] to notify the NRC within 8 hours of a serious degradation of the principal safety barriers.

In addition, a special report is required within 60 days of determining a Performance Criterion was not met. This report is intended to be coincident with the report required by 10 CFR 50.73 (a) (2) (ii) (A) (serious degradation of the principal safety barriers) [4]. In addition to the content required by the regulation, the report must provide information on the specific Performance Criteria exceeded. The 60 day time limit is required by 10 CFR 50.73 [4].

Finally, a special report is required within 120 days of determining a Performance Criterion was not met. This report must describe the basis for the next operating period (Operational Assessment). This determination is important in order to provide reasonable assurance that the Performance Criteria will be met at the time of the next SG inspection. The 120 day completion time is based on an NEI 97-06 [5] requirement that specifies completion of a tube integrity assessment for the next operating cycle within 90 days after startup.

D. During shutdown periods the steam generators will be inspected as required by the licensee's Steam Generator Program. During the inspection, the licensee will perform an integrity assessment of the steam generator tubes. The purpose of the integrity assessment is to ensure that the Performance Criteria have been met for the previous operating period (i.e., condition monitoring), and will continue to be met for the next period (i.e., operational assessment).

Ensuring that the tubes will meet the Performance Criteria at

BASES (continued)

the next inspection involves the use of Repair Criteria. The tube Repair Criteria establish limits for tube degradation that provide reasonable assurance that an affected tube will meet the Performance Criteria at the next scheduled inspection by allowing for anticipated growth during the intervening time interval. If a plant discovers that a tube exceeds the Repair Criteria, the tube must be plugged or repaired before entry into an operating MODE.

Due to technique and analyst uncertainties, sampling plans, and probability of detection there is a possibility that tube(s) exceeding the Repair Criteria will not be detected during a particular steam generator inspection. If the flaw(s) is detected during a subsequent inspection, the condition is not considered a violation of the LCO or a reportable event unless it is determined that the Performance Criteria are not met.

A note in the CONDITION column states that failure to meet the Repair Criteria in MODES 5 or 6 does not render the steam generator inoperable. Because of its allowance for growth, exceeding a tube repair criterion during a steam generator inspection does not render the steam generators inoperable. OPERABILITY is based on meeting the Performance Criteria.

SURVEILLANCE REQUIREMENTS

A. The existence of the Steam Generator Program is required by the license Technical Specifications. The Technical Specifications also place controls on changes to the Performance Criteria. The steam generator Performance Criteria identify the standards against which performance is to be measured. Meeting the Performance Criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity.

NEI 97-06, Steam Generator Program Guidelines [5], and its referenced EPRI Guidelines establish the content of the Steam Generator Program. The Steam Generator Program incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures. The Steam Generator Program requires periodic steam generator tube integrity assessments. The purpose of the integrity assessment

BASES (continued)

is to ensure that the Performance Criteria have been met for the previous operating period (i.e., condition monitoring), and will continue to be met for the next period (i.e., operational assessment). The condition monitoring assessment is an evaluation of the "as found" condition of the tubing with respect to the Performance Criteria for structural and accident leakage integrity. The operational assessment determines the length of the operating cycle by providing reasonable assurance that the tubing will meet the Performance Criteria at the next scheduled inspection. The operational assessment therefore establishes the surveillance frequency.

Verifying steam generator tube integrity in accordance with the Steam Generator Program confirms the OPERABILITY of the steam generators.

B. Steam generator tubes that exceed the Repair Criteria must be repaired by an approved Repair Method or removed from service by plugging prior to entry into MODE 4 after the SG inspection that detected the degradation. This is necessary in order to provide reasonable assurance that tube integrity will be maintained until the next scheduled inspection.

A note included with the verification requirement states that the surveillance need not be met in MODES 5 or 6. This note allows entry into MODES 5 or 6 prior to satisfaction of the surveillance since the surveillance can only be performed during these MODES.

The frequency requirement for this surveillance is contained in the Steam Generator Program. The Steam Generator Program requires periodic steam generator tube integrity assessments including an operational assessment that determines the length of the operating cycle by providing reasonable assurance that the tubing will meet the Performance Criteria at the next scheduled inspection. The operational assessment therefore establishes the surveillance frequency.

REFERENCES

1. 10 CFR 50 Appendix A, GDC 19, *Control Room*

2. 10 CFR 100, *Reactor Site Criteria*

BASES (continued)

3. 10 CFR 50.72, *Immediate notification requirements for operating nuclear power reactors*

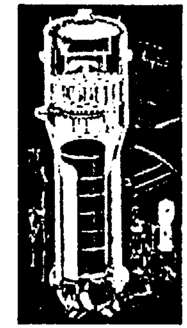
4. 10 CFR 50.73, *Licensee event report system*

5. NEI 97-06, *Steam Generator Program Guidelines*

6. EPRI Report TR-107621, *Steam Generator Integrity Assessment Guidelines*

7. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, *Rules for Construction of Nuclear Facility Components, Class 1 Components*

8. Draft Regulatory Guide 1.121, *Basis for Plugging Degraded Steam Generator Tubes*, August 1976



SG Program Generic License Change Package Status

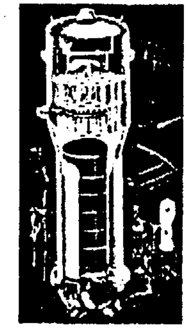
SGTF / NRC Meeting

July 26, 2000

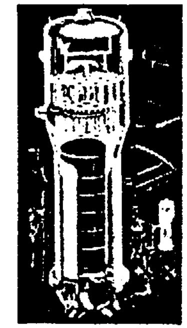
White Flint

Attachment 4

NDE Issues Related to IP-2 Experience

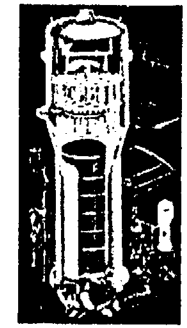


- Data Quality
- S/N Ratio & Characteristics
- Technique Qualifications at U-Bend
- Use of High Frequency Probe
- Dissemination of Information



Data Quality

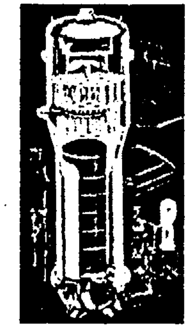
- Data quality is being addressed in Revision 6 of the ISI Guidelines (work in progress)
- Recent EPRI R&D results are being used as the starting point for development of recommendations/requirements in Revision 6
  - Draft data quality reports for bobbin and Plus Point probes have been developed
- Initial Rev 6 guidance based on EPRI R&D was reviewed by vendors who ultimately have to implement them in their acquisition software



