DMB

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

5N 157B Lookout Place

86 SEP 19 P1:34

SEP 17 1986

U.S. Nuclear Regulatory Commission
Region II
ATTN: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Dear Dr. Grace:

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2 - NRC-OIE REGION II INSPECTION REPORT 50-327/86-31 AND 50-328/86-31 - RESPONSE TO VIOLATIONS

Enclosed is our response to Gary G. Zech's August 12, 1986 letter to S. A. White which transmitted Notice of Violation Nos. 50-327/86-31 and 50-328/86-31 for our Sequoyah Nuclear Plant. Enclosure 1 is our response to the subject violations. We do not recognize any other actions described herein or the subject inspection report as commitments.

If you have any questions, please get in touch with G. B. Kirk at 615/870-6549.

To the best of my knowledge, I declare the statements contained herein are complete and true.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R Gridley

R. Gridley, Director
Nuclear Safety and Licensing

Enclosure
cc (Enclosure):
Mr. James Taylor, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. G. G. Zech, Director, TVA Projects
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

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Q PDR

1/1 IE01

ENCLOSURE

RESPONSE - NRC-OIE INSPECTION REPORT
NOS. 50-327/86-31 AND 50-328/86-31
GARY G. ZECH'S LETTER TO S. A. WHITE
DATED AUGUST 12, 1986

VIOLATION 50-327/86-31-01 AND 50-328/86-31-01

Technical Specification 6.12 states that in lieu of the control device or alarm signal required by paragraph 20.203 (c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special (Radiation) Work Permit (SWP).

Contrary to the above, on April 30, 1986, the licensee failed to barricade and conspicuously post a high radiation area. Specifically, an entrance to a high radiation area inside the polar crane wall in lower containment was not barricaded and there was no conspicuous posting of the area. A high radiation sign had been placed on the door to the area, however, the door had been opened to the point where the sign was not visible until after entry into the area.

This is a Severity Level IV violation (Supplement IV).

1. Admission or Denial Of The Alleged Violation

   TVA admits the violation occurred as stated.

2. Reason For The Violation

   The violation occurred because of personnel error. The door at the entrance to the area inside the polar crane wall was posted with a high radiation area sign. However, if this door is inadvertently left open then the "barricade" and "conspicuous posting" is compromised. This was the case on April 30, 1986 when the violation was identified.

3. Corrective Steps Taken and Results Achieved

   A swing gate was placed immediately behind the door and posted as a high radiation area.

4. Corrective Steps Taken to Avoid Future Violations

   Swing gates are utilized at those entrances to high radiation areas which are not equipped with a door or otherwise cannot be properly posted and barricaded. On occasion these gates may not return to a closed position. Under these circumstances the physical barrier may be interpreted "technically" as not being present. However the swing gate is still visible to workers entering the area and are therefore aware of the fact that they are entering a high radiation area. Consequently these swing gates fulfill their intended function even when they are inadvertently left in an open position.

5. Date When Full Compliance Will Be Achieved

The plant was in full compliance on April 30, 1986 when the swing gate was installed behind the subject door.

Violation 50-327/86-31-02 and 50-328/86-31-02

10 CFR 50, Appendix B, Section V requires that activities affecting quality be prescribed and that the applicable instructions, procedures, or drawings include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, the configuration of trip contacts 17/18 and 21/22 on alternate breaker 5B, located on shutdown board 1A1-A, was not addressed in the acceptance criteria of step 2.4.5 of Work Plan 11871. As a result, these were not adequately controlled. Mispositioned contacts 17/18 caused in an interlocked trip of the normal breaker 1B, resulting in a loss of power to radiation monitor RM-90-101 and an Auxiliary Building Isolation (ABI).

This is a Severity Level IV violation (Supplement I).

1. Admission or Denial Of The Alleged Violation

TVA admits the violation occurred as stated.

2. Reason For The Violation

The violation occurred because of personnel error in that the shaft for the trip contacts was reinserted, following a modification, with a contact rotated incorrectly. The workplan did not instruct the personnel doing the work to verify any contact positions other than the ones being modified. In addition, the cognizant engineer did not expect to disturb contacts 17/18 and 21/22 while performing the modification.

3. Corrective Steps Taken and Results Achieved

A test deficiency sheet was completed to correct the mispositioned contact. The contact was corrected and the alternate breaker was verified to work properly. The workplan was changed to check the other contacts before reinstalling the shaft and contact assembly. This event was reported in Licensee Event Report (LER) 327/86-019 dated May 28, 1986.

