ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-13)

LIST OF AFFECTED PAGES

Unit 1

3/4 2-16

Unit 2

3/4 2-14

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TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Total Flow RATE	$\geq$ ~~378400~~ 375000 gpm#

R142

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

\# Includes a ~~3.5%~~ 2.4% flow measurement uncertainty.

R142

MAY 08 1990

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System TOTAL Flow Rate	$\geq$ ~~378400~~ 375000 gpm#

R130

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

#Includes a ~~3.5%~~ 2.4% flow measurement uncertainty.

R130

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-13)

DESCRIPTION AND JUSTIFICATION FOR

REACTOR COOLANT SYSTEM MINIMUM FLOW RATE REQUIREMENT REDUCTION

Description of Change

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 technical specifications (TSs) to reduce the required reactor coolant system (RCS) total flow rate from greater than or equal to 378,400 gallons per minute (gpm) to greater than or equal to 375,000 gpm in TS 3.2.5, Table 3.2-1. The footnote associated with this flow rate requirement in TS Table 3.2-1 is modified to reflect a 2.4 percent flow measurement uncertainty instead of the generic 3.5 percent utilized for initial licensing and incorporates a typographical correction. The typographical error involves the "#" footnote for the Unit 2 TS Table 3.2-1 where the word "uncertainty" was misspelled. An additional clarification has been incorporated to provide consistency between the TSs for both units by using "Reactor Coolant System Total Flow Rate" as the title for this parameter in TS Table 3.2-1.

Reason for Change

The margin between the currently calculated RCS total flow rates and the present TS limiting value is less than one percent for both SQN Units 1 and 2. Changes in the methods for calculating RCS flow rates in combination with the flow reductions associated with steam generator tube plugging and reactor coolant pump impeller wear have affected the calculated margin at SQN over time. With the potential for additional RCS flow reduction from future tube plugging, pump wear, fuel assembly design changes, or other factors, the less than one percent margin to the minimum TS-required RCS flow rate could result in conditions where the existing TS requirement could not be met. This would result in the shutdown of the unit or prevent unit start-up and require subsequent emergency TS changes or TS waivers of compliance to return to power operation. For these reasons, the reduction in the required RCS flow rate provided by the proposed request and justified by the measurement uncertainty value will minimize the potential impact to power operation.

Justification for Change

The justification for this change is based upon the Westinghouse Electric Corporation Safety Evaluation Checklist (SECL) 92-288 and Letter TVA-91-349 provided in Enclosure 4 of this submittal. These evaluations determined the RCS flow measurement uncertainty based upon the use of RCS elbow tap differential pressures normalized to baseline calorimetric flow rates obtained at the beginning of Cycle 1 for both SQN units. The result of these evaluations is that a 2.4 percent measurement uncertainty is applicable to the RCS flow rate instrumentation in contrast to the 3.5 percent value presently used in the SQN TSs. This 3.5 percent value was used for initial licensing of SQN based on generic uncertainty values for RCS flow measurement instrumentation. The most recent evaluation of the measurement uncertainty for SQN is documented in Westinghouse Letter TVA-91-349 and confirms the excessive conservatism in the 3.5 percent value. The reduction in the measurement uncertainty value is based solely on the Westinghouse calculation that provides the basis for superseding the generic 3.5 percent value. No changes in plant equipment have been involved in this reduction.

For this TS change, the proposed RCS flow rate is greater than or equal to 375,000 gpm, which represents the RCS design flow rate of 365,600 gpm (91,400 gpm in each loop) plus 2.4 percent for measurement uncertainty and rounded to the next highest thousand gpm. This rounding effort provides approximately 600 gpm additional margin in the proposed RCS total flow rate value. Therefore, this reduction in the flow measurement uncertainty and the related decrease in the RCS total flow rate minimum requirement do not alter any of the accident analysis assumptions because only the excess margin for RCS flow instrumentation uncertainty is removed. The design flow rate assumed in the accident analysis remains unchanged by this proposal.

Additional discussions are included in SECL 92-288 for continued power operation with lower RCS flows at reduced reactor power. The proposed change does not incorporate those provisions at this time; however, TVA plans to pursue this option along with upcoming TS changes for new flow mixing fuel assembly designs.

Environmental Impact Evaluation

The proposed change request does not involve an unreviewed environmental question because operation of SQN Units 1 and 2 in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or decisions of the Atomic Safety and Licensing Board.

2. Result in a significant change in effluents or power levels.

3. Result in matters not previously reviewed in the licensing basis for SQN that may have a significant environmental impact.

Enclosure 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-13)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Significant Hazards Evaluation

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes do not alter any of the assumptions used in the accident analysis. The reduction in the minimum TS reactor coolant system (RCS) flow rate only eliminates excess measurement uncertainty. Therefore, no change in any accident analysis assumptions or any plant configuration is involved in the proposed TS change. Based on this, no increase in the probability or consequences of an accident can result from this TS change because RCS design flow rates remain unchanged, ensuring no change in the plant response for normal or accident conditions.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   No plant design parameters, equipment, or operating conditions are altered by the proposed TS change and therefore, no possibility of a new or different kind of accident is created. This elimination of the excess RCS flow measurement uncertainty margin will permit plant operation below the existing RCS flow rate requirement, but not below the measured value that ensures operation at greater than or equal to the design flow assumed in the accident analysis.

3. Involve a significant reduction in a margin of safety.

   The proposed TS change only affects the excess RCS flow measurement uncertainty above the required design flow. This reduction does not affect the "margin of safety" because the RCS design flow remains unchanged and thereby maintains the margin between design operating conditions and the RCS flow rate required to maintain the accident analysis assumptions. Therefore, the proposed reduction in minimum RCS flow to account for the excess measurement uncertainty does not involve a reduction in a margin of safety.

Enclosure 4

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-13)

WESTINGHOUSE ELECTRIC CORPORATION EVALUATIONS

FOR REDUCED REACTOR COOLANT SYSTEM MINIMUM FLOW RATE

SECL 92-288

Customer Reference No(s).
N/A
Westinghouse Reference No(s).
N/A

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST (SECL)

1.) NUCLEAR PLANT(S): Sequoyah Units 1 & 2

2.) SUBJECT (TITLE): RCS Flow Measurement Uncertainty Reduction From 3.5 % to 2.4 %

3.) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59(b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10CFR50.59(a)(1)

3.1) Yes __ No X A change to the plant as described in the FSAR?
3.2) Yes __ No X A change to procedures as described in the FSAR?
3.3) Yes __ No X A test or experiment not described in the FSAR?
3.4) Yes X No __ A change to the plant technical specifications? (See Note on Page 2.)

4.) CHECK LIST - PART B - 10CFR50.59(a)(2) (Justification for Part B answers must be included on page 2.)

4.1) Yes __ No X Will the probability of an accident previously evaluated in the FSAR be increased?
4.2) Yes __ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
4.3) Yes __ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?
4.4) Yes __ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.5) Yes __ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.6) Yes __ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
4.7) Yes __ No X Will the margin of safety as described in the bases to any technical specification be reduced?

