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OPPD
Omaha Public Power District

Fort Calhoun Station
P.O. Box 550, Highway 75
Fort Calhoun, NE 68023-0550

October 31, 2003
LIC-03-0148

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated August 28, 2003, Fort Calhoun Station (FCS) Unit No. 1 License Amendment Request, "Measurement Uncertainty Recapture Power Uprate" (LIC-03-0122)
3. Letter from NRC (A. B. Wang) to OPPD (R. T. Ridenoure) dated October 14, 2003, "Fort Calhoun Station Unit No. 1 – Measurement Uncertainty Recapture Power Uprate" (TAC No. MC0029) (NRC-03-198)

SUBJECT: Response to Request for Additional Information - Measurement Uncertainty Recapture Power Uprate (TAC No. MC0029)

The Reference 3 letter from the NRC included a Request for Additional Information (RAI) to support staff review of the Reference 2 License Amendment Request. This letter provides the Omaha Public Power District (OPPD) response to the RAI.

Please contact T. C. Matthews at (402) 533-6938 if you require additional information.

I declare under penalty of perjury that the forgoing is true and correct. (Executed on October 31, 2003) No commitments to the NRC are made in this letter.

Sincerely,

S. K. Gambhir

S. K. Gambhir
Division Manager
Nuclear Projects

TCM/tcm

A001

Attachment 1: Response to NRC Request for Additional Information
Attachment 2: FCS calculation FC6898, "Steam Generator Pressure and Feedwater Temperature Instrument Uncertainty Analysis"
Attachment 3: Independent Check of Calculations associated with the Calorimetric Uncertainty Evaluation

c: B. S. Mallett, Regional Administrator, NRC Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska

OPPD Responses to NRC Request for Additional Information Measurement Uncertainty Recapture Submittal

Instrumentation and Control

1. Provide a detailed description of the FCS plant-specific implementation of the guidelines in the Topical Report CENPD-397-P, Rev. 1, "Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology." This question is based on Item 1.1.C of RIS 2002-03.

OPPD Response:

In the August 28, 2003 OPPD letter (LIC-03-0122), Attachment 2, Section I.C (starting at page 17) includes discussion on the plant-specific implementation of the guidelines in the referenced topical report and the NRC safety evaluation that approved the topical report. In addition, Attachment 2, Section I.D of that letter addresses the NRC criteria per Item I.1.D of RIS 2002-03.

The CROSSFLOW system ultrasonic flow measurement (UFM) sensors at FCS are attached to a mounting bracket installed on the main feedwater supply header to the steam generators, consistent with the guidelines of CENPD-397-P-A. The CROSSFLOW sensors are installed approximately 54 pipe diameters downstream of the nearest elbow, in an area with fully developed flow conditions.

A plant-specific plant computer interface has been developed for use with the CROSSFLOW system. The CROSSFLOW/ERFCS interface provides data between the ERFCS (plant computer) and the CROSSFLOW computer. This data link sends the required plant data from the ERFCS to the CROSSFLOW computer (which generates a correction factor for feedwater flow), and returns the feedwater flow correction factor to the ERFCS. The CROSSFLOW UFM sensors will be used for continuous calorimetric power determination by data link to the plant computer system. New precision matched RTDs have been installed on each steam generator feedwater line for temperature measurement. An audible and visual alarm will be provided to alert plant operators when the UFM sensors are out of service. All components installed conform to the guidelines in CENPD-397-P-A.

2. Provide a detailed description of the FCS calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty. Justify that by using plant-specific data, the FCS total power measurement uncertainty is bounded within 0.4 percent. This question is based on Item 1.1.E of RIS 2002-03.

OPPD Response:

In Attachment 2 of letter LIC-03-0122, Section I.E (starting at page 20) includes information (in Table I-1) on the calculation of total power measurement uncertainty and the affected parameters.

The reactor thermal power (RTP) uncertainty is calculated by combining the individual error terms that contribute to uncertainty using square root sum of squares (SRSS) methodology, as described in Regulatory Guide 1.105 and ISA S67.04 and in accordance with approved plant methodology for instrument uncertainty calculations. The combination of these error terms is described by the following equation:

$$\epsilon_{RTP}^2 = \epsilon_{FWF}^2 + \epsilon_{FWT}^2 + \epsilon_{SGP}^2 + \epsilon_{MCO}^2 + \epsilon_{BDF}^2 + \epsilon_{BDT}^2$$

ϵ_{FWF} = Feedwater flow uncertainty
ϵ_{FWT} = Feedwater temperature uncertainty
ϵ_{SGP} = Steam Generator pressure uncertainty
ϵ_{MCO} = Moisture carry-over uncertainty
ϵ_{BDF} = Blowdown flow uncertainty
ϵ_{BDT} = Blowdown temperature uncertainty

The contribution that each term makes to the total RTP uncertainty is identified in Table I-1. The specific method used for the combination of the error terms is documented in Attachment 3 of letter LIC-03-0122. While there are additional error terms associated with the RTP uncertainty, the terms do not impact the final uncertainty. Attachment 3 of letter LIC-03-0122 also addresses the additional error terms and the impact they have on RTP uncertainty, and documents that, using the plant specific data and plant approved methodology, the FCS total power measurement uncertainty is bounded within ±0.4 percent.

3. Provide an independent "re-check" calculation based on a 0.4 percent uncertainty case to verify that the numbers calculated in the spreadsheet equations are correct. The calculation should be similar to the calculation in Attachment 3, "Calorimetric Uncertainty Evaluation," of your August 28, 2003, submittal (pages 54 through 60), which provided an independent re-check of the flow meter uncertainty calculation for a 0.3 percent uncertainty case. (Page 53 of Attachment 3 states that this attachment does not document the independent re-check of values at 0.4 percent). Because your license amendment request is based on the 0.4 percent uncertainty, the 0.3 percent uncertainty calculation in your August 28, 2003, submittal may not serve the purpose as an independent verification.

OPPD Response:

The applicable portions of the "Independent Check of Calculations" associated with the Calorimetric Uncertainty Evaluation have been revised to reflect a flow uncertainty of 0.4% (see Attachment 3 of this letter). The independent check shows numbers that are in agreement with the spreadsheet numbers in the calculation.

4. Provide a detailed description of the information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric: (i) maintaining calibration, (ii) controlling software and hardware configuration, (iii) performing corrective actions, (iv) reporting deficiencies to the manufacturer, (v) receiving and addressing manufacturer deficiency reports. This question is based on Item 1.1.F of RIS 2002-03.

OPPD Response:

In Attachment 2 of letter LIC-03-0122, Section I.F (starting at page 20) addresses calibration and maintenance procedures as summarized below.

(i) Maintaining calibration - Calibration and maintenance will be performed using site procedures developed from the CROSSFLOW system technical manual and plant operating and maintenance manuals. All maintenance work will be performed in accordance with site work control procedures.

(ii) Controlling software and hardware configuration – Any proposed hardware or software changes related to the CROSSFLOW system and its calibration and maintenance procedures will be controlled and evaluated by the plant design change process. This design change process includes applicable 10 CFR 50.59 evaluations.

(iii) Corrective actions - Corrective actions involving maintenance will be performed by qualified maintenance personnel, who are formally trained on the CROSSFLOW system. As with other maintenance and calibration activities, applicable deficiencies and corrective actions related to the CROSSFLOW system are documented in the FCS Condition Report (corrective action) system.

(iv) Reporting deficiencies to the manufacturer - Reliability engineering personnel will monitor the reliability of the CROSSFLOW system. Deficiencies are documented in the Condition Report system, and those meeting established criteria are reported to the manufacturer.

(v) Receiving and addressing manufacturer deficiency reports - The CROSSFLOW system vendor (Westinghouse) shall inform OPPD of any deficiencies in accordance with agreement reporting provisions. Manufacturer deficiency reports will be noted in the Condition Report system. These activities are consistent with the requirements of 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program."

5. Page 19 of Attachment 2 states that if the *CROSSFLOW* system is not returned to service within 24 hours, power will be reduced and maintained at the 1500 MWt levels until the *CROSSFLOW* ultrasonic flow measurements (UFMs) are returned to service. Provide the technical basis for the time selected. This question is based on Item 1.1.G of RIS 2002-03.

OPPD Response:

If the CROSSFLOW UFM system becomes unavailable, steady state plant operations at a core thermal output up to rated power may continue for a maximum of 24 hours after the last valid UFM correction factor was used in the calorimetric calculation for use in the daily nuclear power range surveillance. The 24 hour period is based on the minimum frequency for the calibration of the power range channels found in FCS Technical Specifications (TS). Per TS 3.1, Table 3-1, the power range channels are adjusted daily against a calorimetric balance standard (channel adjustment to agree with heat balance calculation). Since the nuclear power range channels will have been adjusted using the heat balance calculated with a valid CROSSFLOW UFM correction factor, the nuclear power range channel adjustment will be acceptable until the next performance of the surveillance.

The control room operators will receive a computer alarm if the CROSSFLOW UFM system becomes unavailable. The operators will then enter an operating procedure, which will direct them through the actions for a CROSSFLOW failure. The procedure will require that a power range nuclear instrumentation channel adjustment surveillance test be performed within one hour of the failure, using the last good correction factor. The CROSSFLOW system must then be returned to service prior to the next power range channel surveillance (24 hours from time of last good correction factor). If the CROSSFLOW system cannot be returned to service prior to the next surveillance time, reactor power will be reduced consistent with limits provided in the submittal. The basis for reducing power to 1500 MWt is the calorimetric uncertainty required by the Appendix K rule.

6. The August 28, 2003, submittal states that "Installation of new feedwater temperature resistance thermal detector (RTD) provides more accurate temperature measurement than that assumed in the development of original Appendix K requirements." Provide a detailed comparison between the new RTD and the existent temperature measurement instruments using the plant-specific data with respect to the uncertainty of the temperature measurements.

OPPD Response:

New feedwater temperature instrumentation has been installed to reduce the temperature measurement uncertainty. In addition to the new instrumentation a reduction in the calibration tolerance for the instrument loop was also implemented to further reduce the instrument loop uncertainty. The new instrumentation and calibration tolerance in combination reduces the total loop uncertainty from approximately ±4.8°F to less than ±0.8°F. The determination of this uncertainty value is documented in FCS calculation FC6898, "Steam Generator Pressure and Feedwater Temperature Instrument Uncertainty Analysis" (Attachment 2 of this letter). The temperature uncertainty value is applied in Attachment 3 of the application, Section 5.4 (5.4.2/5.4.3) of FC6896P, "Secondary Calorimetric Uncertainty Analysis."

The reduction in uncertainty is primarily accomplished through the use of transmitter/sensor matching with the new RTD/transmitters. This greatly reduces the overall temperature measurement error and results in a reduction in the transmitter/sensor uncertainty from approximately ±4.7°F (with the original RTD/transmitter combination) to less than ±0.65°F (new RTD/transmitter combination). Entering the temperature-resistance profile specific to the RTD into the transmitter results in transmitter – sensor matching and eliminates the sensor interchangeability error that exists with the current sensor/transmitter combination. In accordance with the vendor information for the new RTD, the sensor interchangeability error (if transmitter – sensor are not matched) is 2.34°F at 392°F and would result in an uncertainty value of 2.43°F when inputted into the uncertainty equation. Matching of the sensor to the transmitter eliminates this error and results in the calculated transmitter/sensor uncertainty value of less than ±0.65°F.

7. On September 5, 2003, Westinghouse issued a Technical Bulletin TB-03-6, "*CROSSFLOW* Ultrasonic Flow Measurement System Signal Issues" to all *CROSSFLOW* users. TB-03-6 identified a potential for contamination of the signals used to determine feedwater flow rate. There are potential errors in the correction factors, produced by the UFM, used in calorimetric calculation for plant power. The NRC staff has advised Westinghouse to verify the integrity of the information contained in previously approved topical report (CENPD-397-P-A, Rev. 1) for generic applications of the CROSSFLOW UFM, and to establish guidelines instructing users of the UFM how to operate their system in a manner that will minimize the potential for signal contamination in the future. Address the "Future Actions" listed in the TB-03-6 for the FCS plant.

<u>OPPD Response</u>:

Technical Bulletin TB-03-6 identified five items as Future Actions. OPPD initiated a Condition Report under the FCS corrective action program to evaluate the technical bulletin and track resolution of any corrective actions applicable to FCS. The five items are listed below with the current status.

1. Westinghouse/AMAG will complete the root cause analysis and communicate the detailed technical results to the CROSSFLOW User community.

STATUS: A draft root cause analysis has been forwarded to OPPD. OPPD will close out this item when the formal root cause analysis is received.

2. Westinghouse/AMAG will update the User's Manual to include technical criteria for identifying potential contamination issues associated with plant hardware changes.

   STATUS: Westinghouse informed OPPD of plans to issue a Nuclear Safety Advisory Letter in November 2003 that will include further guidance to the industry on monitoring for future potential signal contamination.

3. Westinghouse/AMAG will evaluate the viability of procedural changes to formally obtain and document the frequency spectrum analysis as part of the Quality Assured baseline plant data records.

   STATUS: This item in not applicable to FCS. Plant baseline data has already been obtained, analyzed, and found acceptable.

4. If baseline plant data records are currently unavailable, Westinghouse/AMAG will perform a frequency spectrum analysis to establish these records for future use.

   STATUS: This item in not applicable to FCS. Plant baseline data has already been obtained, analyzed, and found acceptable.

5. Westinghouse/AMAG will evaluate the viability of modifying CROSSFLOW electronics and associated software with the goal of protecting against the effects of potential signal contamination.

   STATUS: Westinghouse has informed OPPD that AMAG is developing new software to allow utilities to independently perform frequency spectrum analyses on demand.

OPPD plans to implement applicable Westinghouse/AMAG recommendations as identified above to maintain the operability of the FCS CROSSFLOW system. These items will be tracked, as noted previously, under the FCS corrective action program.

Materials Engineering

8. In Section VII.6.4, "Flow-Accelerated Corrosion (FAC) Program," a listed reference at the end of the section mentions the CHECWORKS program. However, in the body of Section VII.6.4, CHECWORKS is not specifically mentioned as the program used to predict changes in wear rates in piping and other systems as a function of power level uprates.

Please include a reference to CHECWORKS in the body of Section VII.6.4 and briefly describe how CHECWORKS is a part of your FAC program.

OPPD Response:

Section VII.6.4 has been revised as follows to include a new paragraph on CHECWORKS:

VII.6.4 Flow-Accelerated Corrosion (FAC) Program

The purpose of the FAC program is to predict, detect, monitor, and mitigate FAC in plant systems. The scope of the program includes all piping and components that cannot be demonstrated to be non-susceptible to FAC as documented in the current FAC Program System Susceptibility Evaluation. The program conducts ultrasonic pipe wall thickness measurements, predicts corrosion wear rate, establishes pipe section replacement criteria and initiates corrective actions to ensure that all applicable piping systems are adequate to continue performing their design function.

All FAC-susceptible components that are suitable for modeling are modeled using EPRI's CHECWORKS version 1.0g. CHECWORKS incorporates the predictive analysis techniques described in NSAC-202-L. CHECWORKS has the ability to predict wear rates and remaining component life for non-inspected components as well as inspected components. CHECWORKS takes into account such parameters as flow rates, component materials, fluid chemistry and thermodynamic conditions in determining wear rates.