4. Corrective Steps Taken To Avoid Future Violations

The functional test requirements of SQN Administrative Instruction (AI) -19, Part IV, "Plant Modifications: After Licensing" have been stressed to the cognizant engineers.

5. Date When Full Compliance Will Be Achieved

The plant was in full compliance on August 31, 1986 when the requirements of AI-19 were stressed to the cognizant engineers.

Violation 50-327/86-31-03 and 50-328/86-31-03

Section XVI, "Corrective Action" of 10 CFR 50, Appendix B requires that significant conditions adverse to quality, such as deficiencies, deviations, defective material and equipment be promptly identified and corrective action taken be documented and reported to appropriate levels of management.

Contrary to the above, the licensee failed to take appropriate corrective action for test deficiencies on preoperational tests. Deficiency DN-1 for Unit 1 preoperational test procedure W-11.7, Revision 0, Calibration of Steam and Feedwater Flow Instruments at Power, involved flow instrumentation which did not meet the test acceptance criteria at 75% and 100% thermal power. The test portions required to be repeated by the Office of Engineering interim approval of the deficiency were never performed. Subsequently, the Unit 1 deficiency was not adequately addressed by the licensee in the followup deficiency resolution. In addition, the test deficiencies and exception noted in the performance of W-11.7 on Unit 2 were inadequately addressed in their resolution. Unit 2 Deficiency DN-1 consisted of the licensee's inability to obtain adequate zero power feedwater flow indications due to the failure to backfill the special test feedwater flow detectors. Unit 2 deficiency DN-2 was written to document the licensee's failure to meet the required test acceptance criteria for calibrating the feedwater and steam flow process instrumentation at 75% and 100% thermal power. The licensee also took exception to performing the calibration adjustments necessary to bring the flow instrumentation within specifications and to performing the calibration repeatability check specified in the W-11.7 test acceptance criteria.

This is a Severity Level V violation (Supplement 1).

1. Admission Or Denial Of The Alleged Violation

TVA admits the violation occurred as stated.

2. Reason For The Violation

The violation occurred because of personnel error in that the preoperational test section did not assume the lead role in troubleshooting the main steam and feedwater flow instrumentation. The preoperational test engineer failed to adequately communicate the status of the preoperation test to plant management. Consequently, the test procedure did not receive the proper attention and appropriate corrective action was not fully documented in this procedure.

3. Corrective Steps Taken and Results Achieved

A review of the test results for the preoperational test on the main steam and feedwater flow instrumentation was conducted by the preoperational test section and based upon this review the test deficiencies were resolved. However, TVA agrees with the inspection team in that the deficiency resolutions were poorly worded and gave the impression of being inadequate to correct the identified deficiencies. As a result of this, TVA performed an indepth review of all available documentation on the main steam and feedwater flow instrumentation. This review did not identify any changes that needed to be made to the original deficiency resolutions. The findings of this review are documented in the attachment to this response.

4. Corrective Steps Taken To Avoid Future Violations

TVA has placed more emphasis on individual responsibility and accountability for actions taken to ensure problems and deficiencies are properly resolved. In addition, test deficiency resolutions in surveillance, post modification and special tests require review by the responsible section supervisor to ensure adequacy of the resolution.

5. Date When Full Compliance Will Be Achieved

The plant was in full compliance on June 6, 1986 when the review of the test deficiencies was completed.

RII-86 06 10 4469 S53 860606 875

Memorandum

ATTACHMENT

TENNESSEE VALLEY AUTHORITY

(11)

TO : L. M. Nobles, Operations/Technical Services Superintendent, ONP, POB-2, Sequoyah Nuclear Plant

FROM : R. W. Fortenberry, Head, Engineering Group, ONP, O&PS-4, Sequoyah Nuclear Plant

DATE : June 6, 1986

SUBJECT: SEQUOYAH NUCLEAR PLANT - POTENTIAL NRC ISSUES REGARDING PREOPERATIONAL TEST W-11.7, CALIBRATION OF STEAM AND FEEDWATER FLOW INSTRUMENTATION AT POWER, UNIT 1 AND UNIT 2

After the NRC exit meeting held in your office at 3 p.m. on May 12, 1986, Rick Mooney, Bruce Wilson, and I met with Scott McNeil of NRC's I&E staff to clearly define potential issues that Mr. McNeil is still pursuing in regard to W-11.7. That meeting resulted in the following ten questions.

1. Knowing that the 75-percent - 100-percent data did not meet acceptance criteria, prove that SQN did not exceed 100-percent reactor power. This issue involved unit 2 only.