SECL-92-288

NOTES:

If the answer to any of the above questions is unknown, indicate under 5.) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50 59(c) and submitted to the NRC pursuant to 10CFR50.90.

5.) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

FOR FSAR UPDATE

Section: ______ Pages: ______ Tables: ______ Figures: ______

Technical Specification Table 3.2-1, DNB Parameters (marked-up Units 1 & 2 copies attached)

SAFETY EVALUATION APPROVAL LADDER:

Nuclear Safety Preparer: L. V. Tomasic Date: 10.7.92

Coordinated with Engineers: C. R. Tuley (Signature on file) Date: ______

Nuclear Safety Group Manager: J. M. Livingston Date: 10.8.92

1.0 BACKGROUND

Westinghouse was requested by TVA to modify the Sequoyah Unit 1 and 2 Technical Specification Table 3.2-1, DNB Parameters, to reflect a reduced RCS Flow measurement uncertainty from 3.5 % to 2.4 %, as identified in previous documents addressing Baseline Calorimetric Flow Uncertainty Combined With Elbow Tap Normalization Uncertainty (Reference 1). This safety evaluation addresses the uncertainty reduction.

[TVA also requested that Westinghouse address a trade-off between RCS Flow and power to allow continued operation when less than the Technical Specification value would be indicated in the plant. This was reviewed and determined to be feasible requiring additional Sequoyah plant specific effort involving the evaluation of the DNB parameters for the FSAR Chapter 15 Non-LOCA Accidents. The findings of this review are summarized for information as follows:

Based on earlier work performed for McGuire and V. C. Summer (which were subsequently approved by the NRC) Westinghouse has determined suitable modifications to allow continued operation with flow reduced by up to 5 % from Thermal Design Flow (as low as 356,000 gpm) at reduced power levels (a corresponding maximum limit on power of 90 % RTP). The power to flow relationship used for this work is a 2.0 % power reduction for each 1.0 % RCS Flow is below Thermal Design Flow, both parameters in integer units. This is a very conservative relationship since the recognized relationship is typically an ~0.5 to 0.6 % RTP power reduction for each 1.0 % RCS Flow reduction to maintain a balance in DNB space. It should be noted that Thermal Design Flow (TDF) is the system flow used as an initial condition in the various safety analyses (91,400 gpm / loop x 4 loops = 365,600 gpm) without measurement uncertainties and 375,000 gpm with measurement uncertainties. All values of flow quoted in this summary are based on TDF = 375,000 gpm, i. e., with a measurement uncertainty of 2.4 %.

The typical relationship assumed for RCS Flow to DNB is 1.0/1.0. Thus a 1 % reduction in RCS Flow would result in a loss of approximately 1 % in DNB margin. The typical relationship assumed for Reactor Power to DNB is 1.0/2.0, i.e., an increase in reactor power of 0.5 % would result in a loss of approximately 1 % in DNB margin. These are typical sensitivity values that must be verified on a plant specific basis and are a function of the DNB correlation utilized. As noted previously, similar changes to plant technical specifications have been made on other plants. The licensing precedent set in these instances (McGuire and V. C. Summer) is the use of a Reactor Power to DNB sensitivity of 2.0/1.0, i.e., a factor of four in the opposite direction than the correlations demonstrate. This is a very conservative assumption and results in a 2 % reduction in reactor power for each 1 % reduction in RCS Flow from TDF.

With this conservative relationship established between RCS Flow and power, it should be apparent that the Core Limits (Safety Limits in the Technical Specifications, Figure 2.1-1) should remain unchanged if the power level of the plant is interpreted somewhat differently with the reduced flow, i.e., Figure 2.1-1 is based on the Fraction of Rated Thermal Power equal to 1.0 being 3411 MWth. When the power level is reduced due to low measured RCS Flow, the Fraction of Rated Thermal Power should reflect the reduced power being considered 1.0, e.g., if RCS Flow is determined to be 98 % TDF (367,000 gpm), then reactor power should be reduced to 96 % of 3411 MWth, or 3275

MWth. Figure 2.1-1 would then be based on the new power level being the equivalent of full power, i.e., 1.0 Fraction of Rated Thermal Power would be 3275 MWth, not 3411 MWth. With this revised interpretation of what power level corresponds to Rated Thermal Power, the Core Limits should remain unchanged.

With the Core Limits unchanged, Overtemperature ΔT (OTDT) and Overpower ΔT (OPDT) reactor trips should be unaffected. This removes any requirements to modify the constants K_1, K_2 and K_3 for OTDT and K_4, K_5 and K_6 for OPDT. However, it is still necessary to scale OPDT to reflect the loop specific, indicated ΔT and T" values and to scale OTDT to reflect the loop specific, indicated ΔT and T' values at the reduced power operating conditions.

Finally, the lower limit of 95 % TDF (356,000 gpm) was selected for several reasons.

1) The RCS Flow - Low reactor trip setpoint remains unchanged (90 % TDF) and it is believed that further reductions in flow, with consideration of instrument uncertainties, could result in spurious actuation of this protection function.

2) The DNB to RCS Flow and DNB to Reactor Power sensitivities are valid and linear for only relatively small changes in the parameters. Decreases in RCS Flow of greater than 5 %would result in changes to Tavg and ΔT which would fall outside the typical bounds of a full power sensitivity calculation. Thus new sensivity calculations would be required.

3) A comparison of the loss in DNB margin due to the reduced RCS Flow vs the DNB margin available in the NIS Power Range reactor trip due to actuation prior to the Safety Analysis Limit cancel at an approximate 5 % reduction in RCS Flow. Further reductions in RCS Flow as a steady state operating condition would require a corresponding reduction in the NIS Power Range Nominal Trip Setpoint to reflect the effective change in allowed full power (Fraction of Rated Thermal Power equal to 1.0).]

2.0 LICENSING BASIS

The work performed is consistent with the requirements of 10CFR50.36 and information documented in WCAP-11239, Revisions 1 - 6 for setpoint/uncertainty calculations previously performed by Westinghouse. As noted above, the uncertainty calculations are essentially the same as those performed previously for Sequoyah Units 1 & 2. Differences from previous calculations lie in the assumption of the normalization of the Cold Leg Elbow Taps to a single, previously performed RCS Flow Calorimetric measurement (Cycle 1) which requires the inclusion of additional uncertainties in the determination of the indicated RCS Flow uncertainty.