The 1.6% MUR power uprate changes the operating pressure, temperature, quality and velocity in several of the BOP systems. Review of the FAC program and analyses performed by the FAC program erosion prediction model, using MUR power uprate conditions, concluded that:

- The MUR power uprate conditions affect the FAC wear rates in several BOP piping systems
- No additional piping systems should be added to the FAC program
- Changes to piping wear rates at the MUR power uprate conditions have been identified. Monitoring, and mitigating actions are being pro-actively planned in accordance with the FAC program requirements
- The FCS FAC program is adequate to support the MUR power uprate, and will include continued monitoring

The FAC program is not affected by the 1.6% MUR power uprate.

References (Section VII.6.4):
VII.6.4.1 PED-3 "Flow Accelerated Corrosion"
VII.6.4.2 CHECWORKS Predicted Wear Rates at the MUR Uprate Conditions

9. In Section VII.6.4, "Flow-Accelerated Corrosion (FAC) Program," it is stated that "The 1.6 percent MUR power uprate changes the operating pressure, temperature, quality, and velocity in several of the BOP [balance-of-plant] systems," and "Changes to piping wear rates at the MUR power uprate conditions have been identified." Based on FAC calculations for the 1.6 percent MUR power uprate, which plant component will have the largest increase in corrosion rate? How much does the corrosion rate increase in this component?

OPPD Response:

The components with the largest increase in wear rate are found in the moisture separator drain lines, with a projected wear rate increase of 7.5%. The components in these lines with the highest wear rate are the inlet nozzles to the moisture separator drain tanks, with a projected wear rate of 16.125 mils/year (an increase of 1.125 mils/year). This increase in wear rate is not a significant concern because (1) portions of these lines have been replaced, with the highest wearing segments being replaced with a corrosion resistant material; (2) the line pressure is relatively low, which translates into a larger available corrosion allowance; and (3) the CHECWORKS model of these lines is considered to be calibrated, which means that the remaining life predictions are considered to be accurate within the 50% tolerance band. These and other identified components subject to flow accelerated corrosion will continue to be monitored and maintained in accordance with the FCS FAC Program.

Mechanical Engineering

10. In Attachment 7 to the reference, OPPD indicated that the core shroud is the most critical component affected by the proposed power uprate due to the increased thermal loading in the reactor vessel internal structures. Discuss the rational that the power uprate is small (1.6 percent), but the stress for the girth rib flexure component increased from the current 19,632 psi for the current operating condition to 39,981 psi for the proposed 1.6 percent power uprate condition. With this large increase in stress for the core shroud, provide a summary of evaluation for other reactor vessel internals components such as core shroud barrel, control element assembly shroud assembly, core support plate and upper guide structure components that are affected by the proposed power uprate.

OPPD Response:

The proposed Appendix K power uprate for Fort Calhoun Nuclear Station will increase thermal loadings on the Reactor Vessel Internal (RVI) structures. Because the Appendix K power uprate is small (1.6%), this thermal loading increase will be minor. It was therefore assumed that any adverse effects on the RVI structures resulting from this power uprate would be confined to the Core Shroud, which is more sensitive than the other RVI components to minor variations in thermal loading. Accordingly, an evaluation of the Core Shroud under these increased thermal loadings was performed. This evaluation, as documented in Reference 1 and summarized in Reference 2, reprised the Core Shroud analysis-of-record (AOR), as documented in Reference 3 and modified as necessary to optimize methodology and to incorporate the increased thermal loadings (provided in Reference 4).

One set of Core Shroud assembly components considered in the AOR comprised the vertical panel segments, which are subjected to thermal stresses due to the temperature distribution through the thickness of the panel. In reviewing the AOR, it was discovered that only the temperature distribution at the elevation of the 6th girth rib was considered. Per References 1 & 4, the temperature distribution at the elevation of the 1st girth rib results in higher thermal stresses, and should have been used in the AOR.

Also evaluated in the AOR were the girth rib flexures, through which the individual Core Shroud segments are attached to the Core Support Barrel (CSB) via threaded structural fasteners. The temperature differential between the Core Shroud segments and the CSB results in relative thermal expansion, which is accommodated in bending of the flexures. In reviewing the AOR, the following two discrepancies were discovered:

1. The AOR used the Core Shroud-to-CSB temperature differential at an elevation between the 7th and 8th girth ribs. Per Reference 4, the temperature differential at an elevation between the 6th and 7th girth ribs is marginally greater, and should have been used in the AOR.
2. The AOR considered only the longer flexure on the straight segment panel assembly. For a given thermal deflection, thermal stresses are higher in the shorter flexure on the corner segment panel assembly, and the shorter flexure should have been considered in the AOR.

The Appendix K evaluation of the Core Shroud (Reference 1) corrected all of the discrepancies described above and, as a result, maximum thermal stresses in the panels, girth rib flexures and flexure-to-CSB bolts are much higher than those calculated in Reference 3. It should be noted, however, that all acceptance criteria are satisfied with these higher thermal stresses. Stresses in the other Core Shroud components, as calculated in Reference 3, are unaffected by these discrepancies and, in fact, are not affected by the Appendix K power uprate. Evaluations of the remaining RVI components, as documented in the appropriate AOR, are likewise unaffected by the discrepancies in the Core Shroud AOR. It should also be noted that the Appendix K power uprate conditions evaluated in Reference 1 bound the original design condition evaluated in the AOR (Reference 3). The Reference 1 evaluation thus replaces Reference 3 in supporting the current licensing basis.

A Condition Report in the OPPD/FCS corrective action system documents the identification of these discrepancies in the AOR.

References:

1. CN-CI-03-27 Rev. 00, "Evaluation of Core Shroud under Revised Thermal Loadings Associated with Appendix K Power Uprate", 6/12/03.
2. LTR-CI-03-29, "Transmittal of Fort Calhoun Nuclear Station Report on Evaluation of Core Shroud under Revised Thermal Loadings Associated with Appendix K Power Uprate", 6/12/03.
3. 23866-690-008 Rev. 0, "Omaha Stretch Power Study: Core Shroud Thermal Stress Analysis (1560 MWT)", 5/29/81.
4. CN-PS-03-9 Rev. 00, "Normal Operating Design Metal Temperatures for the Core Shroud for Ft. Calhoun for an Appendix-K Uprate (1526 MWt Power Level)", 6/10/03.

11. In Section IV.1 of Attachment 2, provide the calculated maximum stresses and fatigue usage factors at the critical locations of the reactor vessel including the outlet and inlet nozzles, the reactor pressure vessel (RPV) (main closure head flange, studs, and vessel flange), control rod drive mechanism housing, safety injection nozzles, external RPV supports brackets, bottom head to shell juncture, core support guides, and the incore instrumentation tubes, as a result of the power uprate. Also, provide the allowable code limits for the critical components evaluated, and the Code and Code Edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

OPPD Response:

The Design Criteria and Analytical Report for the FCS Reactor Vessel critical components is documented in the analysis of record FC-04277, "Analytical Report for Omaha Public Power District Reactor Vessel" April 1970, with Addenda noted in FC-04278, March 1973 and FC-04284, January 1983. The design was in accordance with the ASME Boiler and Pressure Vessel Code Section III, Nuclear Vessels and Special Case Rulings, 1965. The Safety Injection Nozzle information comes from analysis of record FC-04275, "Analytical Report for Omaha Public Power District Piping", January 1968.

The design criteria and parameters used in these analyses of record are as follows:

Design Pressure	2500 psia
Normal Operating Pressure	2100 psia

Design Temperature	650°F
Normal Operating Inlet Water Temp.	547°F (T_{cold})
Normal Operating Outlet Water Temp	598°F (T_{hot})

The analyses of record calculations were performed using the design temperature and pressure for the design loading conditions. The design criteria and operating parameters noted above that were used for the FCS Reactor Vessel evaluations bound the MUR uprate program design parameters that were identified in Attachment 2, page 15 of LIC-03-0122. The MUR uprate program operating conditions for FCS are anticipated as follows:

Normal Operating Pressure	2100 psia
Normal Operating Inlet Water Temp.	543°F (T_{cold})
Normal Operating Outlet Water Temp.	594.1°F (T_{hot})

Thus, the analysis of record design and operating conditions bound the MUR uprate program operating conditions. A reanalysis was not performed at MUR anticipated operating conditions.

The following table presents the analysis of record calculated results at the design pressure and temperature conditions noted above.

Fort Calhoun Station Analyses of Record
Reactor Pressure Vessel Critical Locations

Critical Location	Calculated Max. Stress, ksi	Fatigue Usage Factor	Allowable, ksi	Code Limit Fatigue Usage Factor	Code Edition
RV outlet nozzle[1,2]	Maximum range of stress intensity occurred on the outside of the nozzle vessel juncture, 48	U=0.33[1] U=0.28[2]	3 S_m=80	U=1.0	ASME Section III, 1965
RV inlet nozzle[1]	Maximum range of stress intensity for the combination of all loads, 48.7	U=0.07[1]	3 S_m=80.1	U=1.0	ASME Section III, 1965
RPV closure head flange[1]	Highest range of stress intensity occurs on outside surface, 46.7	U=0.24[1]	3 S_m=80	U=1.0	ASME Section III, 1965
RPV studs[1]	Highest range of stress was on the inside top surface of the stud, 80.5	U=0.71[1]	3 S_m=110.4	U=1.0	ASME Section III, 1965
RPV flange[1]	Highest range of stress intensity occurs on the inside surface, 54.5	U=0.02[1]	3 S_m=80	U=1.0	ASME Section III, 1965
CRDM housing[1]	Two locations stress intensity, 85.3 and 77.9[3]	U=0.47[1]	3 S_m=69.9	U=1.0	ASME Section III, 1965
Safety injection nozzles[4]	Maximum range of primary plus secondary stress intensity, 38.671	U=.088[4]	3 S_m=58.1	U=1.0	ASME Section III, 1965, through Addenda Winter 1967
Inlet and Outlet nozzle supports[1,2]	Membrane stress intensity in consideration of pipe break loads on vessel support, 44.4	U=0.20[1] U=0.16[2]	56.3	U=1.0	ASME Section III, 1965
Bottom Head to Shell Juncture[1]	Highest range of stress intensity occurred on the inside surface, 34.1	U=0.008[1]	80	U=1.0	ASME Section III, 1965
Core Barrel Stop Lugs[1]	Maximum stress bending stress, 24.1	N/A this is only for consideration of core barrel falling one inch	35	N/A	ASME Section III, 1965
Core Barrel Snubber Lugs[1]	Range of stress intensity, 64.2	U=0.03[1]	70	U=1.0	ASME Section III, 1965
Closure Head Instrumentation Penetrations[1]	Stress intensity, 52.3	U=0.37[1]	69.9	U=1.0	ASME Section III, 1965

[1] FC-04277, "Analytical Report for Omaha Public Power District Reactor Vessel", April 1970, CENC 1134
[2] FC-04278, "Addendum to the Analytical Report for Omaha Public Power District Reactor Vessel", March 1973, CENC 1134A-1
[3] For two locations, the range of primary plus secondary stress intensity was above 3 S_m. For those locations a plastic fatigue analysis was made as per N-417.6 (a) of ASME Section III.
[4] FC-04276, "Analytical Report for Omaha Public Power District Piping", CENC-1131

12. In Section IV.2 of Attachment 2, you indicated that reactor coolant loop piping analyses adhere to the ASME Boiler and Pressure Vessel Code, Section III and USAS B31.1. Provide the Code Editions and Addenda that are applicable for the reactor coolant system component evaluation for the proposed power uprate.

OPPD Response:

The design, fabrication, construction, inspection, testing and classification of all reactor coolant system components were in accordance with the ASME Boiler and Pressure Vessel Code, Section III-1965 and the Code for Pressure Piping USAS B31.1-1955, except as noted below.

The reactor coolant loop piping and fittings were designed and fabricated in accordance with the requirements of USAS B31.1, Power Piping Code, including all requirements of Code Cases N-9, and N-10 except that the Centrifugally Cast Stainless Steel Pipe was supplied in accordance with ASTM-A 451-72 specifications in lieu of the ASTM-A451-63 specifications listed in Case N-9. In addition, the thickness of the Reactor Coolant Pipe and Fittings met the requirements of ASME Section III through and including the Winter 1967 addenda, and a stress analysis similar to the requirements of ASME Section III was performed. Other reactor coolant pressure boundary piping and fittings, including the pressurizer safety and relief valve discharge piping, were designed and fabricated in accordance with the draft code for nuclear power piping (August 1968). These codes and classifications are found in FCS Updated Safety Analysis Report (USAR) Section 4.2.4. Code requirements are listed in Table 4.2-2 of USAR Section 4.2.

13. In Section IV.5 of Attachment 2, the design input parameters changes are provided on page 52, where you also indicated that these changes are well within the design envelope of the FCS steam generators (SG) and demonstrate that the power uprate will not affect SG performance. Accordingly, operation at the proposed power uprate is acceptable. Provide the design values for each of these listed parameters that are changed due to the power uprate.

OPPD Response:

Parameter	Design Value(1)	Design Operating Conditions(1)	Pre-Uprate Operating Conditions	Power Uprate Operating Conditions
Inlet Temperature (T_{hot})	650°F	598°F	593.8°F	594.6°F
Outlet Temperature (T_{cold})	650°F	547°F	542°F	543°F
SG Secondary Side Pressure	1000 psia	770 psia	822 psia	820 psia
Primary Side Pressure Drop	85 psi(2)	39 psi(2)	35.64 psi	35.66 psi
Steam Flow Rate	N/A	3.112×10^6 lbm/hr	3.311×10^6 lbm/hr	3.364×10^6 lbm/hr

Parameter	Design Value[1]	Design Operating Conditions[1]	Pre-Uprate Operating Conditions	Power Uprate Operating Conditions
Circulation Ratio	N/A	Approx. 5.0[3]	3.87	3.80
Secondary Fluid Mass Inventory	N/A	81,642[3] lbs	83,124 lbs	82,763 lbs
Secondary Side Pressure Drop	N/A	36.0 psi[4]	37.9 psi	39.1 psi
Average Heat Flux	N/A	51,133 BTU/hr-ft^2	57,617 BTU/hr-ft^2	58,593 BTU/hr-ft^2

(1) The design values are from the design specification (750S-23-2, Revision 2) and used by the fabricating shop to determine the wall thickness of the pressure vessel. In general, they were used to order the proper thickness of plate from the steel plant. The design operating conditions were used to perform a detailed analysis of the vessel to ensure the requirements of Section III of the ASME Code were satisfied. Thus, those values indicated as "N/A" were not specifically required to define the wall thickness of the pressure vessel but were required to perform the detailed stress analysis.

(2) The 85 psi value is based on the analysis of the primary side divider plate and assumes part-loop operation as well as reactor coolant pump starts and stops. The 39 psi value is based on normal full power operation.

(3) This value was calculated during the initial fabrication of the vessel using a 1-dimensional computer code. It can not be directly compared to the values calculated for the pre-uprate and power uprate conditions. These values were calculated using a more sophisticated 3-dimensional computer code.

(4) This value was based on the original design operating conditions.

14. In Section IV.6 of Attachment 2, OPPD indicated that a review of the revised temperature parameters show that any changes in T_{hot} and T_{cold} are very small and are bounded by the existing pressure stress analysis performed for the FCS (WCAP-15889, Rev. 0, Table 8.1.4). Provide a summary of evaluation and confirm that delta T_{hot} between the pressurizer and the hot leg temperature, and Delta T_{cold} between the pressurizer and the cold leg temperature for the proposed power condition are bounded by the design basis values. Provide the Code and the Code Edition for the evaluation of the pressurizer and surge line piping for the power uprate condition.