Data Quality

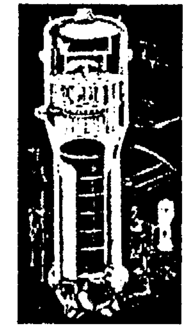
- Objectives of Rev 6
  - Develop generic guidance which applies to all EC probes
  - Develop specific guidance for the bobbin, plus point, pancake, and array probes
- The guidance will contain:
  - Quality parameters
  - Acceptance criteria
  - Frequency of testing
  - Location of test

EPRI



S/N Ratio & Characteristics

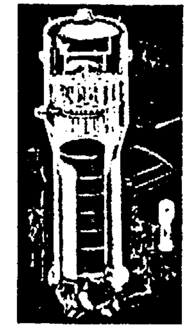
- Statement has been made to the effect that the S/N issue at IP-2 is not unlike other plants
- A comparative study of noise at 3 relevant plants was presented at the SG NDE workshop
- Scott Redner will present the results



Technique Qualifications at U-Bend

- Current appendix H qualifications at u-bend are based on machined flaws in u-bend samples
- Contract work is under way to produce stress corrosion cracks in low row u-bends
- The challenge has been to find low row u-bend tube samples of 600 MA material
- Tube samples have now been located to produce ODSCC and PWSCC at susceptible locations (apex and tangent) of low row u-bends

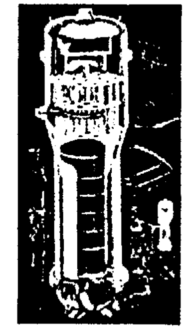
EPRI



Use of High Frequency Probe

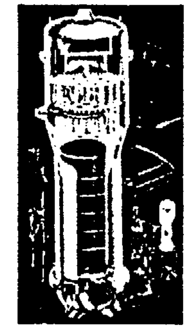
- NRC had inquired about the industry's position on recommending the use of high frequency probes
  - EPRI does not make recommendations on probe usage for technical and commercial reasons
    - Degradation specific applications
    - Other probes may have similar capabilities
    - Could discourage development of new probes

EPRI



Use of High Frequency Probe

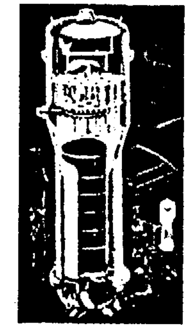
- Probes are chosen by following the program required by NEI 97-06
  - Degradation assessment
  - Inspection
    - Technique qualification -EPRI qualification of high frequency probes is complete
    - Site validation
  - Tube integrity assessment - CMOA



Dissemination of Information

- Update of IP2 issues at TAG meeting In June
- NDE issues related to IP2 were discussed at the SG NDE Workshop in presentations on U-bend experience at:
  - Palo Verde / APS
  - IP2 / ConEd
  - Kewaunee / WPS
  - Prairie Island / NSP
  - NRC Perspective

EPRI



Dissemination of Information

- Lessons learned from presentations
  - U bend is a difficult inspection
  - Enhance training/SSPD
  - Data Quality
  - ODSCC considerations
  - Evaluate noise
- Industry considering issuing a lessons learned letter on IP2 experience
- Rev 6 will strengthen site validation requirements which includes noise considerations

U-BEND NOISE STUDY

NEI SGTF / NRC Meeting
July 26, 2000
Washington, DC
Scott A. Redner
Northern States Power Co.

Attachment 5

OUTLINE

- BACK GROUND
- NOISE STUDY
- RECOMMENDATIONS

BACK GROUND

- Pre IP#2 large leakage event most plants used a Mid Range +Pt. (PP11) for low row u-bend examinations
- Post IP#2 two plants similar in design and vintage to IP#2 elected to use the IP#2 High Frequency +Pt. (aka HFPP9 or the Caius V. Dodd probe)
- Prior to the inspection no Site Validation was performed for the HF +Pt.
  - not possible due to no site data

BACK GROUND

- EPRI NDE/ISI IRG requested a noise study for the most at risk plants
  - I600 low temp. mill annealed tubing
  - No in-situ heat treatment performed
  - Highest EFPY on original SG's
  - Ran HF +Pt.

NOISE STUDY

- Review of the high frequency and mid range EPRI Appendix H data sets to determine the average tubing noise values