3.0 EVALUATION

In late 1991, at the request of TVA, Westinghouse performed calculations to determine the effect of a single normalization of the Cold Leg Elbow Taps to a reference RCS Flow Calorimetric (Reference 1). Based on the current plant configuration; RTD Bypass Elimination, Eagle-21 protection system process racks, the performance of a single normalization of the Cold Leg Elbow Taps to the RCS Flow Calorimetric performed BOL Cycle 1 and indication of RCS Flow via the plant process

computer, the calculated instrument uncertainty is 2.4 % Flow. This is a reduction of ~ 1.1 % Flow from the current NRC mandated value of 3.5 % Flow. Westinghouse has determined that the reduced instrument uncertainty calculation is reasonable and basically consistent with the Westinghouse approach approved by the NRC. The only significant difference is the assumption of normalization to a single, previously performed RCS Flow calorimetric. However, this has been accounted for by the addition of instrument uncertainties usually considered to be zeroed out by the normalization performed each cycle. Based on continued Safety Analysis use of Thermal Design Flow (91,400 gpm per loop) the minimum RCS Flow that must be measured in the plant changes from 378,400 gpm to 375,000 gpm (rounded to the nearest thousand gpm).

4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

While modifications to the plant technical specifications have been determined to be necessary, no unreviewed safety questions have been identified. The seven questions typically answered for a 10CFR50.59 evaluation are noted as follows.

4.1 Will the probability of an accident previously evaluated in the SAR be increased?

With the reduced uncertainty, no increase in the probability of an accident has been noted. No changes to reactor trip setpoints are required, no changes to control system setpoints or gains are required, no changes to installed equipment or hardware in the plant are required, thus the probability of an accident occurring remains unchanged.

4.2 Will the consequences of an accident previously evaluated in the SAR be increased?

With the reduced uncertainty, the initial conditions for all accident scenarios modeled remain unchanged. Therefore, the consequences of an accident will be the same as those previously analyzed.

4.3 May the possibility of an accident which is different than any already evaluated in the SAR be created?

With the reduced uncertainty, no new accident scenarios have been identified. Operation of the plant will be consistent with that previously modeled, i.e., reactor trip setpoints and control function setpoints are the same, thus plant response will be the same and will not introduce any different accident scenarios that have not been evaluated.

4.4 Will the probability of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

No changes to equipment installed in the plant are required. Protection function trip setpoints and control function setpoints remain unchanged and do not introduce additional constraints on equipment important to safety, thus there is no increase in the probability of a malfunction of this equipment.

4.5 Will the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

With the reduced uncertainty the initial conditions present at the initiation of an accident will will not change. Therefore it is expected that the consequences of a malfunction of equipment important to safety will not change.

4.6 May the possibility of a malfunction of equipment important to safety different than any already evaluated in the SAR be created?

No changes to equipment installed in the plant are required. Protection function trip setpoints and control function setpoints remain unchanged and do not introduce additional constraints on equipment important to safety, thus no failure mode not previously evaluated is introduced.

4.7 Will the margin of safety as defined in the BASES to any technical specifications be reduced?

With the changes to the technical specifications required as noted, the margin of safety as defined in the BASES will remain the same.

5.0 CONCLUSIONS

Based on the above it has been determined that the changes noted on the attached for specification Table 3.2-1, DNB Parameters, are acceptable for use at Sequoyah Units 1 and 2.

6.0 REFERENCES

1. TVA-91-349 (ET-NSL-OPL-I-91-628), dated November 6, 1991, entitled Baseline Calorimetric Flow Uncertainty Combined With /Elbow Tap Normalization Uncertainty.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 583°F
Pressurizer Pressure	$\geq$ 2220 psia*
Reactor Coolant System Total Flow	$\geq$ ~~378400~~ gpm# 375,000

R142

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER, physics test, or performance of surveillance requirement 4.1.1.3.b.

\# Includes a ~~3.5%~~ flow measurement uncertainty. 2.4%

R142

MAY 08 1990

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	≤ 583°F
Pressurizer Pressure	≥ 2220 psia*
Reactor Coolant System Flow Rate	≥ ~~370400~~ 375,000 gpm#

R130

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER, physics tes performance of surveillance requirement 4.1.1.

#Inclu es a ~~3.5%~~ 2.4% flow measurement uncertainity.

130

QA Record

B25 '91 1114 251

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

Contract No.
91NNP-86305B

References:
1. N8561
(B25 910807 016)

Mr. P. G. Trudel
Project Engineer
Tennessee Valley Authority
P. O. Box 2000
Soddy Daisy, TN 37379

TVA-91-349
ET-NSL-OPL-I-91-628
November, 6, 1991
Ref: N-021

Tennessee Valley Authority
Sequoyah Nuclear Plants Units 1 and 2
Baseline Calorimetric Flow Uncertainty
Combined with Elbow Tap Normalization Uncertainty

Dear Mr. Trudel:

This is in response to the request of Reference 1.

Attachment A provides the elbow tap measurement repeatability justification for RCS flow verification.

Attachment B provides the results of the uncertainty calculations for the RCS flow measurement employing elbow tap delta - pressure measurements normalized to the baseline calorimetric flow performed at 100% power at the beginning of Cycle 1. The results of the instrument uncertainty for the loss of Flow Reactor trip and revisions to the Setpoint Study (WCAP-11239) are also included (as page mark-ups). WCAP-11239 will be revised and forwarded under separate cover.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

B. J. Garry, Manager
TVA Sequoyah Project
Domestic Projects Department

NJS
1/14

PGT DML CC	DRAFT REPLY	COM COPY	INFO COPY	DUE DATE
EEB MB Sedlacek		✓		12-6
CEB				
MEB VA Bianco		✓		12-6
MTB DL Lundy		✓		12-6
RTB DML	✓			12-11
JA Vogel		✓		12-6
GN Buchanan		✓		12-6
JS Craig			✓	
PH Bryan			✓	

LVT/cld
Attachments

DM Lafever
x8377

cc: D. M. Lafever
R. Fortenberry

RIMS, ET SLE-E - w/attachments

N2N-048
MASTER FILE

C0057:LVT/110691

ELBOW TAP FLOW MEASUREMENT REPEATABILITY

The elbow tap dp measurements on the RCS pump suction piping are being used with increasing frequency, to determine if, and by how much RCS flows might have changed from one fuel cycle to the next. Elbow taps are not considered to be accurate enough to define absolute flows, but the dp measurements have been found to be repeatable and to provide accurate flow change indications.

Elbow flow meters (Ref 1) are a form of centrifugal meter, using the momentum forces developed by the change in flow direction. The principal parameters that determine the dp for a specified flow are the radius of curvature of the elbow and the diameter of the flow channel through the elbow. Experiments on elbow meters have determined that the flow measurements are not affected by differences in surface roughness and have a high degree of repeatability.

Specific phenomena that have affected other types of flow meters, or that might affect the elbow meters in the RCS application have been evaluated to determine if any of these phenomena would affect elbow meter repeatability. In addition, data from an operating plant equipped with a highly accurate flow meter has been compared with the elbow meter measurements at that plant to demonstrate elbow meter repeatability. The results of these evaluations and comparisons are provided in the following sections.