   I provided a detailed explanation of startup test SU-8.5.1, and Mr. McNeil and I reached a point of understanding such that he considered this item closed.

2. The Rosemonts were not calibrated prior to zero (0) percent data being taken. This issue involved unit 2 only.

   Substantial documentation was provided to Mr. McNeil previously to show that the Rosemonts, as well as permanent plant process transmitters associated with W-11.7, unit 2, were in fact calibrated. Some doubt may have existed as to the identification of the Rosemonts since the TVA tag number or the manufacturer's serial number were used on various documents. Attached you will find this documented cross reference and calibration records.

3. In Step 2.2.2, Power Production verified that the prerequisite was complete. Determine how this is acceptable. This issue involved unit 2 only.

   The actual prerequisite in W-11.7 was to verify that prerequisites 2.2, 2.3, and 2.4 in SU-8.5.1 (Rev. 0) were complete. Those prerequisites in SU-8.5.1 (Rev. 0) were a verification that the test instruments were installed. W-11.7 should have more clearly stated the instrument installation was to be verified. The subject prerequisites were deleted from SU-8.5.1 (Rev. 1), but could have been signed off at time of conduct of W-11.7 if they had been retained. The intent of prerequisite 2.2.2--to verify the test instruments were installed--was in fact verified and data was recorded from these instruments, proving they were actually installed. We agree the wording of the prerequisite could have been clarified, but the intent of the prerequisite was met.

4. At 0-percent Power (W-11.7-DN-1, Unit 2):

   A. Provide verification that the instruments were calibrated and water leg established.

      As stated in No. 2 above, substantial documentation has been provided to Mr. McNeil regarding both process and test instrument calibration. The calibration issue is closed. Regarding establishment of water leg, documentation exists proving main steam sense lines were backfilled within 24-hours prior to collection of the 0-percent power data. Firm documentation to show backfilling of the feedwater sense lines does not exist prior to December 21, 1981, which reinforces the belief previously stated to Mr. McNeil that the real problem identified by DN-1 (W-11.7 Unit 2) was air entrapment in the sense lines. There was never a requirement to verify the establishment of the water leg in this procedure, but good engineering practice was normally followed by performing this function. If the instrument lines for a flow instrument were not filled because of inattention or inadvertent loss of water column as can occur at any time, the instrument will show obvious indications such as "bouncing" flow indications or "out of range" readings. These are readily observed by test engineers and are corrected accordingly.

   B. Provide explanation as to why the 0-percent power data was not repeated (prior to power escalation).

      No explanation was given in the procedure for not repeating the 0-percent power data prior to power escalation. However, the 0-percent power data was not tied to any acceptance criteria and documentation exists to prove that the feedwater sense lines were backfilled on numerous occasions after the 0-percent power data of W-11.7 was collected. The subsequent backfilling ensured that the air entrapment problem identified by DN-1 was corrected prior to the time that it could have adversely affected the safety functions of the feedwater flow instrumentation.

      The purpose of the 0-percent power data was to ensure instrumentation operability prior to collection of 30, 50, 75, and 100-percent power data. It fulfilled that purpose by identifying deficient conditions that were corrected and documented in SI-667 on December 21, 1981. This provided confidence that the feedwater instruments were working properly and there was no need to repeat the verification at 0-percent power that the instruments were operable in this procedure. Westinghouse supports this position in the form of a letter from F. L. Langford, W, Pittsburgh, to R. U. Mathieson, W site representative, SQN, dated May 21, 1986.

5. The 75 and 100-percent power data never met acceptance criteria. Interim approval for PT-708 (W-11.7 Unit 1) said to reperform to obtain data. Need justification for failure to reperform as specified in the unit 1 test interim approval memorandum. This same question applies to unit 2 also.

   Feedwater flow data at 75-percent and 100-percent power was collected during initial startup test SU-8.5.1 (RTI-8 has been used after each refueling outage to gather the same data). These tests were used in determining the secondary side calorimetric to determine reactor power per TI-2. The test transmitters (Rosemont) and the process flow transmitters (Foxboro) were used to collect this data. A special flow gauge (Ruska) was used at one time to ensure accuracy.