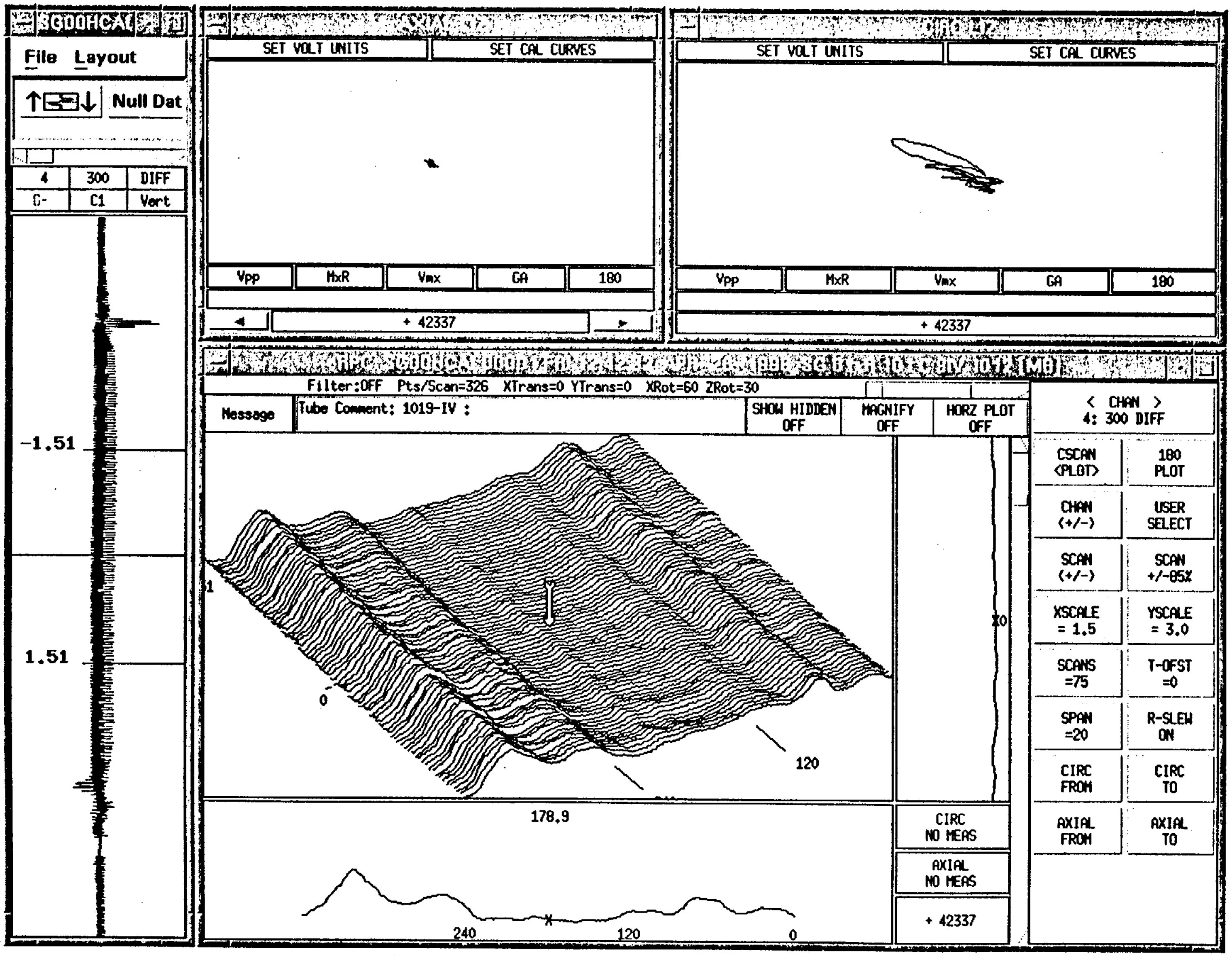
VERSUS

- Review of a sample of row 2 tubes in 3 of the 4 most at risk plants to determine the average tubing noise values (Plants I, K & P)

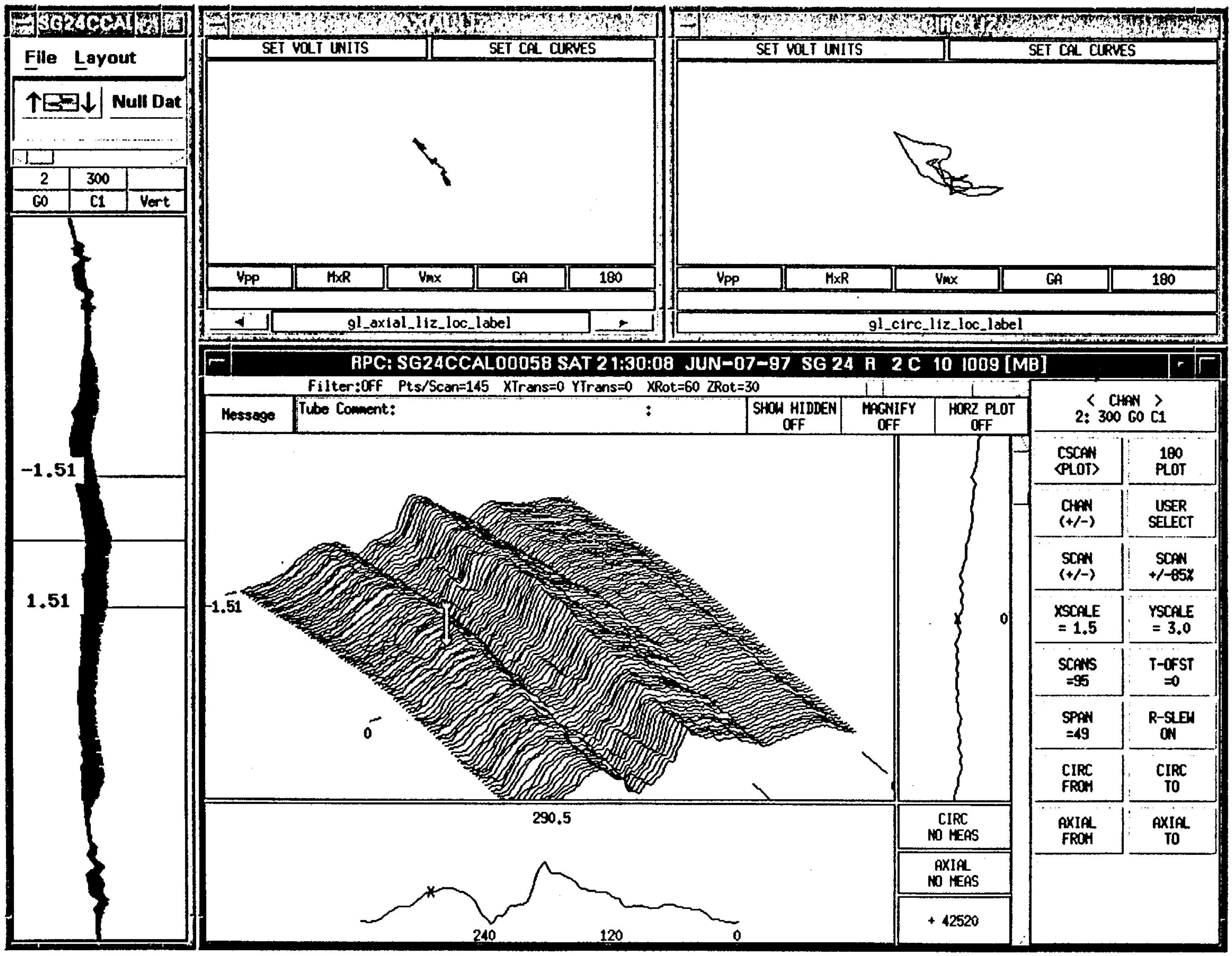
NOISE STUDY

- Procedure
  - Normalize voltage per Revision 5 of the GL's
  - Use qualification frequency of both +Pt. Coils
    - 300 kHz for MR and 800 kHz for HF
  - record both the Vpp and Vmx at the apex of the u-bend on the qualification data set and a sample of approximately 20 row 2 tubes from each plant
- The following are the noisiest results from the 2 qualifications and 3 plants with normalized spans and rotations