OPPD Response:

Evaluation of Delta between T_{hot} and pressurizer, and T_{cold} and pressurizer

The temperature difference (delta) between T_{hot} at MUR conditions and the pressurizer will actually decrease. Anticipated T_{hot} is 594.1°F for MUR conditions as noted in Attachment 2 of LIC-03-0122, page 15. The pressurizer temperature (643°F) will not change; this is due to the fact that the pressurizer pressure will remain at 2100 psia. With the pressurizer operating at saturated conditions, the temperature will remain unchanged (i.e., 643°F). Currently the delta between T_{hot} and the pressurizer is approximately 49.7°F. With the MUR uprate, the delta between T_{hot} and the pressurizer will be 48.9°F. Thus, the current conditions are bounding for the MUR uprate with respect to the temperature difference between the hot leg temperature and the pressurizer.

The temperature difference between T_{cold} at MUR conditions and the pressurizer will remain the same. This is due to the fact that T_{cold} will not change for the MUR uprate. (See page 15 of Attachment 2 LIC-03-0122.) Thus, the temperature difference between the cold leg temperature and the pressurizer remains the same.

Code and Code Editions

The applicable code and code edition for evaluation of the surge line piping is USAS B31.7-1965. The applicable code and code edition for evaluation of the pressurizer is ASME III, 1965 through winter addenda 1966.

Pressurizer Design Criteria

The pressurizer was designed in accordance with the ASME Section III, Nuclear Vessels and Special Case Rulings, 1965 through Winter Addenda 1966. The design parameters used were:

Design Pressure	2500 psia
Design Temperature	700°F
Normal Operating Pressure	2100 psia
Normal Operating Temperature	643°F

The analysis of record calculation FC-04275, "Analytical Report for Omaha Public Power Pressurizer," CENC-1130, April 1970, was performed using the design temperature and pressure for the design loading conditions. The design also evaluated a normal operating temperature of 643°F for plant transient conditions. The design criteria and operating parameters noted above that were used for the pressurizer evaluations bound the MUR uprate program design parameters that were identified in Attachment 2, page 15 of LIC-03-0122. The MUR uprate program operating conditions for the RCS are anticipated as follows:

Normal Operating Pressure	2100 psia
Normal Operating Inlet Water Temp.	543°F (T_{cold})
Normal Operating Outlet Water Temp.	594.1°F (T_{hot})

Because the normal operating pressure will be held constant at MUR conditions (2100 psia), the pressurizer temperature will remain the same at 643°F (saturated condition). Thus, the analysis of record still remains bounding for the MUR conditions.

15. In Section VII.6 of Attachment 2, OPPD evaluated the FCS motor-operated valve and air-operated valve programs for the MUR power uprate conditions. Confirm whether and how your responses to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," on the thermal binding and pressure locking of safety-related power-operated gate valves and to GL 96-06, "Assurance of Equipment

Operability and Containment Integrity During Design Basis Accident Conditions," regarding overpressurization of isolated piping segments are acceptable for the MUR power uprate conditions.

OPPD Response:

GL 95-07 "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves"

The OPPD responses to Generic Letter 95-07 identified the safety related power operated gate valves that are susceptible to pressure locking and thermal binding, and described corrective actions. OPPD has subsequently reviewed these valves based on the MUR operating conditions, using the same criteria used in the original evaluation, and concluded that:

1. The safety related power operated gate valves identified as susceptible at the current operating conditions will also be susceptible at the MUR conditions.
2. The causes of pressure locking or thermal binding (i.e., exposure to heated flow from the boric acid storage tank, heated flow from the shutdown cooling suction header, exposure to elevated upstream or downstream pressure from feedwater pumps, bonnet heating in the event of a MSLB) at the MUR conditions will be similar to the causes at the current operating conditions.
3. The implemented corrective actions are adequate to preclude pressure locking or thermal binding of safety related power operated gate valves at the MUR conditions.

Therefore, the OPPD responses to GL 95-07 are acceptable for the MUR power uprate conditions.

GL 96-06 "Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions"

The OPPD response to Generic Letter 96-06 concluded that:

1. No vaporization would occur in the containment air cooling coils prior to Component Cooling Water (CCW) pump start since the CCW surge tank water level is 41" and the minimum nitrogen overpressure is 34 psig. Because there is no vaporization prior to the CCW pump start, the potential for waterhammer does not exist.

2. No two phase flow condition would occur after a design basis accident in the CCW system and containment air coolers that would adversely affect their ability to perform their credited accident mitigation function.

3. The structural integrity of the piping pressure boundary between closed containment isolation valves will be maintained during the heatup of post design basis accident temperature transients.

The above conclusions were based on evaluation of the Loss of Cooling Accident (LOCA) and Main Steam Line Break (MSLB) design basis accidents, assuming 102% thermal power initial conditions. Thus, the evaluation is bounding and the OPPD response to GL 96-06 is acceptable for the MUR power uprate conditions.

LIC-03-0148
Attachment 2
FCS calculation FC6898, "Steam Generator Pressure and Feedwater Temperature Instrument Uncertainty Analysis"

CALCULATION COVER SHEET

Calculation Number: FC6898 QA Category: [ ] CQE [X] Non-CQE	Page No.: 1 Total Pages: 42
Calculation Title: Steam Generator Pressure and Feedwater Temperature Instrument Uncertainty Analysis	Short Term Calc: [ ] Yes [X] No Vendor Calc. No.: Associated Project:
Software Tracking No.: (from PED-MEI-23, if applicable)	Responsible NED Dept No.: 356

Owner Assignment (by Dept Head):
(Required only if there are affected documents to be changed)

OPPD Engineer Assignment (by Dept Head): JOHN ADAMS
(Required only for verification of vendor/contractor calculations)

Verification of Vendor/Contractor Calc. assumptions, inputs and conclusions complete:

OPPD Engineer: Date:

APPROVALS - SIGNATURE AND DATE
(Multiple preparers shall identify section prepared per PED-QP-3, Section 4.3.)

Rev. No.	Preparer(s)	Reviewer(s)	Required for CQE Independent Reviewer(s)	Supersedes Calc No.	Confirmation Required? Yes	Confirmation Required? No
0	[signature] 5/13/03	[signature] 5/27/03				✓

CALCULATION COVER SHEET

Calculation Number: FC6898	Page No.: 2

Applicable System(s) / Tag Number(s)

P-902
P-905
T-1396
T-1399

EA's and/or Calculations Used as input in this Calculation

NONE

EXTERNAL ORGANIZATION DISTRIBUTION (Groups affected by this calculation)

Name and Location	Copy Sent (✓)	Name and Location	Copy Sent (✓)

CALCULATION REVISION SHEET

Rev. #	Description/Reason for Change
0	Original Issue

PRODUCTION ENGINEERING DIVISION
QUALITY PROCEDURE FORM

PED-QP-3.5
R6

Calculation No. FC6898

Rev.: 0

	REVIEWER'S CHECKLIST-CALCULATIONS	Yes	No	N/A
1.	Is Calculation Cover Sheet Form PED-QP-3.1 completed addressing all the blocks and included with the calculation?	✓		
2.	Is the calculation objective stated? Was this achieved?	✓		
3.	Are inputs correctly selected and incorporated into the calculation and listed on Form PED-QP-3.1?	✓		
4.	Have inputs and/or assumptions which require confirmation at a later data, been identified on the Calculation Cover Sheet Form PED-QP-3.1 and in the calculation body?			✓
5.	Are the applicable codes, standards, regulatory requirements and other references including issue and addenda identified such that they are traceable to source document?	✓		
6.	Was an appropriate calculation method used? Was the basic theory appropriate?	✓		
7.	Have assumptions been noted and justified?	✓		
8.	Are the calculations free of arithmetic errors?	✓		
9.	Is the calculation consistent with the design basis requirements?	✓		
10.	Is the conclusion stated?	✓		
11.	Is the calculation legible and suitable for microfilming?	✓		
12.	Has Form PED-QP-3.2 been used and correctly completed for calculation revision?	✓		
13.	Has Form PED-QP-3.9 been used and correctly completed?	✓		
14.	If the calculation has been prepared to supersede another calculation, has all the valid information been transferred in the new calculation?	✓		
15.	If the calculation determines that an existing or preexisting condition may be outside the design basis of the plant, has a Condition Report been submitted?			✓
16.	Have As Built requirements and affected documents been identified in Form PED-QP-3.8.	✓		

Comments:

[signature] / 5/27/03

Reviewer Date

PRODUCTION ENGINEERING DIVISION
QUALITY PROCEDURE FORM

PED-QP-3.7
R5

Calculation No. FC6898
Rev.: 0

INDEPENDENT REVIEWER'S CHECKLIST-CALCULATIONS		Yes	No	N/A
1.	Are the calculation methods accurate and appropriate?	✓		
2.	Are all inputs sufficiently detailed and listed on Form PED-QP-3.1?	✓		
3.	Are the calculation assumptions reasonable?	✓		
4.	Has the basis for engineering judgement been included in the calculation, when used?	✓		
5.	Is the calculation documented sufficiently such that the analysis is understandable to someone competent in the discipline without recourse to the Preparer?	✓		
6.	Have the design interface requirements been satisfied?	✓		
7.	Are the results reasonable and do they resolve the calculation objective?	✓		
8.	If an alternate calculation was used to verify the adequacy of the calculation, is it attached to the calculation?			✓
9.	If qualification testing was used to verify the adequacy of the calculation, has it been documented using a retrievable source, or attached to the calculation?			✓
10.	Are calculations involving Technical Specification values and associated margins of safety identified?			✓

Comments:

[signature] , 5/27/03

Reviewer Date

Calculation No. FC6898

Page No. 0

CALCULATION AFFECTED DOCUMENTS

The Calculation Preparer is to identify documents affected by this Calculation. Markups are to be provided in an Attachment to the Calculation except those noted with an *. Changes not involving procedures should follow the associated change process. The preparer is to indicate below how the Calculation is to be processed by Document Control.

	Not Required, Calculation supports EC#____ or is used to support EA-FC- - this form can be signed off by the Calculation Preparer. Calculation "As Built" follows direction given for modifications.
	EC, FLC, Preapproved NRC commitment change, or Condition Report need identified. Calculation is closed on receipt of the completed PED-QP-3.8 form.
	Change to a DBD, USAR, etc., without a change to plant procedures identified. Calculation is "As Built" on receipt of the completed PED-QP-3.8 form.
	Change to a DBD, USAR, etc., and plant procedures (no hardware) identified. Calculation is "As Built" on receipt of the completed PED-QP-3.8 form.
	No document changes or other changes are required. Calculation "As Built" on receipt of the completed PED-QP-3.8 form.

NOTE: Markups are to include any inputs or assumptions which define plant configuration and/or operating practices that must be implemented to make the results of the Calculation valid. The Calculation may provide the basis for a 10CFR50.59 analysis or substantiate a 10CFR50.59 analysis.

Affected Documents		
Document Type	Document Number (N/A = not applicable)	Procedure Change No., FLC No., etc.
Emergency Operating Procedure*	N/A	
Abnormal Operating Procedure*		
Annunciator Response Procedure		
Technical Data Book		
Surveillance Test Procedure		
Calibration Procedure	IC-CP-01-1396 IC-CP-01-1399	NOTE 1
Operating Procedure	N/A	

Calculation No. FC6898

Page No. 0

Affected Documents		
Document Type	Document Number (N/A = not applicable)	Procedure Change No., FLC No., etc.
Maintenance Procedure	N/A	
PM Procedure		
EP/EPIP/RERP*		
Security Procedures * (Safeguards)*		
Operating Instructions		
System Training Manuals		
Technical Specification*		
USAR		
Licensing Commitments		
Standing Order		
Security Plan (Safeguards)		
CQE List		
Vendor Manual Changes		
Design Basis Documents		
Equipment Database		
Oil Spill Prevention, Control and Countermeasure (SPCC) Plan		
EEQ Manual		
SE-PM-EX-0600		
Updated Fire Hazard Analysis		
EPIX		
Electrical Load Distribution Listing (ELDL)		
Station Equipment Labeling		
Engineering Analysis		

Calculation No. FC6898

Page No. 0

Affected Documents		
Document Type	Document Number (N/A = not applicable)	Procedure Change No.; FLC No., etc.
Calculations	N/A	
Drawing Number		
Drawing Number		
Other		

Completed by Owner (if Plant Procedure Changes Required or N/A):	Date:
Completed by Preparer: [signature]	Date: 5/13/03

NOTE 1 PROCEDURE CHANGE WILL BE MADE TO DOCUMENT AS-LEFT VALUE FOR TEMPERATURE INPUT TO THE PLANT COMPUTER. THE CHANGE WILL BE TRACKED UNDER MODIFICATION EC29825 WHICH CHANGES OUT THE INSTRUMENTATION.

PRODUCTION ENGINEERING DIVISION
QUALITY PROCEDURE FORM

PED-QP-3.9
R1

Calculation No. FC6898

Rev.: 0

CALCULATION PREPARER CHECKLIST		Yes	No	N/A
1.	Are all ASSUMPTIONS necessary to perform the calculation adequately described?	✓		
2.	If applicable, has the use of Engineering Judgement been documented per PED-QP-14?			✓
3.	Have applicable licensing commitments regarding the Calculation been met?	✓		
4.	Is the computer program identification number (Ref. PED-MEI-23, Section 5.3.1) on the coversheet as part of the Calculation description? **NOTE:** Only applies to DEN-Mechanical and Electrical/I&C Departments.			✓
5.	Is the computer code title and version/level properly documented in the Calculation?			✓
6.	Is the listing or file reference (computer file name and file location) of the final computer input and output provided?			✓
7.	Does the computer run have page number and alphanumeric program number on every sheet?			✓
8.	Have updates been prepared or described for any affected documents as identified on Form PED-QP-3.8? This includes assumptions that may affect plant procedures or design documents.			✓
9.	Where appropriate, have the necessary 10CFR50.59 (FC-154 or FC-155) evaluations been drafted to support changes to procedures or design documents? The FC-154 forms are not to be signed by a qualified reviewer until the calculation reviews are complete.			✓
10.	If required has a Condition Report been prepared and/or submitted in accordance with SO-R-2?			✓
11.	If a Commitment to the NRC that is not part of the FCS Design Basis must be changed to implement this Calculation, has Licensing been notified of the proposed change? Certain Commitments require prior NRC approval before implementing the change. Has the necessary approval been obtained? See NOD-QP-34 for additional guidance.			✓
12.	Does Form PED-QP-3.8 define the Calculation As Built requirements?	✓		

Comments:

Signature: [signature] Date: 5/13/03

Department/Organization:

TABLE OF CONTENTS

1.0 PURPOSE

To determine the individual instrument loop uncertainties for the Steam Generator Pressure and Feedwater Temperature instrumentation used in the plant calorimetric power calculation XC105. The Total Loop Uncertainty (TLU) will be calculated for only those portions of the instrument loops used to provide input to the plant computer for the calorimetric calculation. The instrument loops addressed in this calculation include:

P-902	Steam Generator 2A Pressure
P-905	Steam Generator 2B Pressure
T-1396	Steam Generator 2A Feedwater Temperature
T-1399	Steam Generator 2B Feedwater Temperature

The instrumentation is used during normal operation and therefore the uncertainties are calculated for normal environmental and operating conditions.

2.0 REFERENCES

The following are references used in developing this document.

2.1 OPPD Production Engineering Division Procedure, "Calculation Preparation, Review and Approval", PED Quality Procedure QP-3, Revision 3, dated 4/8/94.

2.2 OPPD Production Engineering Division Standard, "Instrument Loop Uncertainty Setpoint / Tolerance Calculation Methodology" Document Number EEI-3.