1. Venturi Fouling

   Venturi meters are affected by crud deposits, called fouling, that affect surface roughness and throat area. The fouling is apparently caused by an electrochemical ionization plating of copper and magnetite particles in the feedwater, a process associated with the large velocity increase as the flow approaches the venturi throat. This condition is not present in an RCS elbow; there is no large change in cross section to produce a velocity increase and ionization, and changes in surface roughness do not affect the elbow flow measurement.

2. Meter Dimensional Changes

   The elbow meter is primarily part of the RCS pressure boundary, so there would be only minimal dimensional changes associated with pipe stresses, and pressure and temperature would be the same (full power conditions) whenever measurements are made. Erosion of the elbow surface is unlikely since stainless steel is used, and the velocities are not large (42 fps) relative to erosion. The effects of any dimensional change or of erosion could only affect flow by changing elbow radius or pipe diameter, both large relative to any possible dimensional change. Therefore, the elbow meter is considered to be a highly stable flow measurement element.

3. Upstream Velocity Distribution Effects

The velocity distribution entering e elbow meter will be skewed by the upstream 40° elbow on the steam generator outlet nozzle, and the velocity distribution entering the steam generator outlet nozzle may be skewed due to its off-center location relative to the tube sheet. These velocity distributions, including the distribution in the elbow meter, will remain constant through a fuel cycle, so the elbow meter dp would not change.

Steam generator tube plugging is usually randomly distributed across the tube sheet, so the velocity distribution approaching the outlet nozzle would not change. The velocity distribution could change if extensive tube plugging occurred in one location on the tube sheet, but the change would not be transmitted through the outlet nozzle to the elbow meter. The velocity approaching the outlet nozzle is small (6 fps) compared to the pipe velocity, so this large change in flow area would significantly decrease or flatten any upstream velocity gradient. Therefore, any tube plugging, even if asymmetrically distributed, would not affect the elbow flow measurement repeatability.

4. Flow Measurement Comparisons

The Leading Edge Flow Meters (LEFM) installed in both reactor coolant loops at Prairie Island Unit 2 provide a means to confirm repeatability of the elbow flow meters. The comparisons covered 11 years of operation, during which a significant change in system hydraulics was made. One of the reactor coolant pumps was replaced, and the replacement pump produced additional flow. The LEFM measurements after pump replacement were in agreement with the predicted change, and the elbow flow meters indicated similar changes, but slightly lower flows than measured by the LEFM. The comparisons over 11 years show that the average difference between elbow meter flows and LEFM flows was less than 0.3% flow; the largest single difference was 0.5% flow, with the elbow meter indicating a lower flow than the LEFM. Another comparison, performed before and after the pump replacement, showed that the two measurements agreed to within an average of 0.2% on the ratio of flows when one and two pumps were operating, thus further confirming the relative flow measurements from elbow taps. These flow comparisons are listed in the following tables.

LOOP/METER:	A/LEFM	A/ELBOW	B/LEFM	B/ELBOW
DATE				
	FULL POWER RCS FLOW MEASUREMENT COMPARISONS			
Feb 1980	97519 gpm	(same) gpm	97950 gpm	(same) gpm
Jul 1981	98673	98309	9[illegible]763	97267
Aug 1991	98724	98557	97543	97607
	1 PUMP/2 PUMP FLOW RATIOS			
Dec 1974	1.0819	1.0777	1.0852	1.0875
Jul 1981	1.0794	1.0816	1.0820	1.0820

Ref. 1: "Fluid Meters, Their Theory and Application", 6th Edition, Howard S. Bean, ASME, New York, 1971.

BASELINE CALORIMETRIC FLOW UNCERTAINTY COMBINED WITH ELBOW TAP NORMALIZATION UNCERTAINTY CALCULATIONS

(EVALUATION OF THE EFFECTS OF NOT PERFORMING A PRECISION RCS FLOW CALORIMETRIC FOR NORMALIZATION OF COLD LEG ELBOW TAPS EACH CYCLE)

The effects of not performing a normalization of the cold leg elbow taps each cycle were evaluated.

I) The evaluation is based on the premise that a normalization of the Cold Leg Elbow Taps is made to a previously performed Precision RCS Flow Calorimetric (or group of calorimetrics). This effectively establishes the flow coefficient for the elbow. For subsequent cycles, it is assumed that the transmitter is calibrated, either on a bench or in place, such that it responds within the calibration tolerance for a given Δp input.

II) For each cycle after the normalization, it is assumed that adequate RCS flow is confirmed by verification that the indicated Δp of the Cold Leg Elbow Taps is as expected for that cycle and set of measurement conditions, i.e., if no systematic changes have been made to the pi[illegible]de, then indicated Δp should agree with previous cycle(s) indicated [illegible] for that loop and transmitter (within indication tolerances). If modifications have been made, the indicated Δp should be within indication tolerances of the post modification predicted value for the measurement conditions.

The evaluation was performed based on the following assumptions:

1) Installation of the original, Westinghouse supplied Foxboro E13DH transmitters for measurement of cold leg elbow tap Δp.

2) Measurement and Test Equipment accuracy is per TVA specifications for other protection functions, i.e., SMTE = SCA. In this instance SCA = 0.5 % Δp span, thus SMTE = 0.5 % Δp span.

3) The cold leg elbow tap Δp transmitter span is 110 % TDF.

4) The cold leg elbow tap Δp transmitters are calibrated in place once per 18 months. Typically, this would occur during a reload shutdown.

5) The nominal ambient temperature at calibration is at least 70 °F and peak ambient temperature during normal operation is less than 120 °F.

Based on the above, calculations were performed for the Loss of Flow reactor trip and the RCS Flow indication error with use of the plant process computer. The calculations resulted in the following conclusions:

A) Westinghouse has evaluated the RCS Flow measurement uncertainty noted in Sequoyah Technical Specification [illegible] 2.5, Table 3.2-1, and has found that the value remains unchanged, i.e., a change to the Technical Specifications for this specific value is not required. The value used is 3.5 % Flow. This is conservative in that the original indicated value supported by Westinghouse was 2.4 % Flow (calculated value was 2.38 % Flow). The revised value is 2.4 % Flow (calculated value is 2.44 % Flow). The calculation is based on indication via the plant process computer. Any indication process in front of the process computer using a reasonably accurate DVM would result in a smaller indication error.

B) Changes are required to WCAP-11239 Rev. 5 and to the plant Technical Specifications as noted by the attached marked-up pages. Even though the changes are minor (only a slight increase in the Allowable Value for item 12, Loss of Flow, Table 2.2-1, page 2-5 of the Sequoyah Unit 1 version), it is recommended that either the Technical Specifications be modified, or the NRC advised and administrative control be put in place, prior to implementation.