   All of the gauges read off of the same flow element. Also, steam flow versus feedwater flow data has been collected by an ongoing Instrument Maintenance investigation since each unit startup to ensure that the scaling and setpoints are correct for the steam flow instrumentation. These steam flows have been collected periodically during the life of the plant, sometimes as frequently as once per day. Deviations have been found between indicated steam flow and feedwater flow at various times during the life of the plant. The calorimetric data, steam flow, NIS, impulse pressure, and temperatures were compared and corrections made to maintain reactor power as close to, but not over, 100-percent power, as well as ensure compliance with tech spec limits on safeguard instrumentation. Other utilities have had the same problems with accurate flow measurement, as this is generally caused by flow element fouling. Correcting and adjusting feedwater and steam flow instruments to maintain correct and accurate power levels is common within the industry. Even though W-11.7 was not reperformed, subsequent test data and data collected for investigation and calibration purposes has repeated the intended purpose of W-11.7 many times over for both units. Indicated readings were always carefully adjusted to be as close as possible to actual flow rates.

   The acceptance criteria for the feedwater flow comparison in W-11.7 was ± 1-percent. Although feedwater flow data was not recorded for direct comparison during SU-8.5.1, two separate calorimetric power determinations (one based on process feedwater flow instruments, one on test feedwater flow instruments) agree within one percent of full reactor thermal power.

   The steam flow data collected for the Instrument Maintenance investigation was compared with process feedwater flow data to ensure accurate steam flow transmitter scaling. The steam flow transmitters have been rescaled as needed based on that data.

Steam flow data collected since the first cycle has been evaluated with respect to W-11.7 steam flow acceptance criteria of ± 3-percent for both units. Only three isolated data points (one for unit 1 and two for unit 2 of 1,012 total) were outside the 3-percent acceptance criteria of W-11.7 in the nonconservative (negative) direction. Those few points appear to be bad data because followup data returns to the established trend of previous data. The majority of the steam flow data was found to be positive and outside the 3-percent acceptance criteria, but this is a conservative direction compared to the safety margin (steam flow indication slightly above actual flow). However, the indicated steam flow was maintained sufficiently low to prevent a spurious safety injection.

The data collection discussed above satisfied the test objective sufficiently to ensure that the margin of safety for both units could be maintained.

6. In Section 5.3, the step was left unsigned and EX-1 was written but not resolved; gain adjustments were not made in this test. This applies to unit 2 only.

   It is true that gain adjustments were not made in W-11.7 Unit 2, but EX-1 does have a resolution. The steam flow transmitters were replaced during the unit 2 cycle 1 refueling outage and rescaled based upon data collected during the first operating cycle (reference reply to Item 5).

   On both units 1 and 2, the pressure drop across the main steam primary flow element has behaved erratically on a day-to-day basis, but in the long term the ΔP has trended upward. This behavior has resulted in the continuous investigation by the Instrument Maintenance Section to study steam flow ΔP behavior. Steam flow transmitter gain adjustments were made so that the steam flow instrumentation was maintained within tech spec limits. A similar situation was discovered with the main feedwater flow ΔP but to a lesser degree than that for the steam. That discovery resulted in development of a new calorimetric power determination (based upon primary system parameters) by the Nuclear Engineering Section working in conjunction with Westinghouse, and addition of a correction factor to the P250 calorimetric program (U1118).

   The efforts described above demonstrate that no safety issue was generated due to TVA's failure to properly document resolution of EX-1 in a timely manner.

7. Justify why verfication of repeatability was not done (Step 5.6). This involves unit 2 only.

   Previously described data collected by the Instrument Maintenance investigation and Nuclear Engineering Section calorimetrics fully met the objective of W-11.7 to verify instrument repeatabilty. These programs were able to detect the ΔP drift for both steam and feedwater flow and ensure that safety limits were not violated.

8. Justify why unit 1 test was not repeated (PT-708).

9. Interim approval (first cycle) for unit 1 was based on repeating test. Justify why test was not repeated.

   Issues No. 8 and 9 are both restatements of issue No. 5. It is true that TVA did not repeat the test as specifically stated in the interim approval memorandum for PT-708, but, as explained in the response to issue No. 5, other activities accomplished the intent of the retest and no compromise in nuclear safety was experienced.

10. Disposition of unit 1 (DN-1) was used to close out unit 2 deficiencies and exception, while unit 1 (DN-1) was a different problem. Explain and justify the general approach used to close this out.