2.3 ISA RP67.04, Part II, 1994, Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."

2.4 OPPD Fort Calhoun Station Interconnect Diagrams as follows:

a. Drawing 161F561, Sheet No. 122, Rev. 35 (for loop A/P-902 & A/P-905), GSE FILE NUMBER 9620

b. Drawing 136B2331. Sh 79A, Rev. 3 (for loop T-1396), GSE FILE NUMBER 23181

c. Drawing 136B2331, Sh 80A, Rev. 3 (for loop T-1399), GSE FILE NUMBER 23168

2.5 OPPD Fort Calhoun Station Instrument and Control Equipment List as follows:

a. Drawing EM-902, Sheet No. 1, Rev. 14 GSE FILE NUMBER 15719

b. Drawing EM-905, Sheet No. 1, Rev. 14, GSE FILE NUMBER 11639

c. Drawing EM-1395/1399, Sheet No. 1, Rev. 15 GSE FILE NUMBER 15876

d. Drawing EM-1395/1399, Sheet No. 2, Rev. 6, GSE FILE NUMBER 15877

2.6 Fort Calhoun Station Unit No. 1 Calibration Procedure IC-CP-01 -1396, "Calibration of Steam Generator RC-2A Feedwater Temperature Loop T-1396".

2.7 Fort Calhoun Station Unit No. 1 Calibration Procedure IC-CP-01 -1399, "Calibration of Steam Generator RC-2B Feedwater Temperature Loop T-1399"

2.8 Fort Calhoun Station Unit No. 1 Surveillance Test IC-ST-MS-0030; "Channel Calibration of Steam Generator RC-2A Channel A Pressure Loop A/P-905".

2.9 Fort Calhoun Station Unit No. 1 Surveillance Test IC-ST-MS-0026; "Channel Calibration of Steam Generator RC-2A Channel A Pressure Loop A/P-902".

2.10 Rosemount Product Data Sheet, Models 644H and 644R Temperature Transmitters, 00813-0100-4728, Rev. HA.

2.11 Rosemount Product Data Sheet, Series 78 Temperature Sensors, 00813-0100-2654, Rev CA.

2.12 Instruction Manual TD F180.0190 Foxboro Manual PSS 9-1B1 A; N-E11 and N-E13 Series Nuclear Electronic Pressure Transmitters.

2.13 Instruction Manual TM T068.0010, Operation and Maintenance Manual for TEC Model 156 Analog Signal Isolator.

2.14 Instructions, Form 1433-0100-D, Type 1433 Decade Resistor, June 1978

2.15 Fort Calhoun Station Unit 1 Calibration Procedure IC-CP-03-0005, "Calibration of Eaton Model UPS 3000 Digital Pressure Indicator".

2.16 Fort Calhoun Station Unit 1 Calibration Procedure IC-CP-03-0121, "Calibration of General Radio Type 1433 Decade Resistor".

2.17 Fort Calhoun Station Unit 1, Standing Order SO-M-028, Control of Measuring and Test Equipment.

2.18 Foxboro Product Specification PSS 2A-1B3A, "E11GM and E11GH Electronic Gauge Pressure Transmitters."

2.19 Fort Calhoun Nuclear Station Technical Specifications.

2.20 Technical Manual MODACS III, Rev. 4, Reissue 3, December 1981

3.0 ASSUMPTIONS AND GIVEN CONDITIONS

The following assumptions and given conditions (A&GC) are used in development of this calculation.

There are no assumptions used in this calculation that require verification.

3.1

a. In many cases, the calibration or setting tolerance used in plant calibration procedures is different than the manufacturer's accuracy specification. To determine a components uncertainty, the larger of either the manufacturer's accuracy specification or the calibration tolerance will be used.

b. For the instrument loops considered by this analysis the tolerance associated with the loop check will be applied one time as the overall accuracy of the loop. This can only be done if the loop calibration tolerance is larger than the accuracies of the individual instruments in the loop.

c. From Reference 2.20, the A/D accuracy for the plant computer is calculated based on the following error terms. Resolution (12 bit $1/2^{12} = 1/4096 = 0.025\%$ includes 1/2LSB), Offset Setability (0.01%), Gain Accuracy (0.01%), Linearity Error (0.025%) and Noise (0.02%). Combining these terms using SRSS gives an A/D accuracy of 0.043% of Span.

d. This calculation is based on instrumentation that is installed as part of the Appendix K power uprate. As part of this, new feedwater temperature instrumentation is installed and the loop calibration procedures are changed to incorporate tighter calibration tolerances.

3.2 TRANSMITTER A/P-902/905:

a. Temperature Effect (TE) - Foxboro does not provide a specification for the temperature effects of the NE11GM transmitters. However, Reference 2.18 provides a temperature effect (zero shift) of ±1% span/100F temperature shift for non-nuclear transmitters calibrated between 80% and 100% of span. A 50F temperature shift will be used based on a review of the operating temperatures in containment. The data shows temperatures in the area where the transmitter is located ranged between 90F and 105F during the summer months when the temperature shift is the greatest.

Assuming a calibration temperature of 65F gives a delta of 40F. Based on this the use of 50F is conservative. A 50F temperature shift between calibration conditions and operating conditions gives an error of ±1% span/100F x 50F = ±0.5% span or 5 psi. This equates to 0.1 psi/F. This was compared to actual plant data taken over a three month period (Attachment D). The data shows an approximate 2 psi/15 F (0.133 psi/F) change in containment temperature (conservatively assuming the entire change in S/G pressure is due to a temperature effect). Based on this, a temperature effect of 0.15psi/F will be selected to bound these two numbers. Assuming a 50F temperature offset from the calibration point gives an error of 7.5psi (0.15psi/F x 50F)

TE_{TRAN} = ±7.5 psi

b. Drift (DR) - Reference 2.12 provides a Drift/year specification of ±0.25% of calibrated span. For a 22.5 month calibration interval (18 mo x 1.25) this equates to a drift of ±0.469% span or 4.69 psi (1.5(0.25)) ÷ 100 x 1000). Therefore:

DR_{TRAN} = ±0.469% span = ±4.69 psi

c. Measurement & Test Equipment (MTE) - For this analysis, the M&TE uncertainty used will be that associated with the loop check. This uncertainty will be applied to that portion of the instrument loop providing input to the plant computer for use in the calorimetric calculation. Therefore the individual device M&TE is zero.

d. Power Supply Effect (PSE) - Reference 2.12 does not provide a specification for the transmitter power supply effect. The transmitter power supply effect is considered negligible considering the AC to the loop power supply is fed from a static inverter "C" and the power supply is operated within its rated voltage range. Therefore:

PSE_{TRAN} = negligible

e. Process Measurement Effect (PME) - Reference 2.19 requires that the Containment internal pressure shall not exceed 3 psig. A 3 psig containment pressure results in a +3 psi bias on the transmitter. Therefore:

PME_{TRAN} = + 3 psi

f. Sensing Line Head - It is assumed that any transmitter sensing line head is calibrated out during the transmitter calibration.

g. Normal Radiation Effect (RAD) - Reference 2.12 provides a specification for radiation effect of ±0.5% of span. ±0.5% of span x 1000 psi = 5 psi. Therefore:

RAD_{TRAN} = ±5 psi

h. Calibration Tolerance (CAL) vs. Accuracy (AA) - From Reference 2.12, the transmitter Accuracy is ±0.5%, Hysteresis is ±0.1%, Dead Band is ±0.05% and Reproducibility is ±0.15%. The Reproducibility includes effects of hysteresis, repeatability, dead band and drift over a one-hour period. Therefore the Accuracy of the transmitter is ± 0.52% or ± 5.2 psi $[(0.5^2 + 0.15^2)^{0.5} \div 100 \times 1000]$.

From Data Sheet 8 of Reference 2.8 and 2.9, the calibration tolerance for the Steam Generator pressure transmitter is 0.55%. This corresponds to ± 5.5 psi (1000 psi x 0.55%). Because the calibration tolerance is larger than the device accuracy, the calibration tolerance will be used in lieu of accuracy (see 3.1).

3.3 SIGNAL ISOLATORS A/PM-902/905:

a. Drift and Stability (DR) - Reference 2.13 provides a Drift specification of ±0.05%/C for Gain and 30uV/C for offset. The isolators are located in the control room, therefore a temperature deviation of 5C (9F) is considered conservative. Applying this to the error terms: ±0.05%/C x 5C = 0.25% (gain) and 30uV/C x 5C/.8V = ±0.02% (offset). Combining these gives: $[(0.25^2 + 0.02^2)^{0.5} \times 1000/100] = 3.2$ psi
Therefore:

DR_{isol} = ± 3.2 psi

b. Measurement & Test Equipment (MTE) – For this analysis, the M&TE uncertainty used will be that associated with the loop check. This uncertainty will be applied to that portion of the instrument loop providing input to the plant computer for use in the calorimetric calculation. Therefore the individual device M&TE is zero.

c. Dropping Resistors – The drift and temperature effect of the dropping resistors on the input and output of the isolators is considered negligible. Additionally, any inaccuracy of the resistor would be calibrated out during calibration of the isolators.

d. Calibration Tolerance (CAL) vs. Accuracy (AA) – From Reference 2.13, the isolator Accuracy is ±1% (gain) and ±2mV (0.002V/0.8V = 0.25%) for offset, Linearity is ±0.2%, Reproducibility and Repeatability, are included within the Accuracy. Combining these terms gives an Accuracy of ±1.05% or ±10.5 psi $[(1.0^2 + 0.25^2 + 0.2^2)^{0.5} \times 1000]$.

From Data Sheet 9 of Reference 2.8 and 2.9, the calibration tolerance for the loop input to the ERF computer is ± 10 psi. Because the accuracy (AA) is larger than the calibration tolerance, the device accuracy will be used in lieu of calibration tolerance (see 3.1).

3.4 TEMPERATURE ELEMENTS AND TRANSMITTERS T-1396/1399:

Each RTD and its associated temperature transmitter are supplied as a matched set. Transmitter-Sensor matching is accomplished by entering the temperature-resistance profile specific to the RTD into the transmitter. This eliminates the sensor interchangeability error, which greatly improves accuracy. Because of this, the system accuracy is considered for both the transmitter and RTD, therefore it is necessary to address the overall error for these devices in combination.

a. T-1396/1399 Accuracy (AA) – Reference 2.11 specifies an accuracy of ±0.52F for the sensor when matched to the transmitter. From Reference 2.10, the model 644R transmitter has an accuracy of ±0.27F and ±0.03% span for D/A conversion. Combining these terms using SRSS gives a combined accuracy for the transmitter/RTD of ±0.62F. $(0.52^2 + 0.27^2 + (0.0003 \times 700)^2)^{1/2}$.

$AA_{TE} = \pm\ 0.62$ F

b. TE-1396/1399 Self Heating Effect (SHE) - Reference 2.11 specifies a Self Heating effect of ± 1.8F for 16mW power dissipation. Based on discussions with Rosemount, the excitation current is 220 E-6 amps. Using a resistance and temperature of 465.3 ohms at 665F gives a power dissipation of 2.25 E-5 watts, (220 E-6)**2x(465.3). The corresponding self heating effect is 1.8F/16mW x 2.25 E-5 = 0.0025F. Based on this, the self heating effect is considered to be negligible.

SHE_{TE} = negligible

c. Transmitter Temperature Effect (TE) - Reference 2.10 provides a temperature effect for the transmitter of 0.0054F/1.8F change in temperature with a D/A effect of 0.001% of span. These transmitters are located in the control room which is maintained at a constant temperature during normal operation. Therefore assuming a 10F temperature change is conservative. The temperature effect is 0.0054F/1.8F x 10F = 0.03 and 0.00001 x 700F = 0.007F. Combining these two terms using SRSS gives a total temperature effect of 0.031F

$TE_{TM} = \pm 0.031$F

e. Sensor Temperature Stability (ST) - Reference 2.11 provides a temperature stability of ±0.11% maximum ice-point resistance shift. Using an ice-point of 32F results in a ±0.0352F (0.0011 x 32) stability effect.

$ST_{TE} = \pm 0.035$F

f. Measurement & Test Equipment (MTE) - For this analysis, the M&TE uncertainty used will be that associated with the loop check. This uncertainty will be applied to that portion of the instrument loop providing input to the plant computer for use in the calorimetric calculation. Therefore the individual device M&TE is zero.

f. Calibration Tolerance (CAL) vs. Accuracy (AA). From step 7.6.2 from references 2.6 and 2.7, there is no calibration tolerance for the loop input to the ERF computer. The step zeros out any error that exists at the 400F point. This analysis will conservatively assume a 0.5F error at this point. Because the calibration tolerance is larger than the device accuracy, the calibration tolerance will be used in lieu of accuracy (see 3.1).

3.5 MEASUREMENT & TEST EQUIPMENT (MTE):

a. MTE FOR A/P-902/905 - References 2.8 & 2.9 specifies the use of either a Druck model DPI 145 (0-2000psi) or an Eaton model UPS3000 (0-1000psi) From Reference 2.15 the accuracy of the Eaton is ±0.13% FS or 1.3 psi. From Reference 2.17 the Druck has an accuracy of the greater of 0.07% of reading (0.7 psi) or 0.15% of full scale (0.6 psi). Both of these values are less than the accuracy of the Eaton. Therefore, for this calculation the accuracy of the Eaton gauge (1.3 psi) will be used. Based on review of the manufactures data, the accuracy term includes temperature effect. References 2.8 & 2.9 perform a loop check of the S/G pressure input to the plant computer. In this check the only piece of M&TE used is the pressure source, therefore this is the M&TE accuracy for this loop.

$MTE_{PRES}= \pm 1.3$ psi

b. MTE FOR T-1396/1399 - References 2.6 & 2.7 specifies the use of General Radio Model 1433-W decade box. From Reference 2.16, the accuracy of the Decade Box is ±0.01% of the dial setting +0.002 ohms. Using the largest dial setting of 235 ohms gives an accuracy of (235 x 0.01% +0.002 ohms) = ± 0.0235 +0.002 ohms. Conservatively combining these two terms using the SRSS method gives an accuracy of ± 0.0236 ohms, ±0.01% span or ±0.07 F. References 2.8 & 2.9 perform a single point check of the Feedwater temperature input to the plant computer. In this check the only piece of M&TE used is the decade box, therefore this is the M&TE accuracy for this loop.

$MTE_{TEMP} = \pm 0.07$ F

3.6 CALIBRATION TOLERANCE vs. ACCURACY:

a. P-902/905 - From Data Sheet 8 of Reference 2.8 and 2.9, the calibration tolerance for the loop input to the ERF computer is 0.55% or ± 5.5 psi (1000psi x 0.55%). Because the calibration tolerance is larger than the device accuracy, the calibration tolerance will be used in lieu of accuracy (see 3.1).

$CAL_{PRES} = \pm 5.5$ psi

b. T-1396/1399 - From step 7.6.2 from references 2.6 and 2.7, there is no calibration tolerance for the loop input to the ERF computer. The step zeros out any error that exists at the 400F point. This analysis will conservatively assume a 0.5F error at this point. The accuracy for the RTD/Temperature transmitter combination is 0.62F. This is slightly greater than the calibration tolerance (0.62F > 0.5F), therefore the accuracy will be used to determine the overall loop uncertainty and the calibration tolerance will be set to 0.0 F(see 3.1).

$CAL_{TEMP} = \pm 0.0F$

4.0 METHOD OF CALCULATION

Attachment A provides the EM drawing for the P-902/905 channels used for Steam Generator pressure input to the ERF computer. Attachment B provides the EM drawing for the T-1396/1399 channels used for Feedwater temperature input to the ERF computer. As can be seen from these Attachments, the two pressure loops are functionally identical, and the two temperature loops are functionally identical. Also, a review of References 2.6/2.7 and 2.8/2.9 shows that the loops are calibrated the same.