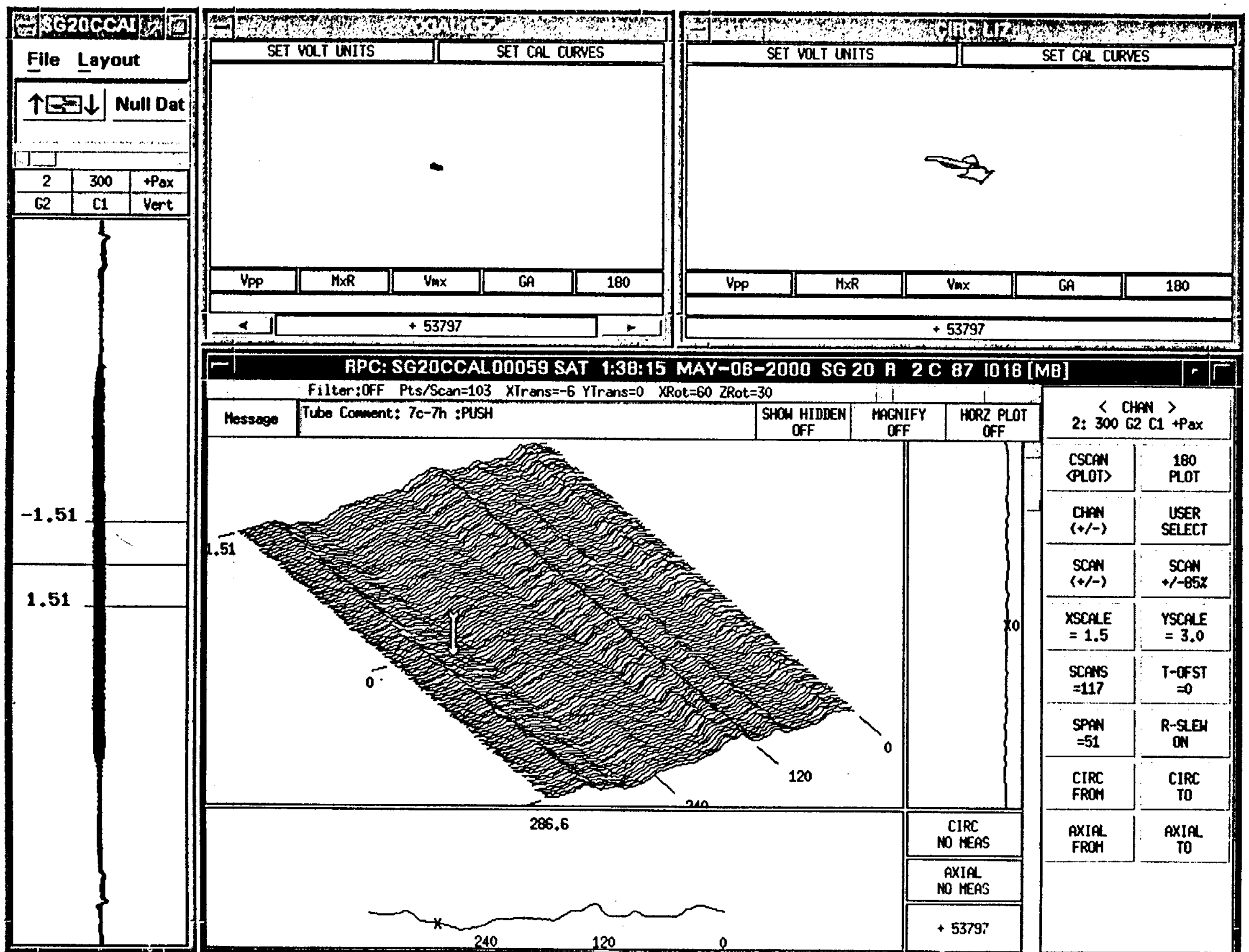
300 kHz MR +Pt. QUAL.