APPENDIX D

TABLE 3-9
LOSS OF FLOW

Parameter	Allowance*	
Process Measurement Accuracy		
[density effects on Δp cell - ± 0.3 percent flow]$^{+a,c}$	± 0.3	
[precision flow calorimetric - ± 2.3 percent flow + 0.05% flow bias]$^{+a,c}$	± 2.1 + ~~0.05~~	0.04
Primary Element Accuracy		
[Elbow tap repeatability - ± 0.5 percent Δp]$^{+a,c}$	± 0.3	
Sensor Calibration [± 0.5 percent Δp span]$^{+a,c}$ ~~[eliminated by normalization to calorimetric]$^{+a,c}$~~ M&TE [± 0.5 percent Δp span]$^{+a,c}$	~~0.0~~	± 0.3 ± 0.3
Sensor Pressure Effects ~~[eliminated by normalization to calorimetric]$^{+a,c}$~~ [± 0.5 percent Δp span]$^{+a,c}$	~~0.0~~	± 0.3
Sensor Temperature Effects ~~[eliminated by normalization to calorimetric]$^{+a,c}$~~ [± 0.5 percent Δp span]$^{+a,c}$	~~0.0~~	± 0.3
Sensor Drift [± 1.0 percent Δp span]$^{+a,c}$	± 0.6	
Environmental Allowance	0.0	
Rack Calibration		
Rack Accuracy [± 0.4 percent Δp span]$^{+a,c}$	± 0.2	
M&TE [± 0.4 percent Δp span]$^{+a}$	± 0.2	
Rack Temperature Effects [± 0.3 percent Δp span]$^{+a,c}$	± 0.1	
Rack Drift		
0.3 percent Δp span	± 0.1	

+a,c

* In percent flow span (110 percent Thermal Design Flow) percent Δp span converted to flow span via Equation 3-26.1, with F_{max} = 110% and F_N = 100%

Channel Statistical Allowance =

$$\left[\begin{array}{l} \{(0.3)^2 + (2.1)^2 + (0.3)^2 + (\overset{0.3+0.3}{\cancel{0.0}} + 0.6)^2 + (\overset{0.3}{\cancel{0.0}})^2 + (\overset{0.3}{\cancel{0.0}})^2 \\ + (0.2 + 0.2 + 0.1)^2 + (0.1)^2\}^{1/2} + 0.0 + 0.0\underset{4}{\cancel{5}} = \underset{2.5}{\cancel{2.3}}\%\ \text{span} \end{array} \right]^{+a,c}$$

4

TABLE 3
REACTOR PROTECTION SYSTEM/ER
ACTUATION SYSTEM CHANNE
SEQUOYAH UNITS 1 AN

	PROTECTION CHANNEL			S... R					
		1	2	3	4	5	6	7	8
		PROCESS MEASUREMENT ACCURACY (1)	PRIMARY ELEMENT ACCURACY (1)	CALIBRATION CURACY (1)	MEASUREMENT & TEST EQUIPMENT ACCURACY (1)	PRESSURE EFFECTS (1)	TEMPERATURE EFFECTS (1)	DRIFT (1)	ENVIRON-MENTAL ALLOWANCE (1)
1	POWER RANGE, NEUTRON FLUX - HIGH SETPOINT	1.7 (13) & 4.2	—	(4)	—	—	(4)	(4)	—
2	POWER RANGE, NEUTRON FLUX - LOW SETPOINT	1.7 (13) & 4.2	—	(4)	—	—	(4)	(4)	—
3	POWER RANGE, NEUTRON FLUX - HIGH POSITIVE RATE	(14)	—	(14)	—	—	(14)	(14)	—
4	POWER RANGE, NEUTRON FLUX - HIGH NEGATIVE RATE	(14)	—	(14)	—	—	[illegible]	(14)	—
5	INTERMEDIATE RANGE, NEUTRON FLUX	(15)	(15)	(15)	(15)	(15)	(15)	(15)	(15)
6	SOURCE RANGE, NEUTRON FLUX	(15)	(15)	(15)	(15)	(15)	(15)	(15)	(15)
7	OVERTEMPERATURE ΔT — ΔT CHANNEL	1.1	—	0.5	—	—	—	0.8	
8	Tavg CHANNEL	—	—	—	—	—	—	—	
9									
10	PRESSURIZER PRESSURE CHANNEL	—	—	0.2	0.2	—	0.2 & 0.4 (12)	0.3	—
11	ΔI CHANNEL	1.7 (16) & 0.8 (17)	—	—	—	—	—	—	
12	OVERPOWER ΔT - ΔT CHANNEL	1.1	—	0.5	—	—	—	0.8	0.14 (27)
13									
14	Tavg CHANNEL	—	—	—	—	—	—	—	
15	PRESSURIZER PRESSURE - LOW, REACTOR TRIP (Barton)								
16	PRESSURIZER PRESSURE - HIGH (Barton)	—	—	0.5	0.5	—	0.5 & 1.5 (12)	1.0	
17	PRESSURIZER WATER LEVEL - HIGH (Barton)	—	—	0.5	0.5	—	0.8	4.25 (19)	—
18	LOSS OF FLOW	2.0	—	0.5	0.5	0.5	0.5	1.0	—
19	STEAM GENERATOR WATER LEVEL - LOW-LOW Adverse (Unmodified Barton)	0.3 (21) & 2.1 (22)	0.3	0.3 (28)	0.3	0.3 (28)	0.3 (28)	0.8	—
20	STEAM GENERATOR WATER LEVEL - LOW-LOW EAM (Unmodified Barton)	2.0	—	0.5	0.6	0.5	0.5	1.0	10.0, 5.2(11), 0.0(27)
21	STEAM GENERATOR WATER LEVEL - LOW-LOW Adverse (Modified Barton)	2.0	—	0.5	0.5	0.5	0.5	1.0	4.3, 3.1(11)
	STEAM GENERATOR WATER LEVEL - LOW-LOW EAM (Modified Barton)	7.0	—	0.5	0.6	0.5	0.6	1.0	6.0, 5.2(11), 0.0(27)
	UNDERVOLTAGE - RCP	2.0	—	0.6	0.5	0.6	0.8	1.5	4.0, 3.1(11)
24	UNDERFREQUENCY - RCP	(20)	(20)	(20)	(20)	(20)	(20)	(20)	(20)
25	CONTAINMENT PRESSURE - EAM (Foxboro)	—	—	—	—	—	—	—	—
26	CONTAINMENT PRESSURE - HIGH (Foxboro)	—	—	0.5	0.4	—	0.8	0.8	2.1
27	CONTAINMENT PRESSURE - HIGH-HIGH (Foxboro)	—	—	0.5	0.4	—	0.8	0.8	2.8
28	CONTAINMENT PRESSURE - EAM (Barton)	—	—	0.5	0.4	—	0.8	0.8	2.8
29	CONTAINMENT PRESSURE - HIGH (Barton)	—	—	0.5	0.3	—	1.0	1.0	—
30	CONTAINMENT PRESSURE - HIGH-HIGH (Barton)	—	—	0.5	0.3	—	0.7	1.0	1.4
31	PRESSURIZER PRESSURE - LOW, SI (Barton)	—	—	0.5	0.3	—	0.7	1.0	1.4
32	STEAMLINE PRESSURE - LOW (Foxboro)	—	—	0.5	0.5	—	0.5	1.0	13.7
33	STEAM GENERATOR WATER LEVEL - HIGH-HIGH (Barton)	—	—	0.5	0.3	—	1.3	1.0	10.0
34	RWST LEVEL - LOW (Barton)	11.0 (28)	—	0.5	0.5	0.5	0.5	1.0	—
35	RWST LEVEL - HIGH (Barton)	(24)	(24)	(24)	(24)	(24)	(24)	(24)	(24)
36	NEGATIVE STEAMLINE PRESSURE RATE - HIGH	(24)	(24)	(24)	(24)	(24)	(24)	(24)	(24)
37	VESSEL ΔT EQUIVALENT TO POWER	—	—	(14)	—	—	(14)	(14)	—
		1.1	—	0.5	—	—	—	0.8	0.14 (27)