The following provides instrument model numbers, ranges and a brief description of the functions in the loop. Only the instrumentation associated with channels P-902 and T-1396 is identified. As previously discussed, channels P-905 and T-1399 are the same. The instrument Model Numbers, Tag Numbers and ranges are obtained from Attachments A and B.

A/PT-902	FOXBORO Model N-E11GM-HIE2-AD 0 to 1000 psi input with 10 to 50 mA output
A/PQ-902	GEMAC Model 570-01 Loop Power Supply Provides loop power for PT-902
A/PM-902	TEC Model 156K Class 1E/Non Class 1E Isolator Provides isolated signal to the plant computer

TE-1396	ROSEMOUNT Model 0078 RTD 93 to 237.04 ohms/0 to 700 F
TM-1396	ROSEMOUNT Model 644R Temperature Transmitter matched to the RTD 93 to 237.04 ohms/4 to 20 mA signal Provides input to plant computer.

BODY OF CALCULATION

Inputs used in this section are obtained from Section 2.0 "REFERENCES" and/or Section 3.0 "ASSUMPTION AND GIVEN CONDITIONS" as indicated under "SOURCE".

5.1 DEVICE UNCERTAINTY FOR TRANSMITTER (DU_{TRAN})

						SOURCE
TAG		PT-902 & PT-905				
MFG		FOXBORO				
MODEL		N-E11GM				
RANGE	=	0.00	to	1000 psig		
OUTPUT	=	10.00	to	50 mADC		
TE_{TRAN}	= ±	7.50 psi				3.2.a
DR_{TRAN}	= ±	4.69 psi				3.2.b
RAD_{TRAN}	= ±	5.00 psi				3.2.g
PME_{TRAN}	= +	3.00 psi		(bias)		3.2.e

The device uncertainty for the transmitter (DU_{TRAN}) is calculated by combining the above error terms using the SRSS method:

DU_{TRAN} = ± 10.16 psi (+) 3 psi bias

Conservatively combining the above by adding the bias in both directions yields an error term of ± 13.16 psi.

DU_{TRAN} = ± 13.16 psi

5.2 DEVICE UNCERTAINTY FOR SIGNAL ISOLATOR

					SOURCE
TAG	A/PM-902 & A/PM-905				
MFG	TEC				
MODEL	156K				
INPUT	=	10.00	to	50 mADC	
OUTPUT	=	4.00	to	20 mADC	
DR_{ISOL}	= ±	3.20 psi			3.3.a

The device uncertainty for the signal isolator is calculated by combining the above error terms using the SRSS method:

DU_{ISOL} = ± 3.20 psi

5.3 DEVICE UNCERTAINTY FOR TEMPERATURE ELEMENT - TRANSMITTER

TAG	TE-1396 & TE-1399 / TT-1396 & TT-1399		
MFG	ROSEMOUNT		
MODEL	78 / 644H		
RANGE	=	0.00 to 700 F	

			SOURCE
AA_{TE}	= ±	0.62 F	3.4.a
SHE_{TE}	=	negligible	3.4.b
TE_{TM}	= ±	0.031 F	3.4.c
ST_{TE}	= ±	0.035 F	3.4.d

calculated by combining the above error terms using the SRSS method:

DU_{TE} = ± 0.62 F

5.4 CALCULATION

Inputs used in this section are obtained from Section 3.0 and/or Section 5.0.

Section 5.1 and 5.2 combines the individual instrument uncertainties for the loop components using the SRSS methodology. These uncertainties are combined below to determine an overall loop uncertainty. The loop uncertainty consists of the individual component uncertainties in the loop combined with the M&TE uncertainty and the setting tolerance where the tolerance is greater than the components accuracy. The following equations are used.

5.4.1 For the Steam Generator pressure uncertainty, combining DU_{TRAN}, DU_{ISOL}, MTE_{PRES} and CAL_{PRES} using SRSS, the loop uncertainty is:

DU_{TRAN}	= ±	13.16 psi		5.1
DU_{ISOL}	= ±	3.20 psi		5.2
MTE_{PRES}	= ±	1.30 psi		3.5.a
CAL_{PRES}	= ±	5.50 psi		3.6.a
CMPTR A/D	= ±	0.430 psi	(1000psi x 0.043%)	3.1.c

Combining the above terms:

SG pres	= ±	14.68 psi

5.4.2

For the Feedwater Temperature uncertainty, combining Dut and MTEt using SRSS, the loop uncertainty is:

DU_{TE}	= ±	0.62 F		5.3
MTE_{TEMP}	= ±	0.07 F		3.5.b
CAL_{TEMP}	= ±	0.00 F		3.6.b
CMPTR A/D	= ±	0.301 F	(700F x 0.043%)	3.1.c

Combining the above terms:

FW temp = ± 0.69 F

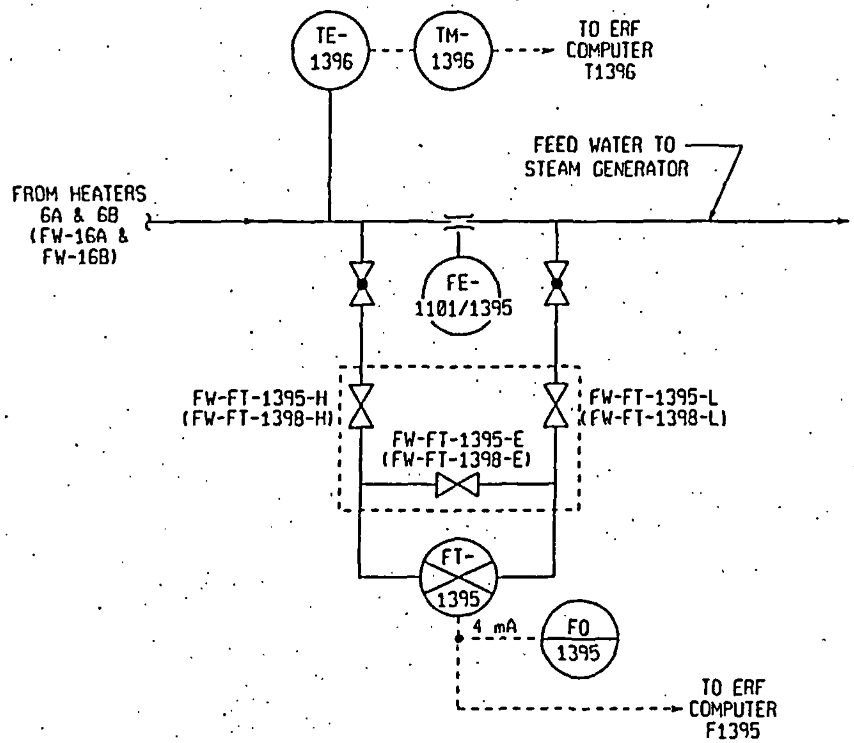
6.0 CONCLUSIONS

The Total Loop Uncertainty (TLU) for the Steam Generator Pressure and Feedwater Temperature instruments which provide input to the plant calorimetric calculation are as follows:

P-902/905 = ± 14.68 psia

T-1396/1399 = ± 0.69 F

ATTACHMENT A



STEAM GEN. No.	CHANNEL No.	
	FLOW	TEMP.
RC-2A	1101/1395	1396
RC-2B	1102/1398	1399

TAG NUMBER	SPEC. SH.	FUNCTION/ DESCRIPTION	VENDOR/ MANUFACTURER	P.O. NUMBER	LOCATION	SET POINT
1395		F.W. FLOW TO STEAM GEN. • RC-2A				
FE-1101/ 1395	1.12	FLOW ELEMENT	G.E.	762		
FT-1395	2.19 .19	FLOW TRANSMITTER	ROSEMOUNT	762	SW GR 1011'	
F1395		ERF COMPUTER POINT				
FO-1395		POWER SUPPLY FOR FT-1395	CHROMALOX		SW GR 1011'	
1396		TEMP. OF F.W. FLOW TO RC-2A				
TE-1396	1.82	TEMP. ELEMENT	G.E.	762		
TM-1396		RESISTANCE TO CURRENT CONVERTER	YOKOGAWA		INS. AI-12	
T1396		ERF COMPUTER POINT				

REF. DRAWING
1. 11405-M-253, SH.1 & 3..........P & ID
2. 136B2331, SH.79 & 79A.........LOOP DIAG
3. 11405-M-54, SH.15,21,30 & 49...INSTR DET

FORT CALHOUN STATION

INSTRUMENT AND CONTROL EQUIPMENT LIST

DWG. EM-1395/1399, SH.1
REV.SH. 2676
FILE 15876
APVD [illegible] 10/17/2000
REV 15

29/42

FOR SKETCH REFER TO
CHDR DWG.11405-EM-1395/1399
SHEET 1
GSE FILE NUMBER 15876

TAG NUMBER	SPEC. SH.	FUNCTION/ DESCRIPTION	VENDOR/ MANUFACTURER	P.O. NUMBER	LOCATION	SET POINT
1398		F.W. FLOW TO STEAM GEN. • RC-2B				
FE-1102/ 1398	1.12	FLOW ELEMENT	G.E.	762	ROOM 81	
FT-1398	2.19 .19	FLOW TRANSMITTER	ROSEMOUNT	762	SW GR 1011'	
F1398		ERF COMPUTER POINT				
FO-1398		POWER SUPPLY FOR FT-1398	CHROMALOX		SW GR 1011'	
1399		TEMP OF F.W. FLOW TO RC-2B				
TE-1399	1.82	TEMP. ELEMENT	G.E.	762		
TM-1399		RESISTANCE TO CURRENT CONVERTER	YOKOGAWA		INS. AI-12	
T1399		ERF COMPUTER POINT				

REF.DWGS.:
1. 13682331, SH.79.....LOOP DIAG
2. 11405-M-253, SH.3...P & ID

FORT CALHOUN STATION

INSTRUMENT AND CONTROL EQUIPMENT LIST

DWG. EM-1395/1399, SH.2

REV.SH. 3976

FILE 15877

APVD [illegible]
REV 6

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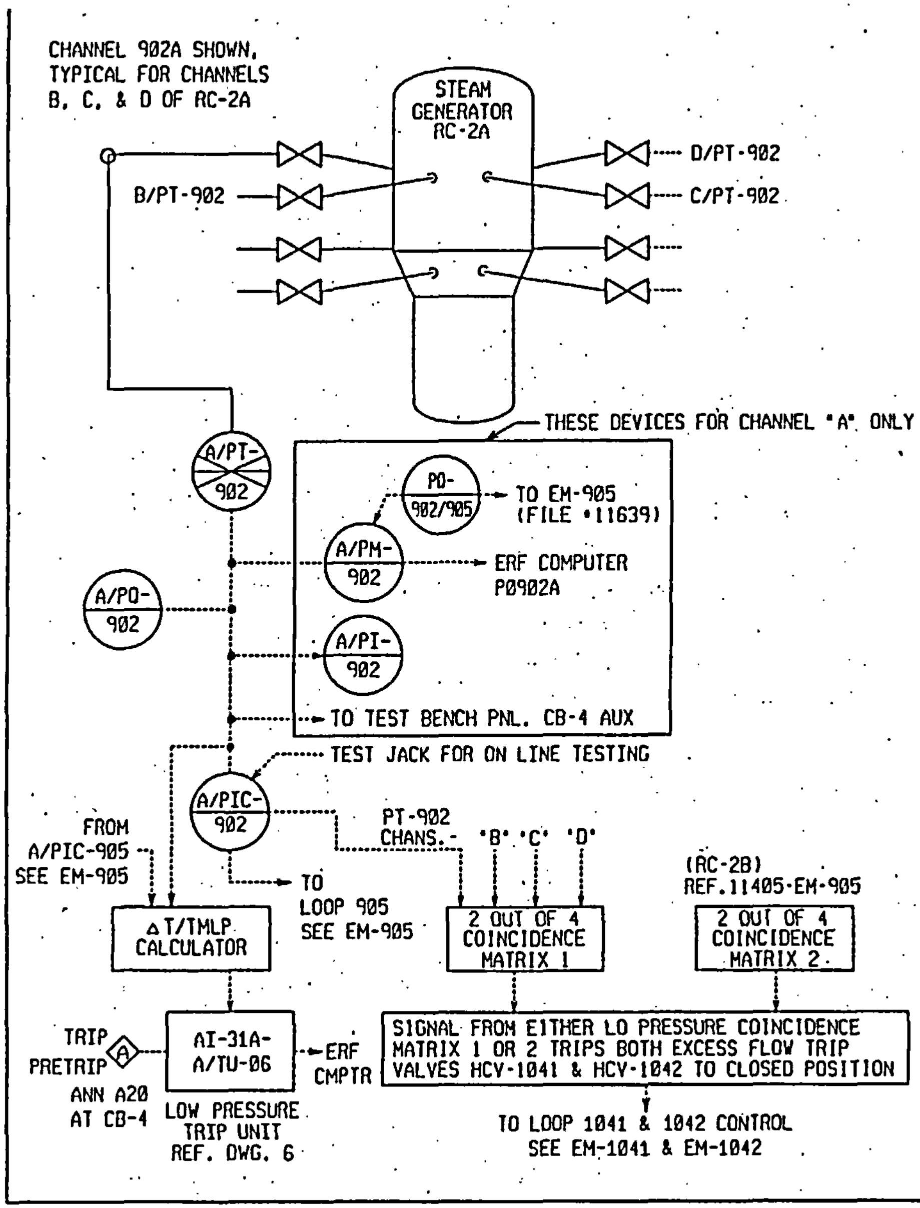
ATTACHMENT B



TAG NUMBER	SPEC. SH.	FUNCTION/ DESCRIPTION	VENDOR/ MANUFACTURER	P.O. NUMBER	LOCATION	SET POINT
902		BRD. MTD. INDICATOR CONTROL FOR RC-2A STEAM PRESSURE				
A/PT-902	252	PRESSURE TRANSMITTER	FOXBORO N-E11GM-HIE2-AD		CONT. AI-135A	
A/PI-902	5.23	PRESSURE INDICATOR	G.E.		AI-179	
A/PQ-902	254	UNIT OF MULTIPLE POWER SUPPLY A/Q-2	G.E. / MAC 570-01		CB-1,2,3	
A/PIC-902	637	PRESSURE INDICATOR & CONTROLLER	DIXSON 8101	762	CB-4	
A/PM-902	52.2	CLASS 1E/ NON CLASS 1E ISOLATOR	TEC MOD. 156K		CB-4	
P0902A		ERF COMPUTER INPUT				
P902A		ERF COMPUTER INPUT				
B/PT-902	252	PRESSURE TRANSMITTER	FOXBORO N-E11GM-HIE2-AD		CONT. AI-127B	
B/PQ-902	254	UNIT OF MULTIPLE POWER SUPPLY B/Q-2	G.E. / MAC 570-01		CB-1,2,3	
B/PIC-902	637	PRESSURE INDICATOR & CONTROLLER	DIXSON 8101		CB-4	
PQ-902/905		POWER SUPPLY	ACTION INSTRUMENTS T609		CB-4	
P902B		ERF COMPUTER INPUT				

REF. DRAWINGS:
1. 11405-M-253, SH. 1 P&ID
2. 161F561 SH.122,123,124,& 125 INTERCONNECTION
3. 11405-M-54, SH. 16
4. 11405-M-54, SH. 15
5. 11405-M-54, SH. 35
6. E-23866-411-302

CQE
NUCLEAR SAFETY RELATED
EQUIPMENT IS SHOWN ON THIS DRAWING.