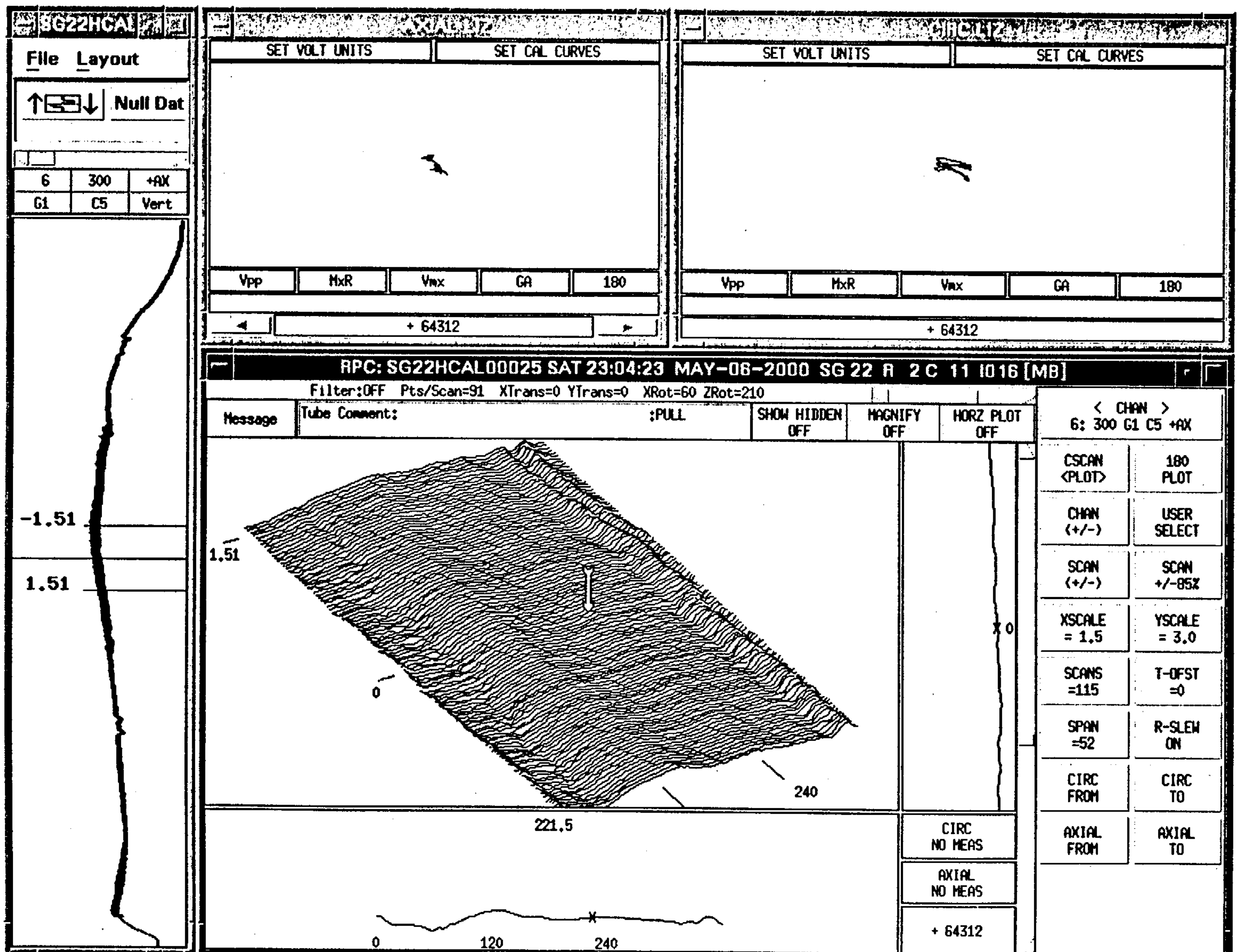
300 kHz MR +Pt. PLANT I



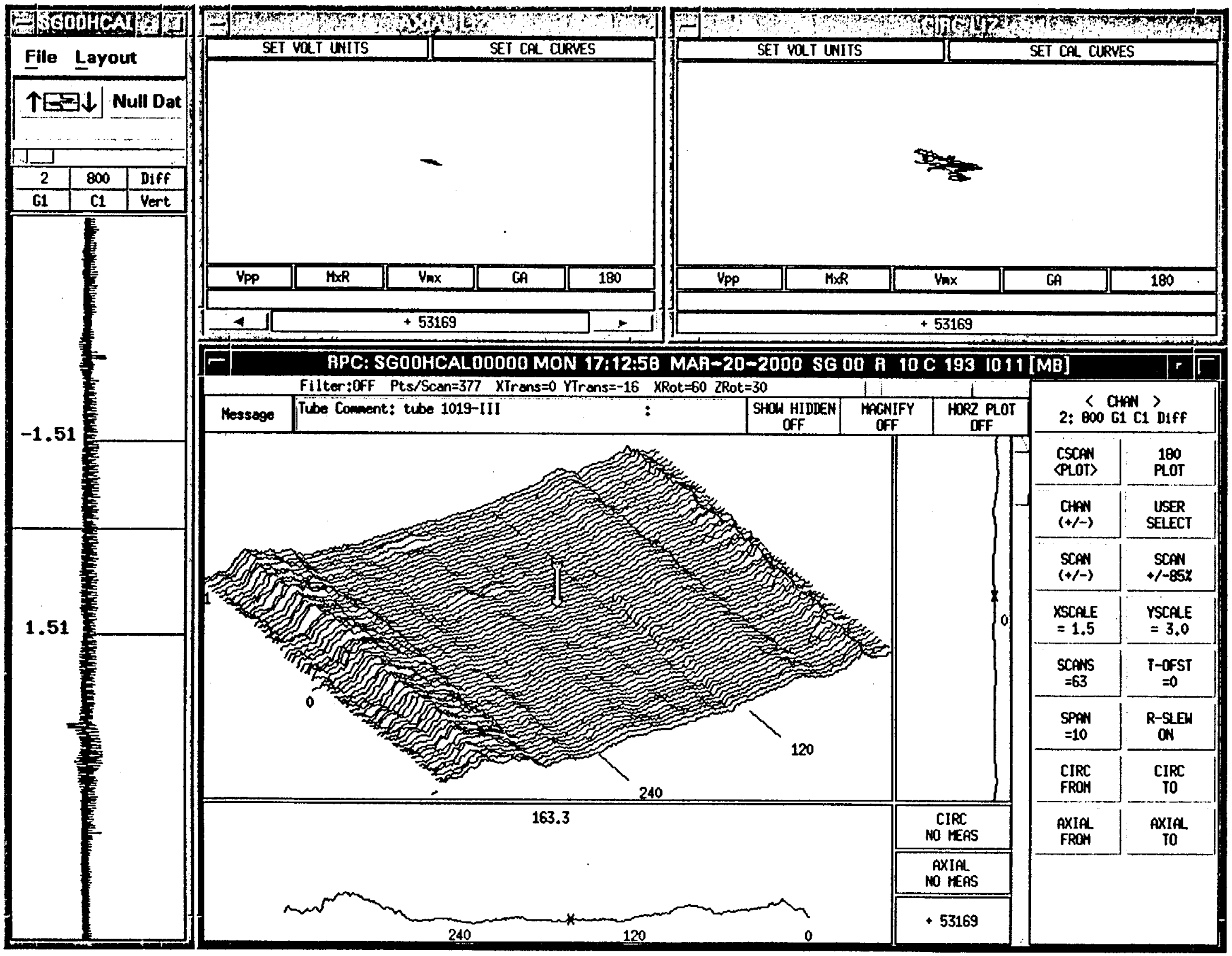
300 kHz MR +Pt. PLANT K



300 kHz MR +Pt. PLANT P



800 kHz HF +Pt. QUAL.