OTES:

ALL VALUES IN PERCENT SPAN.
AS NOTED IN TABLE 15.1.3-1 OF FSAR.
AS CALCULATED USING THE APPROVED METHODOLOGY AND NOTED ON TABLE 4-3 OF THIS REPORT.
[INCLUDED IN CALORIMETRIC ALLOWANCE IN PROCESS MEASUREMENT ACCURACY.]^+a,c
NOT USED IN SAFETY ANALYSIS.
AS NOTED IN FIGURE 15.1.3-1 OF FSAR.
AS NOTED IN TABLE 2.2-1 NOTE 1 OF PLANT TECHNICAL SPECIFICATIONS.
AS NOTED IN TABLE 2.2-1 NOTE 3 OF PLANT TECHNICAL SPECIFICATIONS.
NOT NOTED IN TABLE 15.1.3-1 OF FSAR BUT USED IN SAFETY ANALYSIS

10. SUPERSEDES INFORMATION IN FSAR TABLE 15.1.3-1
11. [ALLOWANCE FOR REFERENCE LEG HEATUP]^+a,c
12. [ALLOWANCE FOR BARTON THERMAL NON-REPEATABILITY, TREATED AS A BIAS]^+a,c
13. [POWER CALORIMETRIC ALLOWANCE.]^+a,c
14. [USE OF A RATE (DERIVATIVE) ELIMINATES SENSOR STEADY STATE ERRORS]^+a,c
15. NOT IN WESTINGHOUSE SCOPE - SEE TVA CALCULATION SQN-EEB-PS-TI28-0001.
16. INCORE EXCORE ΔI COMPARISON AS NOTED IN TABLE 4.3-1 OF PLANT TECHNICAL SPECIFICATIONS.
17. [INCORE FLUX MAP ΔI UNCERTAINTY.]^+a,c
18. SAFETY ANALYSIS LIMIT ENSURES THIS RWST SWITCHOVER TO CONTAINMENT SUMP IS COMPLETED BEFORE VORTEXING OCCURS IN THE RWST.
19. [DRIFT ALLOWANCE FOR BARTON TRANSMITTER, TREATED AS A BIAS]^+a,c

5

:ERED SAFETY FEATURES
:ROR ALLOWANCES
VISION 5

INSTRUMENT RACK											
9	10	11	12	13	14	15	16	17	18	19	
CALIBRATION ACCURACY (1)	MEASUREMENT & TEST EQUIPMENT ACCURACY (1)	COMPARATOR SETTING ACCURACY (1)	TEMPERATURE EFFECTS (1)	DRIFT (1)	SAFETY ANALYSIS LIMIT (3)	STS ALLOWABLE VALUE (2)	STS TRIP SETPOINT (2)	TOTAL ALLOWANCE (1)	CHANNEL STATISTICAL ALLOWANCE (1)	MARGIN (1)	
0.5	—	0.5	0.5	1.0	118% RTP	111.?% RTP	109% RTP	7.5	5.0	+2.5	1
0.5	—	0.5	0.5	1.0	30% RTP	27.4% RTP	25% RTP	8.5	5.0	+3.5	2
0.5	—	0.5	0.5	0.5	(5)	6.3% RTP	5.0% RTP	—	1.8	—	3
0.5	—	0.5	0.5	0.5	6.8% RTP (5)	6.5% RTP	5.0% RTP	1.8	1.8	+0.0	4
(15)	(15)	(15)	(15)	(15)	(5)		27% RTP	—	(15)	—	5
(15)	(15)	(15)	(15)	(15)	(5)	(15)	1.05E+6 CPS	—	(15)	—	6
0.5	0.5	—	1.2	0.4							7
—	—	—		—							8
					(8)	(7) + 1.9% ΔT SPAN	(7)	5.7	4.0	+1.7	9
0.2	0.2	—	0.3	0.3							10
0.2	0.2	—	0.3	0.3							11
0.5	0.5	—	1.2	0.4							12
					(8)	(8) + 1.7% ΔT SPAN	(8)	4.8	3.0	+1.8	13
—	—	—		—		89.6					14
0.2	0.2	—	0.3	0.3	1846 PSIG	1964.6 PSIG	1970 PSIG	15.6	3.7	+11.9	15
0.2	0.2	—	0.3	0.3	2446 PSIG (10)	2580.2 PSIG	2386 PSIG	7.5	5.6	+1.9	16
0.2	0.2	—	0.3	0.3	(5)	92.7% SPAN	92% SPAN	—	3.0		17
0.2	0.2	—	0.1	0.1	88.9% DESIGN	[illegible]% DESIGN	90% DESIGN	2.8	~~2.8~~ 2.5	+0.35	18
0.2	0.2	—	0.1	0.3	0% SPAN	18.2% SPAN	18.8% SPAN	18.8	18.2	+0.6	19
0.2	0.2	—	0.3	0.3	0% SPAN	12.4% SPAN	13.0% SPAN	13.0	12.4	+0.6	20
[illegible]	0.2	—	0.3	0.3	0% SPAN	14.2% SPAN	14.8% SPAN	14.8	14.2	+0.6	21
	0.2	—	0.3	0.3	0% SPAN	10.1% SPAN	13.7% SPAN	10.7	10.1	+0.6	22
(20)	(20)	(20)	(20)	(20)	4800 VAC	(19)	5022 VAC (30)	(20)	(20)	(20)	23
0.8	0.5	—	—	1.0	56.8 HZ (9)	56.0 HZ	56.0 HZ	3.5	2.5	+1.0	24
0.2	0.2	—	0.3	0.[illegible]	1.2 PSIG (9)	0.6 PSIG	0.5 PSIG	4.4	3.8	+0.6	25
0.2	0.2	—	0.3	0.3	2.42 PSIG (9)	1.6 PSIG	1.54 PSIG	5.5	4.4	+1.1	26
0.2	0.2	—	0.3	0.3	3.60 PSIG (9)	2.9 PSIG	2.81 PSIG	5.5	4.4	+1.1	27
0.2	0.2	—	0.3	0.3	1.2 PSIG (9)	0.6 PSIG	0.5 PSIG	3.9	2.2	+1.7	28
0.2	0.2	—	0.3	0.3	2.42 PSIG (9)	1.7 PSIG	1.54 PSIG	4.8	3.5	+1.4	29
0.2	0.2	—	0.3	0.3	3.60 PSIG (9)	2.9 PSIG	2.81 PSIG	4.9	3.5	+1.4	30
0.2	0.2	—	0.3	0.3	1750 PSIG (10)	1864.3 PSIG	1870 PSIG	17.5	15.9	+1.6	31
0.2	0.2	—	0.3	0.3	456 PSIG (9)	592.2 PSIG	600 PSIG	13.9	12.3	+1.6	32
0.2	0.2	—	0.3	0.3	96.2% SPAN (10)	81.7% SPAN	81.0% SPAN	12.2	11.2	+1.0	33
(24)	(24)	(24)	(24)	(24)	(18) (24)	(24)	27.4% SPAN	(24)	(24)	(24)	34
(24)	(24)	(24)	(24)	(24)	(24) (28)	(24)	27.4% SPAN	(24)	(24)	(24)	35
0.2	0.2	—	0.3	0.3	(5)	107.2 PSIG	100.0 PSIG	—	0.7	—	36
0.5	0.5	—	1.2	0.4	4.0% ΔT SPAN (5) (28)	(28) + 1.7% ΔT SPAN	(28)	6.0	3.0	+3.0	37