FORT CALHOUN STATION

INSTRUMENT AND CONTROL EQUIPMENT LIST

DWG. EM-902, SH. 1
REV. SH. 4813
FILE 15719
APVD 8/4/93
REV 14

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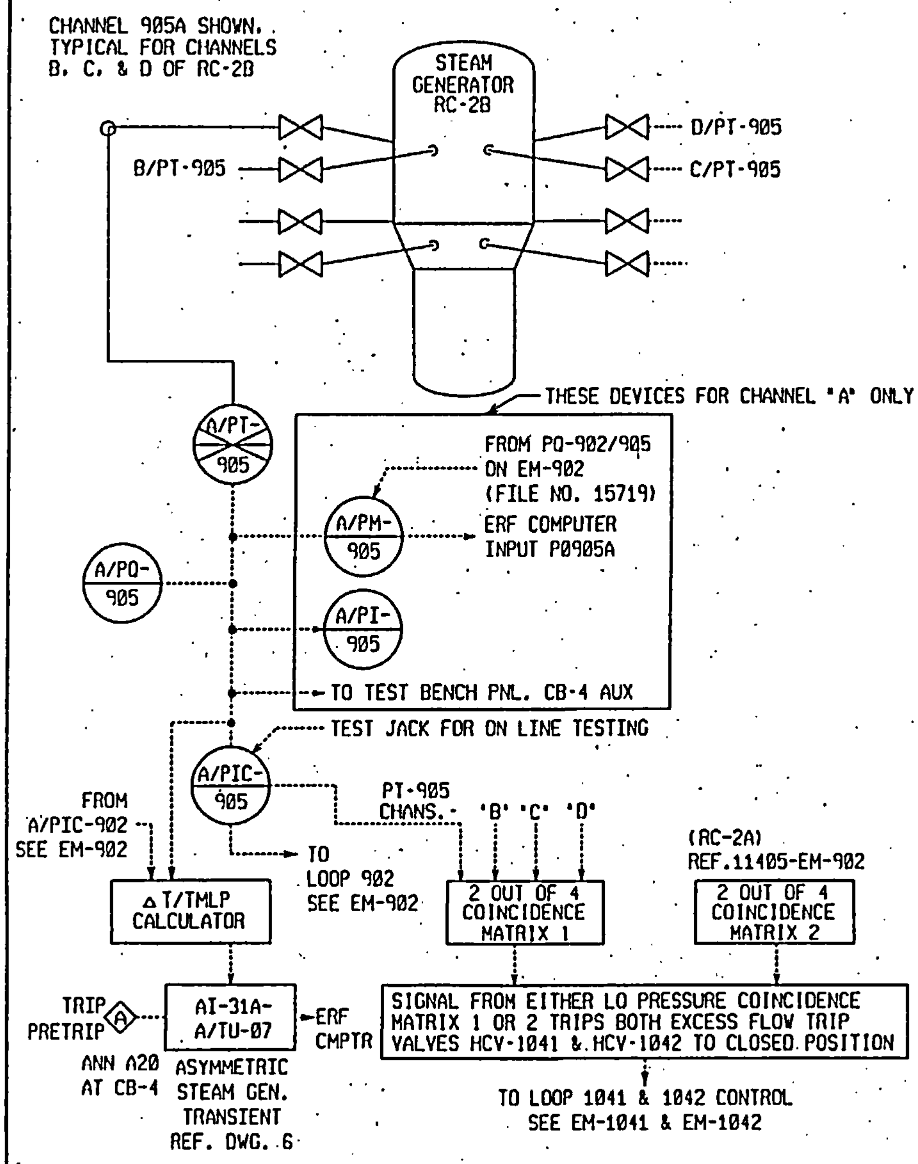
ATTACHMENT C



TAG NUMBER	SPEC. SH.	FUNCTION/ DESCRIPTION	VENDOR/ MANUFACTURER	P.O. NUMBER	LOCATION	SET POINT
905		BRD.MTD.INDICATOR CONTROL FOR RC-2B STEAM PRESSURE				
A/PT-905	252	PRESSURE TRANSMITTER	FOXBORO N-E11GM-HIE2-AD		CONT. AI-136A	
A/PI-905	5.23	PRESSURE INDICATOR	G.E.		AI-179	
A/PQ-905	254	UNIT OF MULTIPLE POWER SUPPLY A/Q-2	G.E. / MAC 570-01		CB-1,2,3	
A/PIC-905	637	PRESSURE INDICATOR & CONTROLLER	DIXSON B101		CB-4	
A/PM-905	52.2	CLASS 1E/ NON CLASS 1E ISOLATORS	TEC MODEL 156K		CB-4	
P0905A		ERF COMPUTER INPUT				
ASGTA		ERF COMPUTER INPUT				
B/PT-905	252	PRESSURE TRANSMITTER	FOXBORO N-E11GM-HIE2-AD		CONT. AI-136B	
B/PQ-905	254	UNIT OF MULTIPLE POWER SUPPLY B/Q-2	G.E. / MAC 570-01		CB-1,2,3	
B/PIC-905	637	PRESSURE INDICATOR & CONTROLLER	DIXSON B101		CB-4	
ASGTB		ERF COMPUTER INPUT				

REF. DWGS.:
1. 11405-M-253, SH. 1 P&ID
2. 161F561 SH. 122,123,124 & 125 INTERCONNECTION
3. 11405-M-54 SH. 16
4. 11405-M-54 SH. 15
5. 11405-M-54 SH. 36
6. E-23866-411-302

CQE
NUCLEAR SAFETY RELATED
EQUIPMENT IS SHOWN ON THIS DRAWING

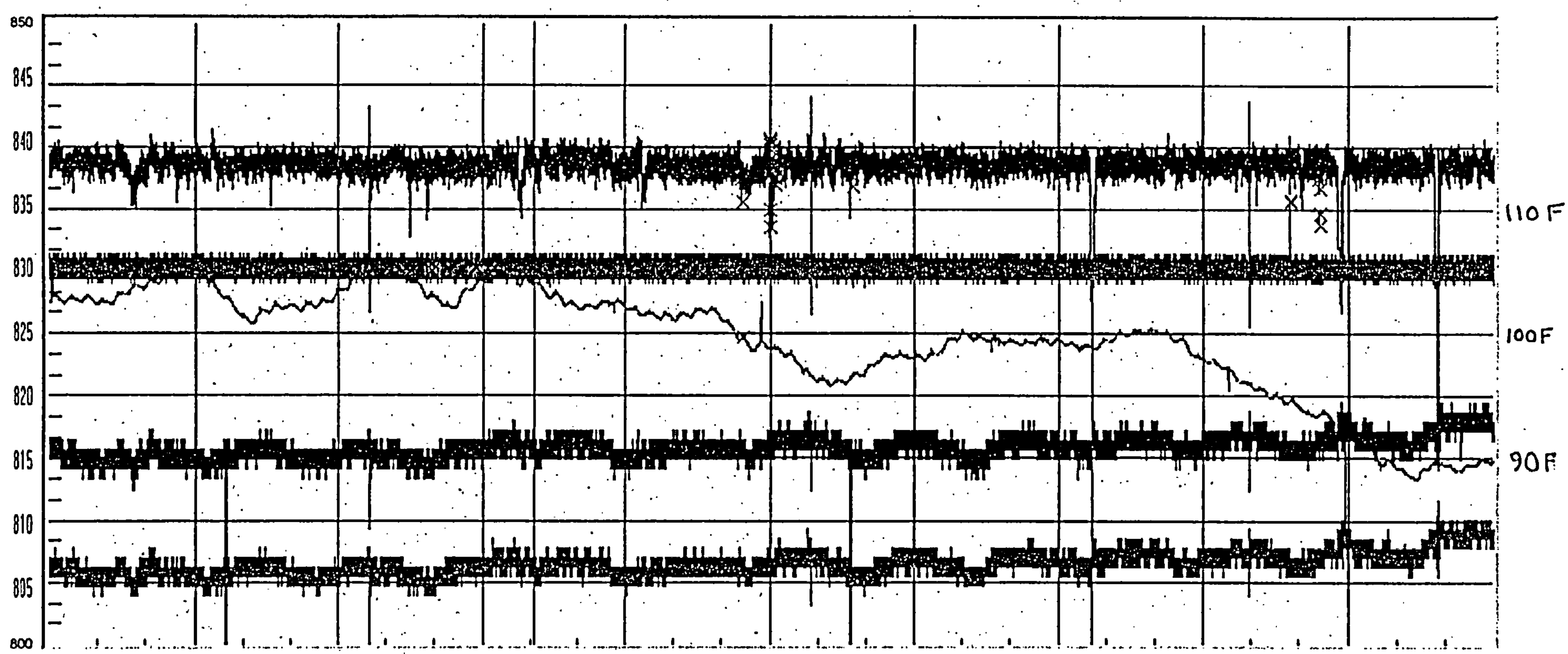
FORT CALHOUN STATION

INSTRUMENT AND CONTROL
EQUIPMENT LIST

DWG. EM-905, SH. 1
REV.SH. 4014
FILE 11639
APVD [illegible] 8/4/93
REV 14

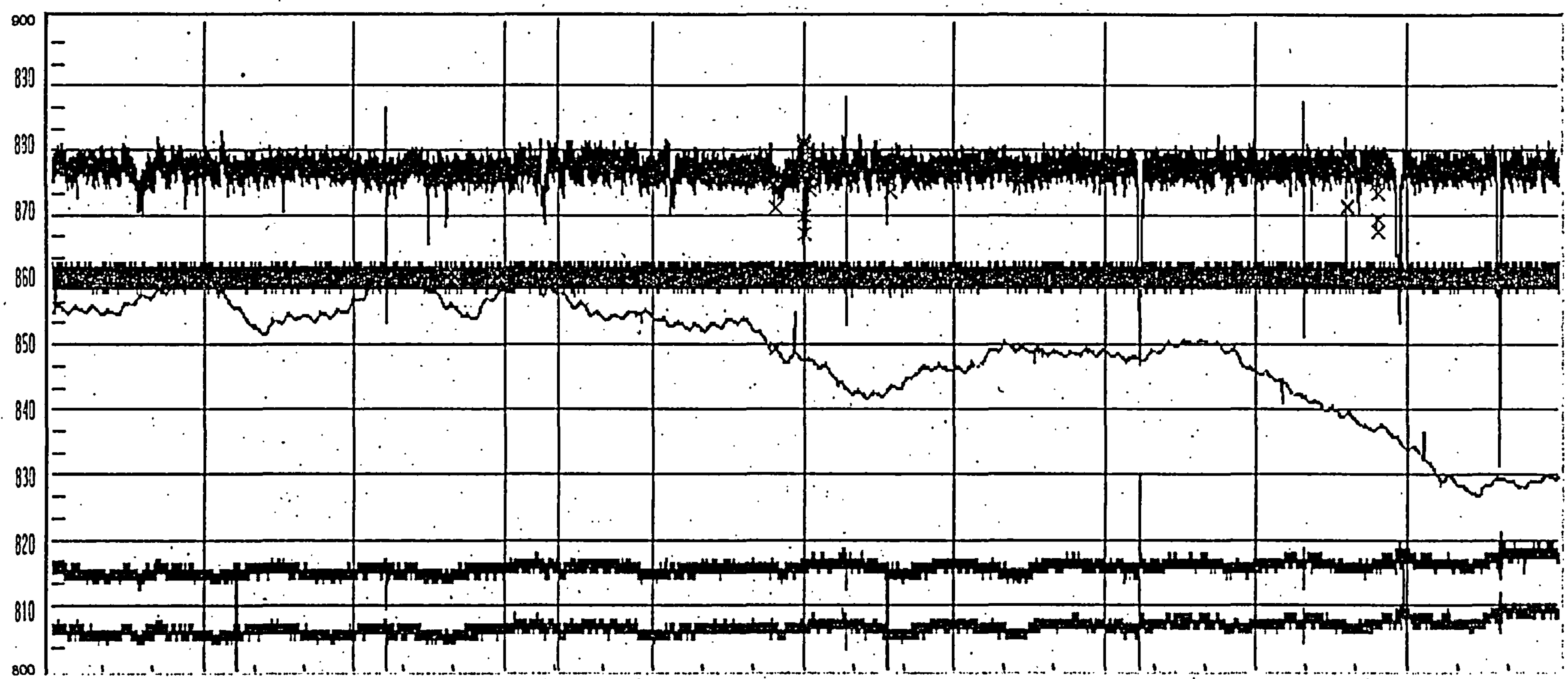
34/42

ATTACHMENT D



7/1/2002 10:34:25.843 AM 92 2:26:05 672 10/1/2002 1:00:31.515 PM

Name	Server	Description	Value	Status	Plot Min.	Plot Max.	Aggregate	Eng. Units	Map
P0905A	fcspds	STM GEN 2 STM PRESS CH A	808.594	Good	800	850		PSIA	HG2_ANALOG...
TA112C	fcspds	LOOP 1A COLD LEG TEMP	541	Good	450	600		DEG F	HG1_ANALOG...
XC105	fcspds	XC105 REACTOR PWR 10 M...	1498.02	Good	1460	1510		MW	HG1_ANALOG...
P0902A	fcspds	STM GEN 1 STM PRESS CH A	817.969	Good	800	850		PSIA	HG2_ANALOG...
T888	fcspds	CONTAINMENT AMBIENT TE..	89.6875	Good	75	125		DEG F	HG4_ANALOG...



	Name	Server	Description	Value	Status	Plot Min.	Plot Max.	Aggregate	Eng. Units	Map
	P0905A	fcspds	STM GEN 2 STM PRESS CH A	808 594	Good	800	900		PSIA	HG2_ANALOG...
	TA112C	fcspds	LOOP 1A COLD LEG TEMP	541	Good	450	600		DEG F	HG1_ANALOG...
	XC105	fcspds	XC105 REACTOR PWR 10 M...	1498 02	Good	1460	1510		MW	HG1_ANALOG...
	P0902A	fcspds	STM GEN 1 STM PRESS CH A	817.969	Good	800	900		PSIA	HG2_ANALOG...
	T888	fcspds	CONTAINMENT AMBIENT TE...	89.6875	Good	75	125		DEG F	HG4_ANALOG...