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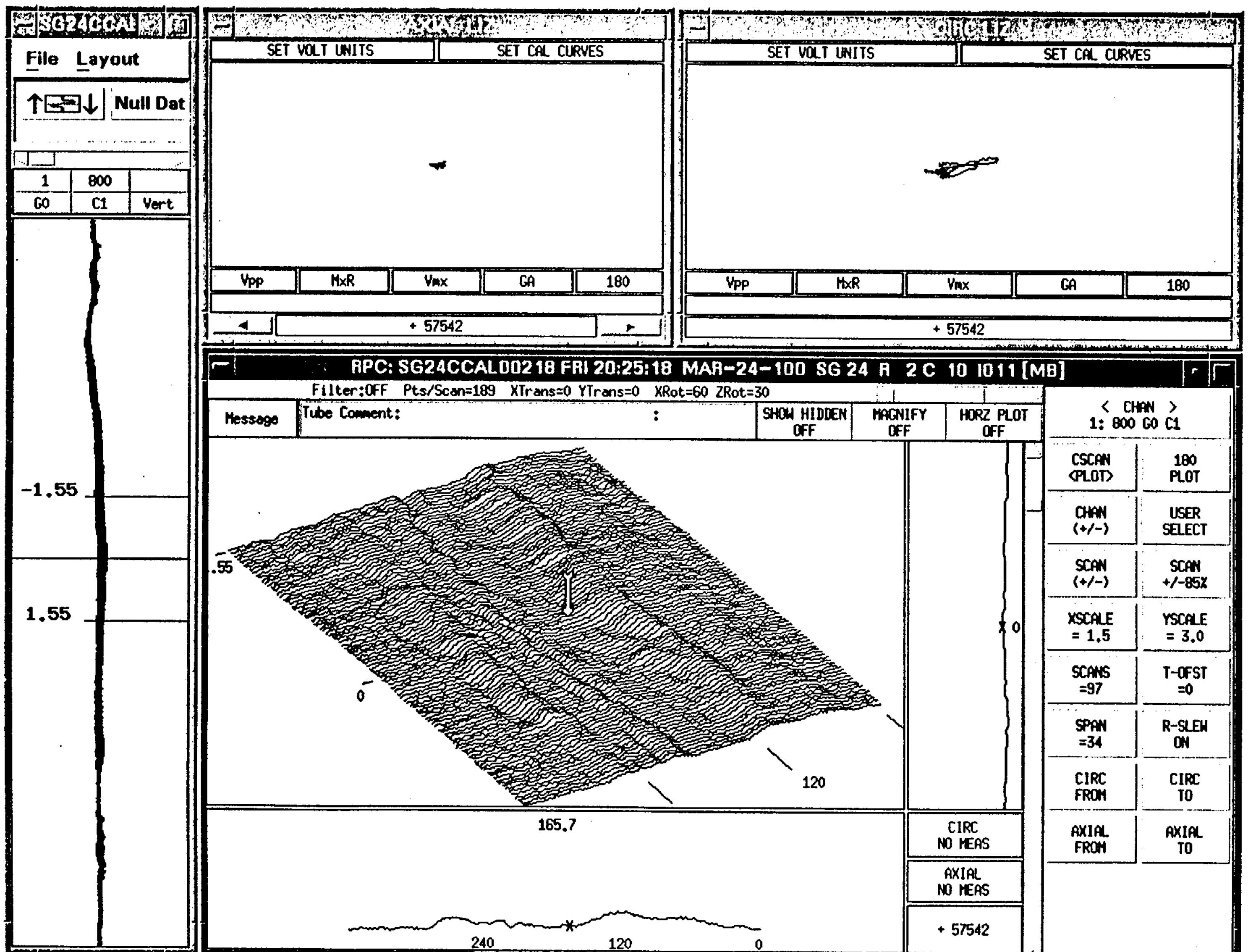