    Disposition of DN-1, W-11.7 Unit 1, (rework of steam flow transmitter and retest) was not the sole basis for closure of the unit 2 deficiencies and exception. The final disposition of W-11.7 Unit 2 (Form 4) made clear reference to PT-708 and is explained by the following statements:

    > Both PT-708 (W-11.7 Unit 1) and DN-2 (W-11.7 Unit 2) identify the inability to obtain data that will meet the specified acceptance criteria for 75 and 100-percent power. This problem is by far the most significant problem in both tests.
    >
    > The interim approval memorandum for PT-708 (W-11.7 Unit 1) allowed credit to be taken for the efforts of Instrument Maintenance and defer rerunning of the test (for unacceptable data, PT-708, and rework of steam flow transmitters, DN-1) until startup following the first refueling outage. The writing of EX-1 (W-11.7 Unit 2), which referenced replacement of steam flow transmitters and retest, and submittal of the test data package to EN DES for review is a direct result of the similarities between PT-708 (W-11.7 Unit 1), including the interim approval, and DN-2 (W-11.7 Unit 2). The test director realized that the major problem of unacceptable data at 75 and 100-percent power with W-11.7 Unit 2 had already been evaluated and given interim approval on unit 1. The test director

was also aware that the same Instrument Maintenance efforts being carried out on unit 2 were also the basis for interim approval on unit 1. His submittal of the W-11.7 Unit 2 test data package to EN DES for approval with DN-1, DN-2, and EX-1 being unresolved did not constitute a previously unanalyzed condition.

Only DN-1 (W-11.7 Unit 2) concerning 0-percent power data was not addressed specifically in the Form 4 for W-11.7 Unit 1; however, as discussed previously in Question 4, this was a minor issue with no safety significance.

In addition to the discussion of the issues above, it should be noted that a discrepancy exists in the NRC inspection reports attached to the letter from Olshinski to White dated April 25, 1986. The discrepancy is between a statement in Section 6.m and a statement describing IFI 327, 328/86-12-02 near the top of page 23 of the report. The first statement referred to is ". . . and review of outstanding preoperational test open items and a determination made that their status does not constitute an unreviewed safety question." The second statement referred to is "Completion of the review and approval of all preoperational tests was identified by the licensee as an item requiring resolution prior to restart." The key point is that only those tests with outstanding items, not all tests, are involved in the review. The first statement correctly specifies those actions committed to in Volume I of the Nuclear Performance Plan and identified as NRC IFI 327, 328/86-12-02.

R.W. Fortenberry
R. W. Fortenberry

RMM 6/24
RMM:BBW:DLR
Attachment(s)
cc: RIMS, MR 4N 72A-C
SQN Master Files - W-11.7, Unit 1, test data package
SQN Master Files - W-11.7, Unit 2, test data package
H. D. Elkins, ONP, POB-2, Sequoyah
N. E. Featherston, DNE, W10A7C-K
J. E. Staub, DNE, DSC-D, Sequoyah (Attention: George Bell)

BBW.003

TI-2 - Unit 1
Data Sheet C5
Page 1 of 1
Rev. ________

FEEDWATER DIFFERENTIAL PRESSURE

Date 12/18/81   BD - 180 gpm

Power 100%

	Loop 1	Loop 2	Loop 3	Loop 4
Instrument No.	100271	100270	100268	100267
Calibration Date	1/15/81	1/15/81	1/15/81	1/15/81
Calibration Due Date	1/15/82	1/15/82	1/15/82	1/15/8[illegible]
Tap Number	2	1	2	1
Time	FW ΔP (volts)	FW ΔP (volts)	FW ΔP (volts)	FW ΔP (volts)
1322 CST	7.628	7.494	7.773	7.375
1325	7.627	7.557	7.815	7.364
1328	7.627	7.573	7.781	7.425
1331	7.648	7.516	7.792	7.397
1334	7.636	7.487	7.719	7.323
1337	7.604	7.597	7.746	7.303
Average Volts	7.6283	7.5323	7.7710	7.364[illegible]
Conversion Factor "H_2O = (Volts-2) x 80	~~[illegible]~~ 450.3	442.6	461.7	424.2

Remarks: 178 gpm @ 1320 CST
177 gpm @ 1339 CST

Data by: [illegible] / 12/1[illegible]