?. NOT IN WESTINGHOUSE SCOPE - SEE TVA CALCULATION RCP-UV-DEVICE 27-SYSTEM 202.

?. [DENSITY EFFECTS ON ΔP CELL]***

?. [PRECISION FLOW CALORIMETRIC UNCERTAINTY ±1.3% FLOW + 0.05% FLOW BIAS]***

?. ~~[NORMALIZATION OF ELBOW TAPS TO PRECISION FLOW CALORIMETRIC ELIMINATES SENSOR CALIBRATION, TEMPERATURE, AND PRESSURE ERRORS]***~~

?. NOT IN WESTINGHOUSE SCOPE - SEE TVA CALCULATION SQN-EEB-MS-TI28-0015.

25.

26. SAFETY ANALYSIS LIMIT ENSURES THAT AT RWST SWITCHOVER, THE CONTAINMENT SUMP LEVEL SAFETY LIMIT IS NOT VIOLATED.

27. CABLE IR, AS PROVIDED BY TVA, TREATED AS A BIAS.

28. NOTE 5 OF TABLE 2.2-1 OF PLANT TECHNICAL SPECIFICATIONS.

29. PER TVA CALCULATION SQN-CSS-076.

TA[illegible]
WESTINGHOUSE PROTECTION S[illegible]
SEQUOYAH UN[illegible]

	PROTECTION CHANNEL	TOTAL ALLOWANCE (TA)(9)	(9) (A)(1)	(9) (S)(2)	(9) (T)(3)
1	POWER RANGE, NEUTRON FLUX - HIGH SETPOINT	7.5	20.78	0.0	2.0
2	POWER RANGE, NEUTRON FLUX - LOW SETPOINT	8.3	20.78	0.0	2.0
3	POWER RANGE, NEUTRON FLUX HIGH POSITIVE RATE	1.8	0.25	0.0	1.1
4	POWER RANGE, NEUTRON FLUX - HIGH NEGATIVE RATE	1.8	0.25	0.0	1.1
5	INTERMEDIATE RANGE, NEUTRON FLUX	(14)	(14)	(14)	(14)
6	SOURCE RANGE, NEUTRON FLUX	(14)	(14)	(14)	(14)
7	OVERTEMPERATURE ΔT	5.7	6.21	1.8 + 0.8	1.9
8	OVERPOWER ΔT	6.8	2.54	1.8	1.7
9	PRESSURIZER PRESSURE - LOW, REACTOR TRIP (BARTON)	15.8	0.91	2.0	0.7
10	PRESSURIZER PRESSURE - HIGH (BARTON)	7.5	0.91	1.0	0.7
11	PRESSURIZER WATER LEVEL - HIGH (BARTON)	8.0	4.88	2.0	0.7
12	LOSS OF FLOW	2.8	4.55 ~~4.88~~	1.1 ~~0.5~~	0.4 ~~0.5~~
13	STEAM GENERATOR WATER LEVEL - LOW-LOW ADVERSE (UNMODIFIED BARTON)	18.8	4.58	2.0	0.7
14	STEAM GENERATOR WATER LEVEL - LOW-LOW EAM (UNMODIFIED BARTON)	13.0	4.58	2.0	0.7
15	STEAM GENERATOR WATER LEVEL - LOW-LOW ADVERSE (MODIFIED BARTON)	14.8	4.58	2.0	0.7
16	STEAM GENERATOR WATER LEVEL - LOW-LOW EAM (MODIFIED BARTON)	10.7	4.88	2.0	0.7
17	UNDERVOLTAGE - RCP (12)	(16)	(16)	(16)	(16)
18	UNDERFREQUENCY - RCP	3.33	0.0	0.0	2.3
19	CONTAINMENT PRESSURE - EAM (FOXBORO)	4.4	0.7	1.5	0.8
20	CONTAINMENT PRESSURE - HIGH (FOXBORO)	5.5	0.7	1.5	0.7
21	CONTAINMENT PRESSURE HIGH-HIGH (FOXBORO)	5.5	0.7	1.5	0.7
[illegible]	CONTAINMENT PRESSURE EAM (BARTON)	3.9	1.06	1.8	0.7
[illegible]	CONTAINMENT PRESSURE - HIGH (BARTON)	4.9	0.98	1.8	0.7
24	CONTAINMENT PRESSURE HIGH-HIGH (BARTON)	4.9	0.98	1.8	0.7
25	PRESSURIZER PRESSURE LOW, S.I. (BARTON)	17.5	0.91	2.0	0.7
26	STEAMLINE PRESSURE - LOW (FOXBORO)	13.9	1.62	1.8	0.7
27	STEAM GENERATOR WATER LEVEL - HIGH-HIGH (BARTON) (TG-ST)	12.2	121.58	2.0	0.7
28	RWST LEVEL - LOW (BARTON)	(15)	(15)	(15)	(15)
29	RWST LEVEL - HIGH (BARTON)	(15)	(15)	(15)	(15)
30	NEGATIVE STEAMLINE PRESSURE RATE - HIGH	2.8	3.09	0.0	0.7
31	VESSEL ΔT EQUIVALENT TO POWER	6.0	2.54	1.8	1.7

NOTES:

(1) [A= [(PMA)² + (PEA)² + (SPE)² + (STE)² + (RTE)²]]^{a,c}

(2) [S=(SCA+SMTE+SD)]^{a,c}

(3) [T, =(RD+RCA+RMTE+RCSA) OR T = TA-((A+S)²)$^{1/2}$+EA)

OR T =((RD+RCA+RMTE+RCSA)² +(RD+RCA+RMTE+RCSA)²)$^{1/2}$]^{a,c}

(4) [Z=((A)²)$^{1/2}$+ EA]^{a,c}

(5) TAVG - 100°F, ΔP - 800 PSI, φ - 120% RTP, ΔT - 150% RTP, ΔI - ±60% [illegible]

(6) TAVG - 100°F, ΔP - 800 PSI, φ - 120% RTP, ΔT - 150% RTP

(7) [THIS COLUMN PROVIDES THE MAXIMUM VALUE FOR A BISTABLE ASS[illegible] EQUATION TA ≥ Z + S + R IMPLIES SOME MAXIMUM VALUE FOR R, I.E., [illegible] OF S [illegible] LESS THAN THAT ASSUMED, A BISTABLE SETPOINT IN EXCESS [illegible] CO[illegible]ED REPORTABLE.]^{a,c}

(8) AS NOTED IN NOTE 1 OF TABLE 2.2-1 OF STS.