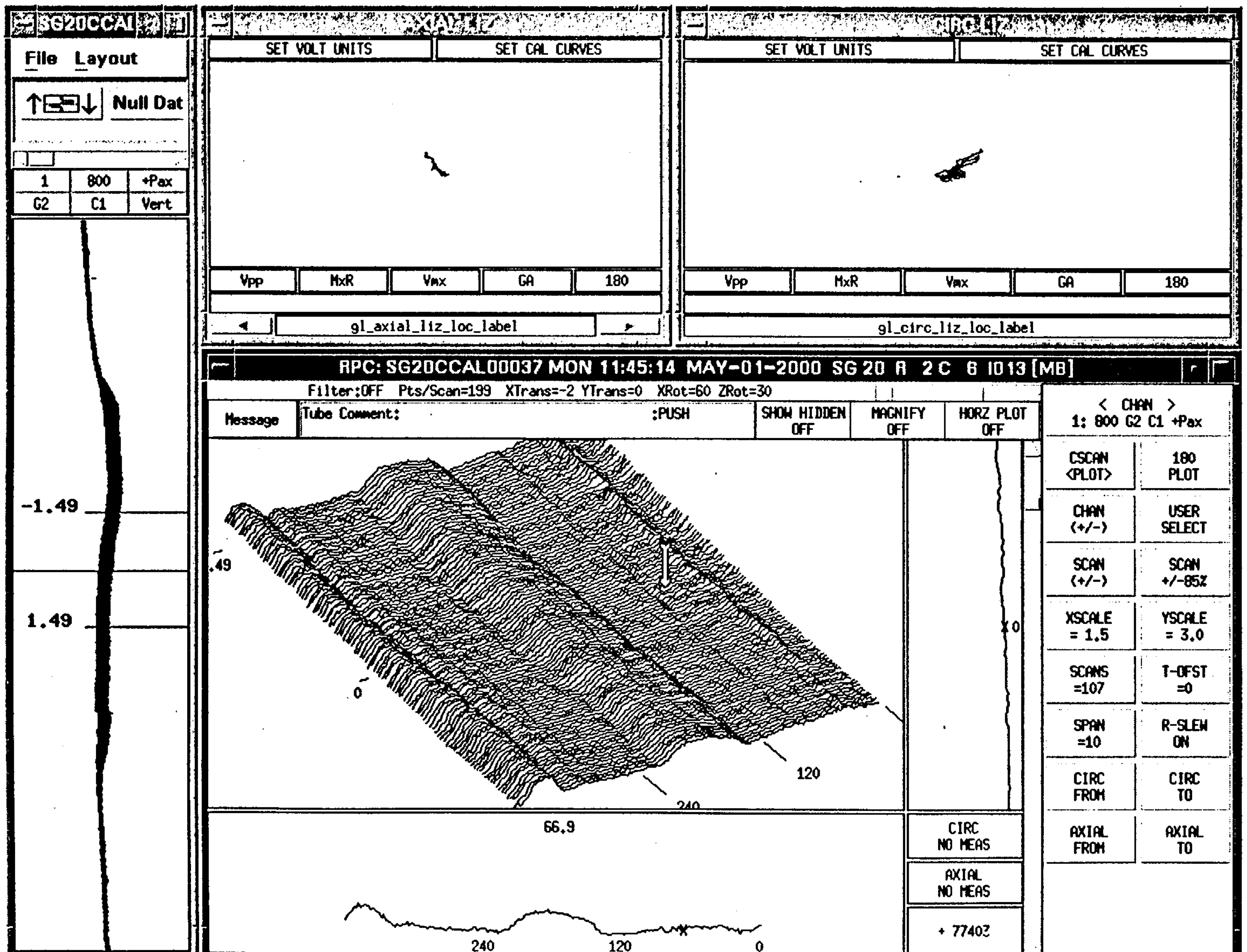
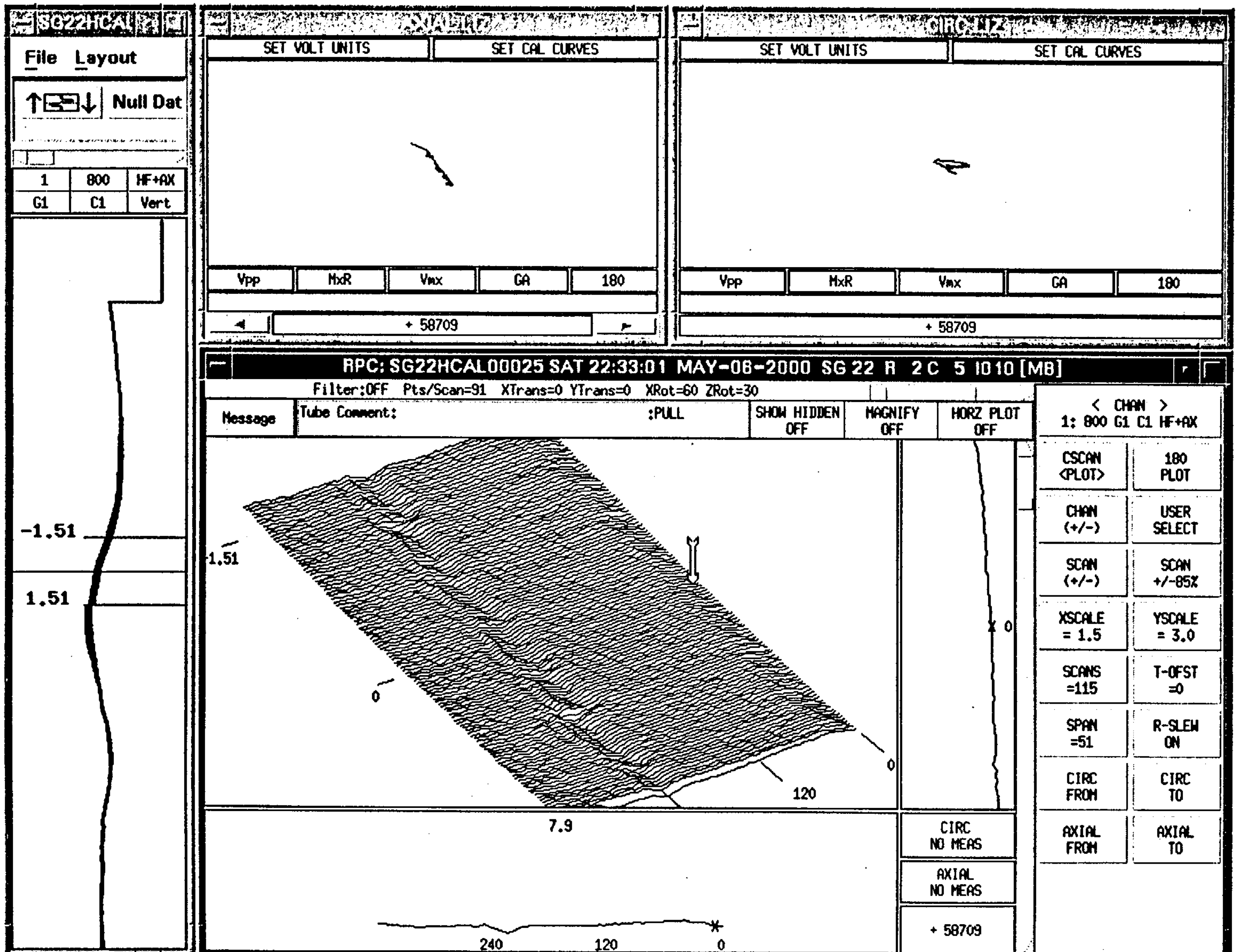
800 kHz HF +Pt. PLANT I