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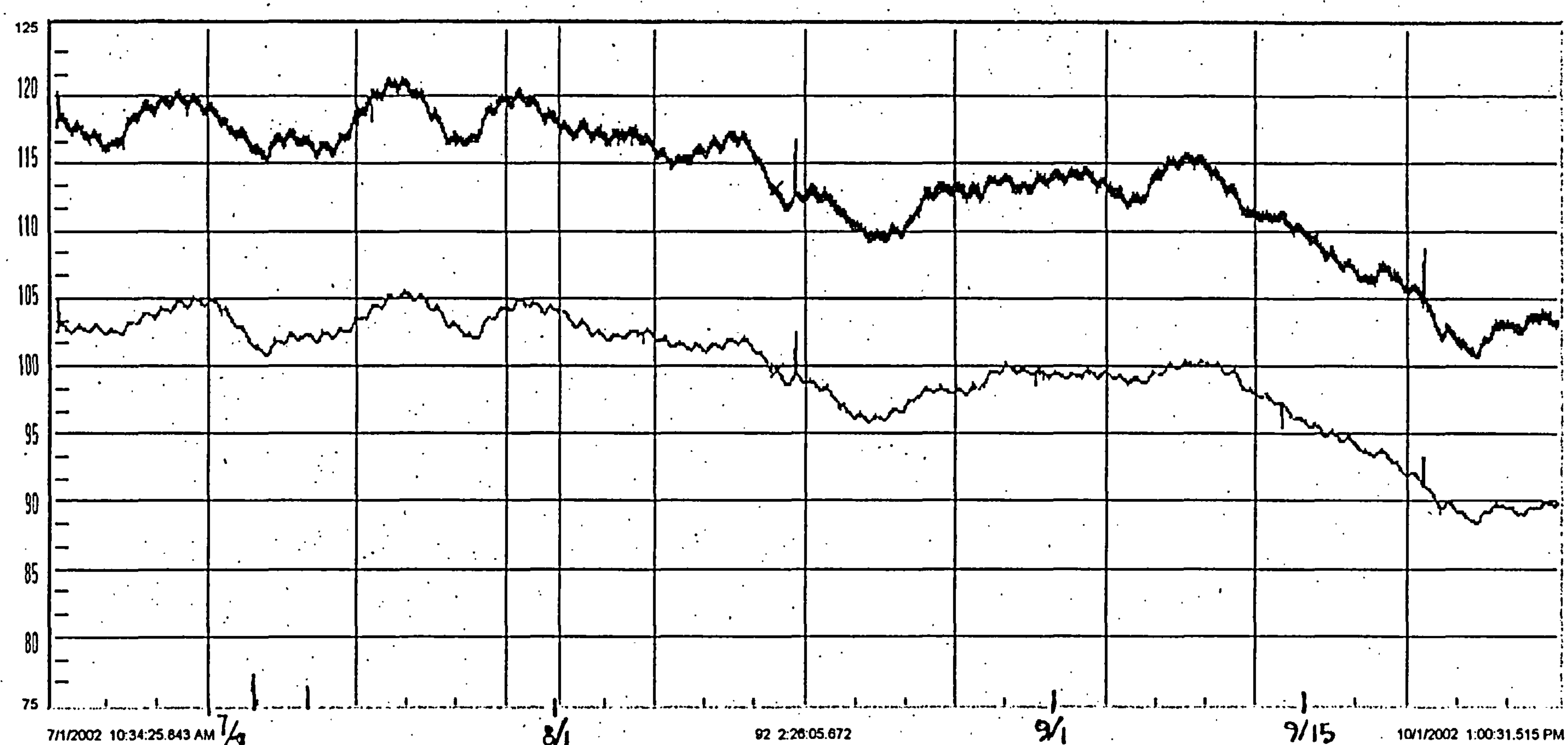

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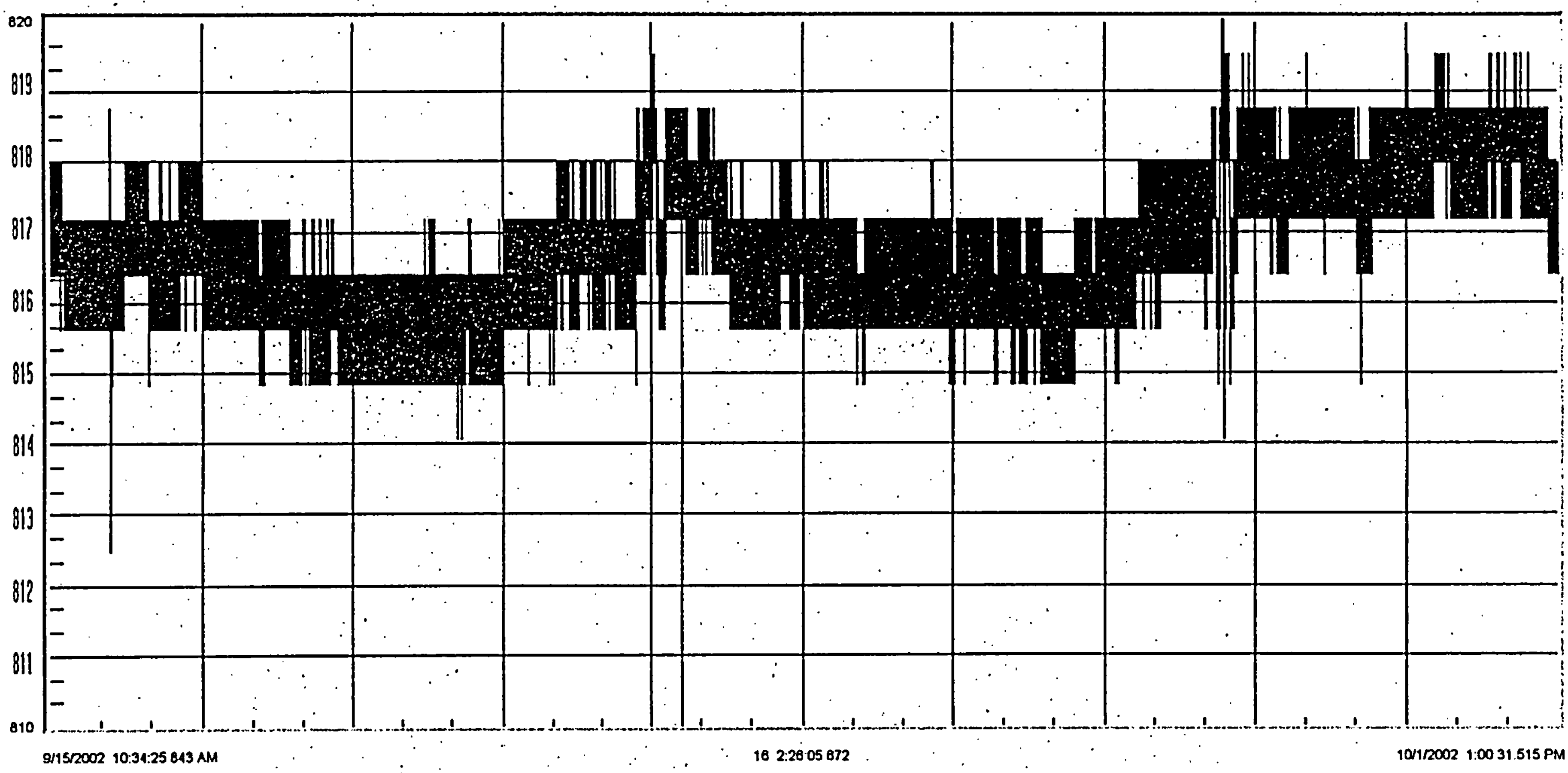
	P0902A	fcspds	STM GEN 1 STM PRESS CH A	[illegible]	[illegible]	0	1000		PSIA	HG2_ANALOG
	T890	fcspds	CONTAINMENT AMBIENT TE..	103.281	Good	75	125		DEG F	HG4_ANALOG...
	T888	fcspds	CONTAINMENT AMBIENT TE..	89 6875	Good	75	125		DEG F	HG4_ANALOG...

9/15 - 10/1 = $\Delta P / \Delta T$ = 1.5 / 6.66 F

7/9 - 7/10 $\Delta P / \Delta T$ = 0 / 4 F

~~39~~ 38/42



Name	Server	Description	Value	Status	Plot Min.	Plot Max.	Aggregate	Eng. Units	Map
B0902A[illegible]	[illegible]	STM GEN 1 ST	817.969	Good	810	820		PSIA	HG2 ANALOG

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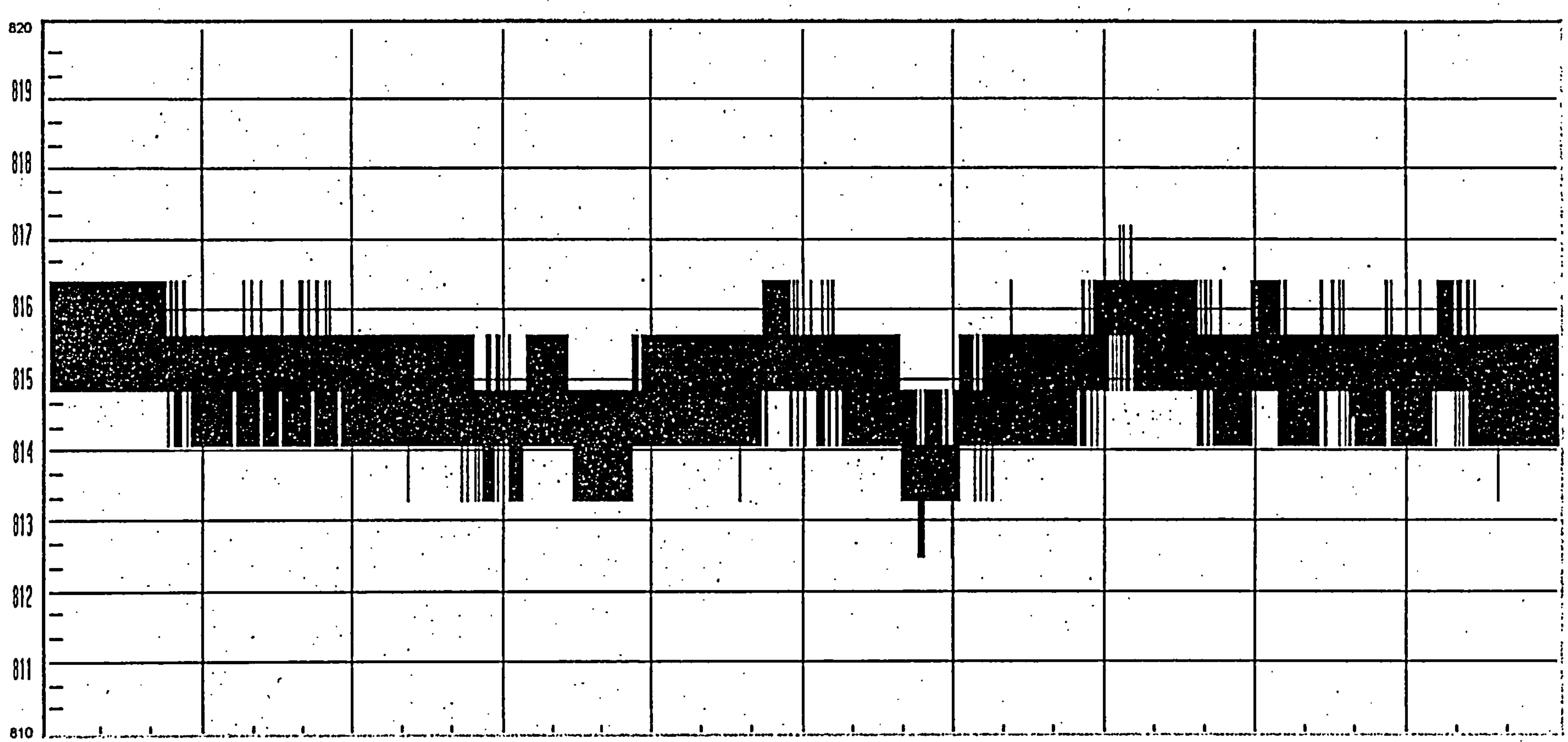

7/1/2002 10:34:25 843 AM 9 2:26 05.672 7/10/2002 1:00:31.515 PM

	Name	Server	Description	Value	Status	Plot Min.	Plot Max.	Aggregate	Eng. Units	Map
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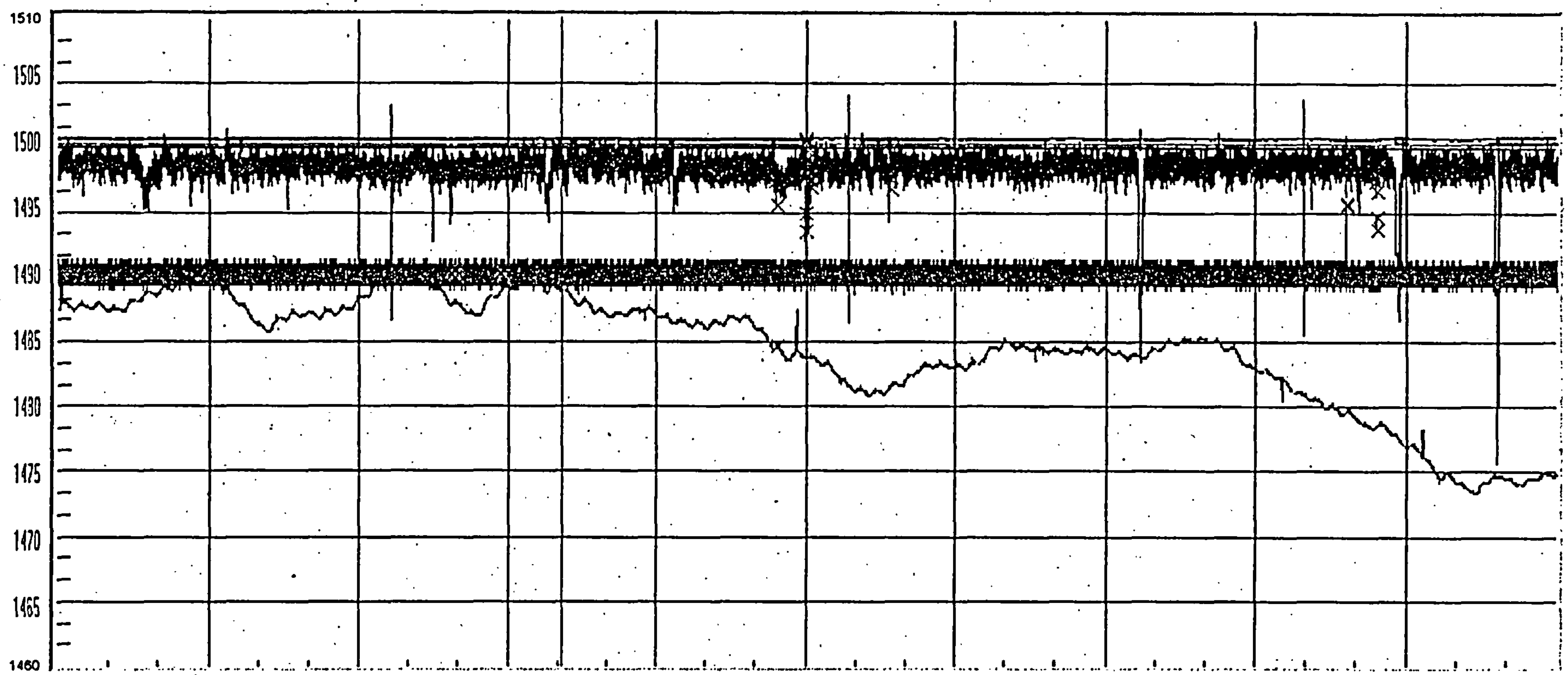

7/1/2002 10:34:25.843 AM 92 2:26:05.672 10/1/2002 1:00:31.515 PM

Name	Server	Description	Value	Status	Plot Min.	Plot Max.	Aggregate	Eng. Units	Map
P0905A	fcspds	STM GEN 2 STM PRESS CH A	808.594	Good	0	1000		PSIA	HG2_ANALOG...
TA112C	fcspds	LOOP 1A COLD LEG TEMP	541	Good	450	600		DEG F	HG1_ANALOG...
XC105	fcspds	XC105 REACTOR PWR 10 M...	1498.027	Good	1460	1510		MW	HG1_ANALOG...
P0902A	fcspds	STM GEN 1 STM PRESS CH A	817.969	Good	0	1000		PSIA	HG2_ANALOG...
T888	fcspds	CONTAINMENT AMBIENT TE..	89.6875	Good	75	125		DEG F	HG4_ANALOG...