(9) ALL VALUES IN PERCENT SPAN.

(10) AS NOTED IN TABLE 3.3-4 IN STS.

(11) AS NOTED IN NOTE 3 OF TABLE 2.2-1 OF STS.

(12) VALUES REFLECT THE 95% GUARANTEED ACTUATION SETTING AS PER T[illegible]

 7

M STS SETPOINT INPUTS
I AND 2

(9) (2)(4)	INSTRUMENT SPAN	STS TRIP SETPOINT	STS ALLOWABLE VALUE	MAXIMUM VALUE (7)	
4.56	120% RTP	109% RTP	111.4% RTP	112.5% RTP	1
4.56	120% RTP	25% RTP	27.4% RTP	28.5% RTP	2
0.50	120% RTP	5.0% RTP	6.3% RTP	6.3% RTP	3
0.50	120% RTP	5.0% RTP	6.3% RTP	6.3% RTP	4
(14)	(14)	25% RTP	(14)	(14)	5
(14)	(14)	1E+05 CPS	(14)	(14)	6
2.93	(5)	(5)	(5) + 1.2% AT SPAN	(5) + 0.8% AT SPAN	7
1.74	(5)	(11)	(11) + 1.7% AT SPAN	(11) + 1.5% AT SPAN	8
2.08	300 PSI	1970 PSIG	1964.8 PSIG	1977.5 PSIG	9
4.81	800 PSI	2385 PSIG	2390.2 PSIG	2398.5 [illegible]	10
2.14	100% SPAN	92% SPAN	92.7% SPAN	95.2% SPAN	11
~~2.14~~ 2.18	110% DESIGN FLOW	90% FLOW	89.6 [illegible]% FLOW	90.0 [illegible]% FLOW	12
17.34	100% SPAN	18.8% NARROW RANGE SPAN	18.2% NARROW RANGE SPAN	18.8% NARROW RANGE SPAN	13
11.54	100% SPAN	13.0% NARROW RANGE SPAN	12.4% NARROW RANGE SPAN	13.0% NARROW RANGE SPAN	14
13.34	100% SPAN	14.8% NARROW RANGE SPAN	14.2% NARROW RANGE SPAN	14.8% NARROW RANGE SPAN	15
9.24	100% SPAN	10.7% NARROW RANGE SPAN	10.1% NARROW RANGE SPAN	10.7% NARROW RANGE SPAN	16
(16)	1800 VAC	5[illegible] VAC	(16)	(16)	17
0.0	4 Hz	56.0 Hz	55.8 Hz	55.8 Hz	18
2.94	18 PSI	0.5 PSIG	0.6 PSIG	0.5 PSIG	19
3.44	18 P[illegible]	1.54 PSIG	1.8 PSIG	1.8 PSIG	20
3.44	18 PSI	2.81 PSIG	2.9 PSIG	2.9 PSIG	21
2.03	18 PSI	0.5 PSIG	0.6 PSIG	0.7 PSIG	22
2.14	18 PSI	1.54 PSIG	1.7 PSIG	1.7 PSIG	23
2.14	18 PSI	2.81 PSIG	2.9 PSIG	3.0 PSIG	24
14.28	800 PSI	1870 PSIG	1864.8 PSIG	1880.1 PSIG	25
11.27	1200 PSI	600 PSIG	590.2 PSIG	586.7 PSIG	26
11.03	100% SPAN	81.0% SPAN	81.7% SPAN	81.0% SPAN	27
(15)	100% SPAN	27.4% SPAN	(15)	(15)	28
(15)	100% SPAN	27.4% SPAN	(15)	(15)	29
0.25	1200 PSI	100 PSIG	107.8 PSIG	121.0 PSIG	30
2.29	150% RTP	(13)	(13) + 1.7% AT SPAN	(13) + 2.7% AT SPAN	31

NG Z AND THE LARGEST VALUE FOR S. SATISFYING THE S COLUMN. WITHOUT DETERMINATION THAT THE VALUE THE ALLOWABLE VALUE AND THIS COLUMN WOULD BE

(13) AS NOTED IN NOTE 5 OF TABLE 2.2-1 OF PLANT TECHNICAL SPECIFICATIONS.

(14) NOT IN WESTINGHOUSE SCOPE - SEE TVA CALCULATION SQN-EEB-PS-TI28-0001.

(15) NOT IN WESTINGHOUSE SCOPE - SEE TVA CALCULATION SQN-EEB-MS-TI28-0016.

(16) NOT IN WESTINGHOUSE SCOPE - SEE TVA CALCULATION RCP-UV-DEVICE 27-SYSTEM 202

HOUSE CALCULATION NO. RCP-UV-DEVICE 27.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Total Allowance (TA)	Z	Sensor Drift (S)	Trip Setpoint	Allowable Value
12.	Loss of Flow	2.8	~~2.14~~ 2.18	~~0.6~~ 1.1	≥ 90% of loop design flow*	≥ ~~89.4%~~ 83.6% of loop design flow*
13.	Steam Generator Water Level - Low Low (Modified Barton Transmitters)					
	a. Vessel ΔT Equivalent to Power ≤ 50% RTP	6.0	1.74	1.6	Vessel ΔT variable input ≤ 50% RTP	Vessel ΔT variable input ≤ trip setpoint + 2.5% RTP
	Coincident with					
	Steam Generator Water Level-Low-Low (Adverse) and	14.8	13.34	2.0	≥ 14.8% of Narrow Range Instrument span	≥ 14.2% of Narrow Range Instrument span
	Containment Pressure-EAM	4.4	2.94	1.5	≤ 0.5 psig	≤ 0.6 psig
	or					
	Steam Generator Water Level-Low-Low (EAM)	10.7	9.24	2.0	≥ 10.7% of Narrow Range Instrument span	≥ 10.1% of Narrow Range Instrument span
	With a time delay (T_s) if one Steam Generator is affected				≤ T_s (Note 5)	≤ (1.01) T_s (Note 5)
	or					

*Loop design flow = 91,400 GPM