800 kHz HF +Pt. PLANT K



800 kHz HF +Pt. PLANT P



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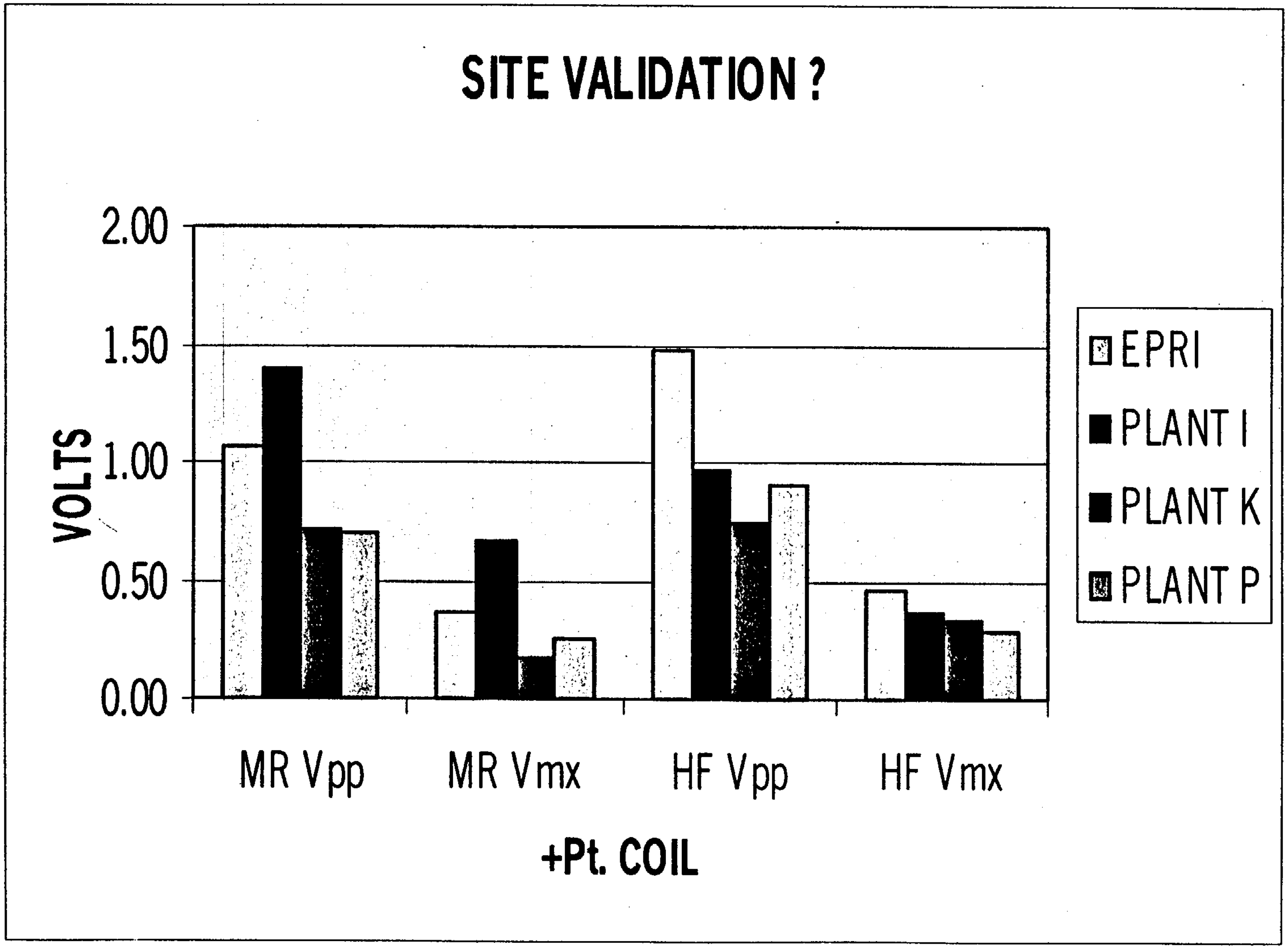
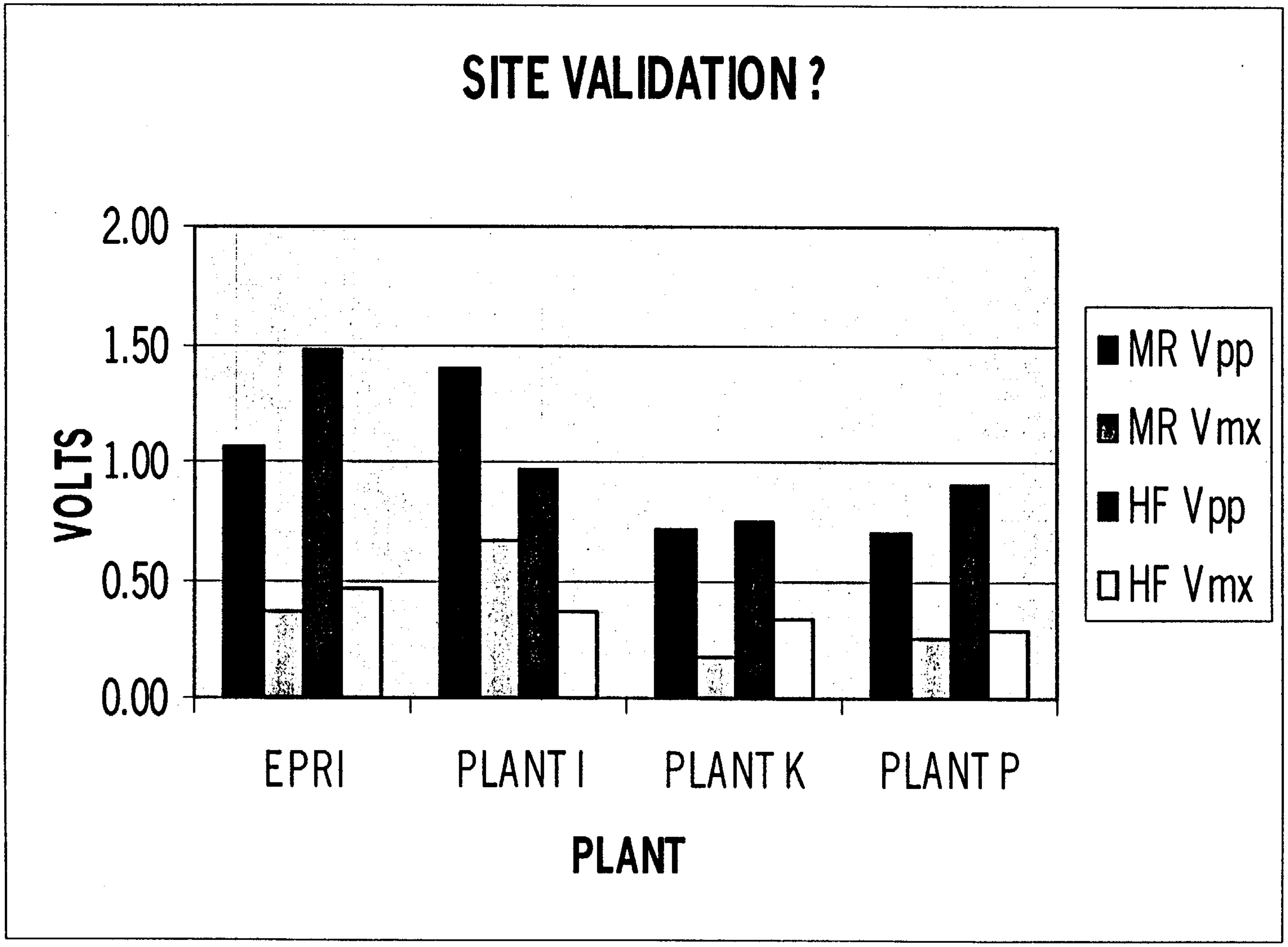
NOISE STUDY

(≅ 20 tube average)



NOISE STUDY

(≅ 20 tube average)



NOISE STUDY

- CONCLUSION
  - The MR +Pt. is not site validated for plant I
    - More noise than EPRI qualification
  - The MR +Pt. is site validated for plants K & P
    - Less noise than EPRI qualification
  - The HF +Pt. is site validated for plants I, K & P
    - Less noise than EPRI qualification

RECOMMENDATIONS

- HIGH FREQUENCY +Pt.
- SHOULD EVERY PLANT USE IT? NO, BUT
  - If the noise in your site data exceeds the noise in the MR qualification - YES
- IF YOU CHOOSE TO USE IT OR NEED TO USE IT - CAN IT REPLACE THE MR +Pt.? NO
  - The HF +Pt. Does not detect OD EDM notches below approximately 60% TW

SG Tube Burst Testing Status

Mati Merilo for R. F. Keating
Westinghouse NSBU
Madison, PA

Attachment 6

Status

- Program described to staff on 7/6/2000
  - Burst pressure may be dependent on pressure rate for certain morphologies
  - Goals are to assess
    - ANO test data
    - Potential effects on industry test procedures
    - Potential effects on industry evaluation models
- Target completion by end of August
  - Currently on schedule

Status (Cont.)

- Project status
  - Survey of industry experience initiated
  - Metallurgical work in progress
    - Flaw profiling
    - Tensile test
- Staff comments from the 7/6/00 presentation