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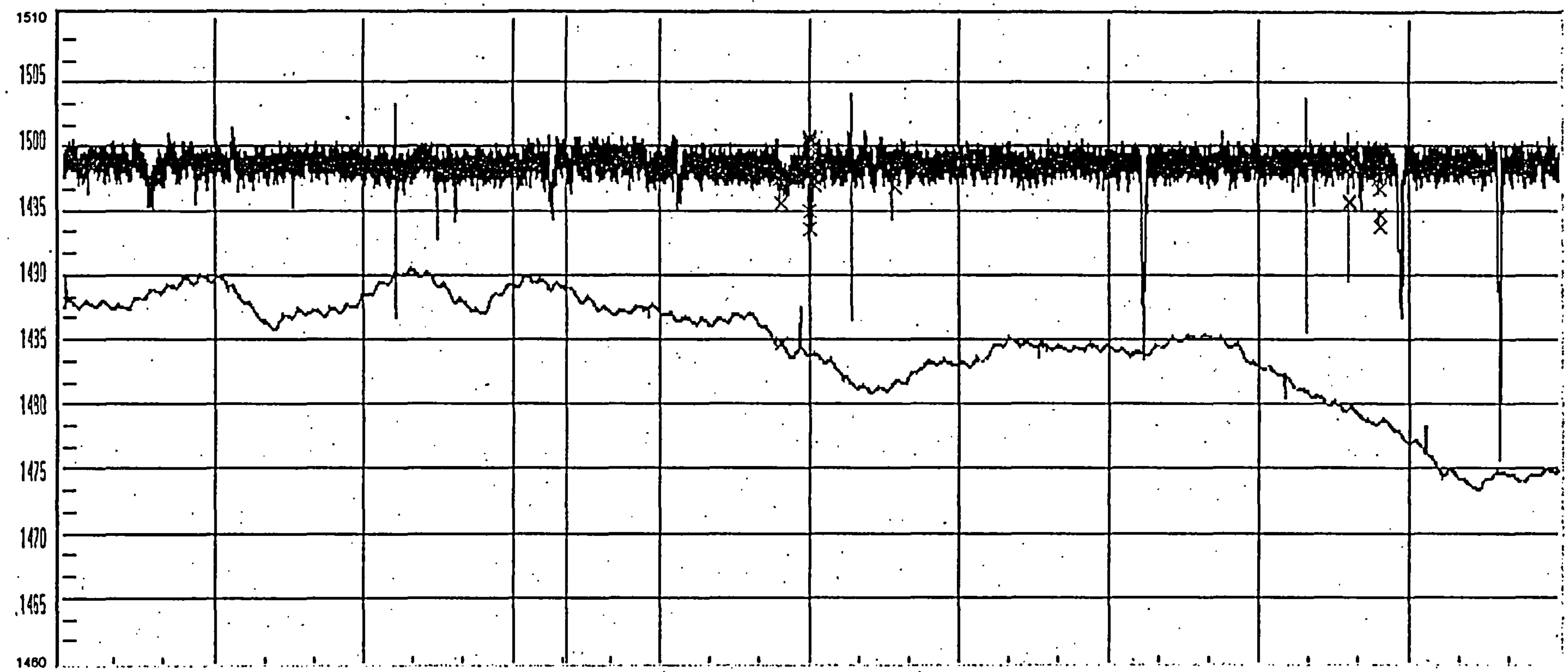

Name	Server	Description	Value	Status	Plot Min.	Plot Max.	Aggregate	Eng. Units	Map
P0905A	fcspds	STM GEN 2 STM PRESS CH A	•••••	-1	0	1000		PSIA	HG2_ANALOG...
TA112C	fcspds	LOOP 1A COLD LEG TEMP	•••••	-1	450	600		DEG F	HG1_ANALOG...
XC105	fcspds	XC105 REACTOR PWR 10 M...	1498.02	Good	1460	1510		MW	HG1_ANALOG...
P0902A	fcspds	STM GEN 1 STM PRESS CH A	•••••	-1	0	1000		PSIA	HG2_ANALOG...
T888	fcspds	CONTAINMENT AMBIENT TE...	89 6875	Good	75	125		DEG F	HG4_ANALOG...

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LIC-03-0148
Attachment 3
"Independent Check of Calculations" associated with the Calorimetric Uncertainty Evaluation

The following documents an independent check that was performed to verify the spreadsheet calculations made in this analysis. This check verifies the numbers which are calculated in the spreadsheets and the correctness of the spreadsheet equations. Calculation checks using flow meter uncertainty were made using a value of 0.4%. This verified the accuracy of the calculations.

PRODUCTION ENGINEERING CALCULATION SHEET

PROJECT MUR REVISION No

SUBJECT Equations 5.2 PAGE No OF

5.2.1

XC10 =(XC088-XC086)((1-X1)hf +(X1)hg1)-(XC098)hc +(XC096)hB1

XC101= (3351.105938-24.71798522)((1-0.9975)(311.555)+(0.9975)1197.1))-(3351.105938)419.457+(24.71798522)487.938

XC10 = 2599403.372 BTU/HR

XC089= K F1 $\times$((FA1 $\times$T1396 +C)$\times$(F1395/FWSCMP))$^{1/2}$

XC099= 24.062478$\times$(.1987$\times10^{-5}$ 440+0.99896)$\times\left(\frac{366.5}{0.01917}\right)^{1/2}$ = 3351.105938

XC096 = KF3$\times$((FA1$\times$T1391 +C1)$\times\left(\frac{F1392}{FCMP2}\right)^{1/2}$

XC096 = 1.6995736$\times$(1.4987$\times10^{-5}$ 500 +0.99886 $\times\left(\frac{4.25}{(0.02048)}\right)^{1/2}$ = 24.71798522

XC1102= (XC089-XC097)((1-X2)(hf2 +(X2)hg2-(XC089)hc2+(XC097)hB2

XC102 = (3351.105938-24.71798522)((1-0.9983)542.555+0.9983(1200.44))-(3351.105938)419.457+(24.71798522)487.938

XC102 = 2591233.032 BTU/HR

XC099 = XC088

XC097 = XC086

XC105=XC101+XC102+XC104-K0014

XC105 = 2589403.372+2591233.032+4934-2526 = 5160309.404

PRODUCTION ENGINEERING CALCULATION SHEET

PROJECT MUR REVISION No

SUBJECT VV PAGE No 2 OF

5.6.3.1.2.1 3.3.1

$FW_2 = \frac{FW_1}{\rho_1} \times \rho_2 = \frac{3351.10593}{52.1648409} \cdot 52.63157895 = \underline{3381.089517}$

$\rho = 50$

* CHECK out

$W_3 = \frac{FW_2}{\rho_1} = \frac{3381.089517}{52.63157} \cdot 51.813475 = 3328.53372$

5.3.13 $dp_{FW}/dT_{FW} = \frac{d\rho}{dT} = \left(\frac{51.8135 - 52.6316}{(450 - 430)}\right) = -0.040905$

* SEE P: 59 FOR ADDITIONAL 5.3.13 VERIFICATION TM 10/21/03

5.3.4 B ldn Temp 500 already calculated page 1 line 10-11

B ldn Temp 490

$1.699573C(1.8987\times10^{-5} \cdot 490 + 0.99886)\times\left(\frac{4.25}{.0202}\right)^{1/2} = 24.95362737$

Bldn Temp 510

$.699573C(1.8987\times10^{-5} \cdot 510 + 0.99886)\times\left(\frac{4.25}{0.0202}\right)^{1/2} = 24.56087572$

5.4

using 5.3.5 $(3384.616997 - 24.71798522)((1-0.9975)(511.555) + 0.9975(1199.11))$
$-(3384.616997)(419.457) + (24.71798522)487.938 + (3384.616997$
$-24.71798522)((1-0.9983)511.555 + 0.9983(1199.11))$
$-(3394.616997)419.457 + (24.71798522)487.938 + 4934 - 25261$
$= 5212466.629$

$(3317.594878 - 24.71798522)((1-0.9975)(511.555) + 0.9975(1199.11))$
$-(3397.594878(419.457)) + (24.71798522)487.938 + (3317.594878 -$
$24.71798522)((1-0.9983)511.555 + 0.9983(1199.11)) - 3317.594878 \cdot 419.45$
$+(24.71798522) \cdot 487.938 + 4934 - 25261 = 5108152.178$

5.4.2 T_{AFW} 450: 5105832.939 ... 430: 5220813.113

5.4.3 T_{BFW} 5105920.323 ... 5220829.606

5.4.4 PA_{SG} 860: 5155535.963 ... 760: 5164694.93527

5.4.5 PB_{SG} 5155509.375 ... 5164675.928

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5.4.9 $A = \frac{5160133.5 - 5160484.902}{0.02} = -17550.$

$B =$ Same as 5.4.5

$C = 1.8757$

$D =$ D of 5.4.8

$(C \cdot D) = (C \cdot D)$ of 5.

$(A^2 \cdot B^2 \cdot C^2 \cdot D^2) = 6.64654 \times 10^{-4}$

5.4.10 $A = \frac{5160714.3 - 5159919.974}{20} = 39.7163$

$B = \frac{500}{5160307.404} = 9.6895 \times 10^{-5}$

$C = 2.94$

$D = \frac{1}{600} \cdot 100$

$(C \cdot D) = 0.588$

$(A \cdot B \cdot C \cdot D)^2 = 5.2008 \times 10^{-6}$

5.4.11 $A = \frac{5160714.4 - 5159919.899}{20} = 39.72505$

$B =$ B of 5.4.10

$C = 2.94$

$D =$ D of 5.4.10

$C \cdot D = C \cdot D$ of 5.4.10

$(A \cdot B \cdot C \cdot D)^2 = 5.12234 \times 10^{-6}$

$U_{1.96}$ Confidence interval $1.96\sqrt{0.0411228773z} = 0.3220480766889$

$U_{1.645} = \frac{U_{1.96}}{1.96} \cdot 1.645 = 0.2708213557925$

$2 - U_{1.645} = 1.7291786420$

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5.4.1 $\frac{521246.6 - 510815.2 - 174}{0.02} = 52$ 5721.1

$B = \frac{1}{516309.404} = 9.937.1 \times 10^{-7}$

$C = 0.2\ 99.6\%$

$D = .00$

$(C \cdot D) = \cancel{0.2846}$ 0.3922

$(A \cdot B \cdot C \cdot D)^2 = (52\ 572 \cdot (1.93.7.1 \times 10^{-7}) \cdot \cancel{0.2846}\ 0.3922\ ))^2 = \cancel{0.04564179}$ = .15715

5.4.2 $\frac{510593.2.9 - 5220913.113}{20} = -5749.0\ 065$ TKL 10/27/03

$B = \frac{440}{516309.404} = 8.5266 \times 10^{-5}$

$C = 0.69$

$D = \frac{1}{440} \times 100 = 0.2272727$

$(C \cdot D) = 0.\ 568\ 8\ 63$

$(A \cdot B \cdot C \cdot D) = 0.005909228$

5.4.3 $\frac{5105920.5 - 5220429.605}{20} = -5750.45525$

B = B of 5.4.2

C = 0.69

D = D of 5.4.2

C·D = C·D of 5.4.2

$(A\ B\ C\ D)^2 = 0.005912\ 98$

5.4.4 $\frac{515553.6 - 5164648.835}{100} = -91.12835$

$B = \frac{810}{516309.404} = 1.5697 \times 10^{-4}$

$C = 4.68$

$D = \frac{1}{810} \cdot 100 = 0.1234568$

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5.4.4 $(C \cdot D) = (14.68)(0.1234569) = 1.8\ 234524$

$(A \cdot B \cdot C \cdot D)^2 = 6.72082 \times 10^{-4}$

5.4.5 $A = \frac{5155504.4 - 5164675.127}{1.00} = -1\ 66527$

$B = B$ of 5.4.4

$C = 14.69$

$D = D$ of 5.4.4

$(C \cdot D) = (C \cdot D)$ of 5.4.4

$(A \cdot B \cdot C \cdot D)^2 = 6.80025 \times 10^{-4}$

5.4.6 $A = \frac{5160023.5 - 5160595.288}{0.10} = -5717.98$

$B = \frac{1}{5160309.404} = 1.9379 \times 10^{-7}$

$C = 0.13$

$D = \frac{0.1}{0.25} \cdot 100 = 400$

$(C \cdot D) = 52$

$(A \cdot B \cdot C \cdot D)^2 = 0.00332$

5.4.7 $A = \frac{5160115 - 5160503.805}{0.10} = -3888.05$

$B = B$ of 5.4.6

$C = 0.05$

$D = \frac{1}{0.17} \cdot 100 = 588.23529$

$C \cdot D = 29.4117645$

$(A \cdot B \cdot C \cdot D)^2 = (-3888.05 \cdot 1.9379 \cdot 10^{-7} \cdot 0.05 \cdot 588.23529)^2 = 4.9310 98 \times 10^{-4}$

5.4.8 $A = \frac{5160134 - 5160484.766}{0.02} = -17538.3$

$B = \frac{1}{5160309.404} = 1.9379 \times 10^{-7}$

$C = 1.8737$

$D = \frac{1}{24.717985} \cdot 100 = 4.0456372$

$(C \cdot D) = 7.58031 0421$

$(A \cdot B \cdot C \cdot D)^2 = 6.637609 \times 10^{-4}$

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5.4.6	XC105 (M20A)	1.05	0.95
		5160023.519	5160595.289
5.4.7	XC105 (MC0B)	5160115.00283	5160503.806
		1.01	0.99
5.4.8	WA_{BD}	5160134.042	5160484.767
5.4.9	WB_{BD}	5160133.906	5160484.902
5.4.10	TA_{BD}	5160714.27	5159911.974
5.4.	TB_{BD}	5160714.357	5159919.899
		1.01	0.99
5.5.1		5160056.794	5160562.014
5.5.2		5160358.744	5160260.064

dXC105/dW_{FW}

5.4.1 $= \dfrac{\left(\dfrac{5212466.029 - 5108152.178}{0.02}\right)}{5160309.04} = 1.01073 8329$

u W given

5.3.13 $uW_{FW}\% = \left(.4^2 - \left(\dfrac{0.345025765 \cdot 100}{440}\right)^2\right)^{1/2} = 0.3922$ ~~$\left(\cancel{300}^2 - \left(\dfrac{\cancel{0.35025765} \cdot 100}{440}\right)^2\right)^{1/2} = 2845705374$~~

$\left(\dfrac{dXC105}{dXW_{FW}} \cdot uF_{FW}\right)^2 = 0.157173044$ ~~0.0856671606~~

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$u\Gamma_{FW} = \dfrac{0.74 \cdot 100}{440} = 0.16818 \quad 1.8$

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Population

$$\sigma^2 = \frac{1}{N}\sum_{i=1}^{N}(x_i - \mu)^2$$

$$\mu = \frac{1}{N}\sum_{i=1}^{N} x_i$$

$$\mu = \frac{1}{3}(0.29 + 0.27 + 0.19) = 0.25$$

$$\sigma^2 = \frac{1}{3}\left((0.29-0.25)^2 + (0.27-0.25)^2 + (0.19-0.25)^2\right) = 0.001867$$

$$\sigma = \sqrt{0.001867} = \underline{0.0432}$$

$$\mu = \frac{1}{3}\sum_{i=1}^{N} x_i = \frac{1}{3}(0.15 + 0.17 + 0.20) = 0.1733$$

$$\sigma^2 = \frac{1}{3}\left((0.15-0.1733)^2 + (0.17-0.1733)^2 + (0.20-0.1733)^2\right) = 4.222\times10^{-4}$$

$$\sigma = \sqrt{4.222\times10^{-4}} = 0.0205$$