Reviewed by: [illegible] / 12/20

TI-2 - Unit 1
Data Sheet C5
Page 1 of 1
Rev. ______

FEEDWATER DIFFERENTIAL PRESSURE

Date 12/18/81

AD - 180gpm

Power 100%

	Loop 1	Loop 2	Loop 3	Loop 4
Instrument No.	100271	100270	100268	100267
Calibration Date	11/15/81	11/15/81	11/15/81	11/15/81
Calibration Due Date	11/15/82	11/15/82	11/15/82	11/15/8[illegible]
Tap Number	2	1	2	1
Time	FW ΔP (volts)	FW ΔP (volts)	FW ΔP (volts)	FW ΔP (volts)
1322 CST	7.628	7.494	7.773	7.375
1325	7.627	7.557	7.815	7.364
1328	7.627	7.573	7.781	7.425
1331	7.648	7.516	7.792	7.397
1334	7.636	7.487	7.719	7.323
1337	7.604	7.597	7.746	7.303
Average Volts	7.6283	7.5323	7.7710	7.364[illegible]
Conversion Factor "H_2O = (Volts-2) x 80	~~[illegible]~~ 450.3	442.6	461.7	429.2

Remarks: 178 gpm @ 1320 CST
177 gpm @ 1339 CST

Data by: [signature], 12/1[illegible]

Reviewed by: Kathryn A. Whitty, 12/20[illegible]

0 8 7 - 090569

Rosemount ΔP X-mitter

FT 3 TEST

G Strickland 247

690' Aux Bldg

11/16/81

Moving the 4 Rosemont ΔP transmitters from U-2 (AB 690' next to monitor tank) to U-1 side. Other Connect the rosemounts to the U-1 Foxboro FW ΔP draing ~~to~~ lines, vent the lines, and leave the rosemounts ready to use. Please have rosemounts installed by Wed 11/18/81. Contact G Strickland or R Forten Sarry 247 - 3430 to ~~TACF~~ starting work.

TACF

81-689-3

Due Date	Equipment Category			Clearance	SAP	OP Rest		RAP OK		
/ /			Test	NO	NO	NO		NO		yes

NO TFC

FILMED FROM BEST AVAILABLE COPY

11-17-81 11/17/81 2135 11/18/81 2135

11/17/81

0 0 0 2 4 9   0 9 6 6

Test Equip. Used	Cal. Due Da
466815	12-4-81
418928	6-6-82

INPUT		INST. NO. TVA 450631		
HEAD		Test Inst connected to 1-FT-3-103B		
PROCESS RANGE	INPUT CORR RANGE	Required	As Found	As Left
	0	2000 ±		2001
	160	4018 ±		4.017
	320	6.036 ±		6.049
	480	8.054 ±		8.063
	640	10.072 ±		10.070
	480	8054 ±		8.057
	320	6.036 ±		6.054
	160	4.018 ±		4.024
	0	2000 ±		2001
		±		

INPUT		INST NO. TVA 450627		
HEAD		Test Inst connected to 1-FT-3-55B		
PROCESS RANGE	INPUT CORR RANGE	Required	As Found	As Left
	0	2000 ±	1.983	2000
	160	4018 ±	3928	4017
	320	6.036 ±	5.879	6051
	480	8.054 ±	7.815	8.058
	640	10.072 ±	9.763	10.075
	480	8.054 ±	7.825	8.061
	320	6.036 ±	5.892	6.057
	160	4.018 ±	3.942	4.020
	0	2000 ±	1.983	2.000
		±		

[illegible] / [illegible] 11-17-81
Inst. Mechanic   Date

Reviewed   Date

000249 0965

INPUT		INST. NO. TVA 450628		
HEAD		Test Inst. connected to 1-FT-3-48B		
PROCESS RANGE	INPUT CORR RANGE	Required	As Found	As Left
	0	2000 ±	1.969	2.002
	160	4.018 ±	3.922	4.017
	320	6.036 ±	5.871	6.046
	480	8.054 ±	7.813	8.046
	640	10.072 ±	9.766	10.075
	480	8.054 ±	7.824	8.062
	320	6.036 ±	5.883	6.052
	160	4.018 ±	3.938	4.029
	0	2000 ±	1.969	2002
		±		

INPUT		INST NO. TVA 450629		
HEAD		Test Inst. connected to ~~1-FT-3-90A~~ 1-FT-3-90B		
PROCESS RANGE	INPUT CORR RANGE	Required	As Found	As Left
	0	2.000 ±	1.996	1.999
	160	4.018 ±	3.946	4.023
	320	6.036 ±	5.900	6.048
	480	8.054 ±	7.931	8.054
	640	10.072 ±	9.775	10.020
	480	8.054 ±	7.834	8.058
	320	6.036 ±	5.904	6.052
	160	4.018 ±	3.946	4.025
	0	2.000 ±	2.001	1.999
		±		

Test Equip. Used	Cal. Due Da
[illegible]	12-4-81
4[illegible]929	6-6-82

[illegible] / [illegible] 11 18 81

Inst. Mechanic Date

Reviewed Date