TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	≤ 10% Turbine Impulse Pressure Equivalent	≤ 12.4% Turbine Impulse Pressure Equivalent	R145
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Trip	≤ 35% of RATED THERMAL POWER	≤ 37.4% of RATED THERMAL POWER	R145
23. Power Range Neutron Flux - (P-10) - Enable Block of Source, Intermediate, and Power Range (low setpoint) Reactor Trips	≥ 10% of RATED THERMAL POWER	≥ 7.6% of RATED THERMAL POWER	R145
24. Reactor Trip P-4	Not Applicable	Not Applicable	
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	≤ 50% of RATED THERMAL POWER	≤ 52.4% of RATED THERMAL POWER	R145

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \{K_1 - K_2 \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)[T-T'] + K_3(P-P') - f_1(\Delta I)\}$

R145

Where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = Lead-lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 \cancel{=} \cancel{12} \geq 5$ secs, $\tau_5 \cancel{=} \leq 3$ sec.

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 ≤ 1.15

K_2 $\cancel{=} \geq$ 0.011

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, & τ_2	=	Time constants utilized in the lead-lag controller for T_{avg}, $\tau_1 \cancel{\ngtr} \geq 33$ secs., $\tau_2 \cancel{\ngtr} \leq 4$ secs.
T	=	Average temperature °F
T'	$\leq$	578.2°F (Nominal T_{avg} at RATED THERMAL POWER)
K_3	=	0.00055
P	=	Pressurizer pressure, psig
P'	=	2235 psig (Nominal RCS operating pressure)
S	=	Laplace transform operator (sec^{-1})

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 29 percent and + 5 percent $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

R145

R145

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -29 percent, the ΔT trip setpoint shall be automatically reduced by 1.50 percent of its value at RATED THERMAL POWER. R23

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by 0.86 percent of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6(T-T'') - f_2(\Delta I)\}$ R145

Where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = as defined in Note 1

τ_4, τ_5 = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ 1.087 R118

K_5 $\ngtr$ 0.02/°F for increasing average temperature and 0 for decreasing average $\geq$ temperature

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation R145

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: (Continued)

τ_3	=	Time constant utilized in the rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs. R145
K_6	$\geq$	0.0011 for $T > T''$ and $K_6 \geq 0$ for $T \leq T''$
T	=	as defined in Note 1
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 578.2°F)
S	=	as defined in Note 1
$f_2(\Delta I)$	=	0 for all ΔI

NOTE 3: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 1.9 percent ΔT span.

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 1.7 percent ΔT span. R145

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \{K_1 - K_2 \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)[T - T'] + K_3(P-P') - f_1(\Delta I)\}$ R132

where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = Lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT, $\tau_4 \geq \cancel{\geq}$ 5 ~~12~~ secs, $\tau_5 \leq$ 3 secs

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.15 R21

K_2 $\geq$ 0.011

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation R132

τ_1, & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, $\tau_1 \geq$ 33 secs., $\tau_2 \leq$ 4 secs.

T = Average temperature °F

T' $\leq$ 578.2°F (Nominal T_{avg} at RATED THERMAL POWER) R132

K_3 = 0.00055

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

S = Laplace transform operator. sec^{-1}

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 29 percent and + 5 percent $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER). R21

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -29 percent, the ΔT trip setpoint shall be automatically reduced by 1.50 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by 0.86 percent of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6[T - T''] - f_2(\Delta I)\}$ R132

where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = as defined in Note 1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 2: (Continued)

τ_4, τ_5 = as defined in Note 1 R132

ΔT_o = as defined in Note 1

K_4 $\leq$ 1.087 R104

K_5 $\geq$ 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation R132

τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, τ_3 $\geq$ 10 secs.

K_6 $\geq$ 0.0011 for T > T" and K_6 $\geq$ 0 for T $\leq$ T" R21

T = as defined in Note 1

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 578.2°F)

S = as defined in Note 1

$f_2(\Delta I)$ = 0 for all ΔI

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-95-11, REVISION 1)

DESCRIPTION AND JUSTIFICATION FOR

REVISION OF OVERTEMPERATURE AND OVERPOWER

DELTA TEMPERATURE EQUATION TIME CONSTANTS

Description of Change

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 technical specifications (TSs) to revise the overtemperature delta temperature (OTΔT) and overpower delta temperature (OPΔT) equation constant numerical value τ_4 in TS Table 2.2-1. The τ_4 value will be changed from 12 seconds to 5 seconds and the τ_5 value will remain at a value of 3 seconds. An additional enhancement has been incorporated to change the equality signs for the τ and K constant numerical values to either a less than or equal to or a greater than or equal to function as applicable.

Reason for Change

SQN has experienced OPΔT turbine runback alarms on individual channels resulting in partial runback signals. During functional testing at power, as required by TSs, these occurrences could result in turbine runbacks or reactor trips because the tested channel is placed in the trip condition completing the required logic for actuation. The τ constant numerical values have been reanalyzed to provide additional margin to these setpoints and minimize the potential for turbine runback and reactor trip signals. Incorporation of the new τ numerical value will provide a reasonable margin to the turbine runback and reactor trip setpoints and minimize the challenges to safeguard function actuation. The revision of the equality signs for the equation constants is consistent with standard TS (NUREG-1431). The use of greater than or equal to or less than or equal to functions for these constant numerical values is more appropriate and will provide the flexibility to set these values within the limits utilized in the safety analysis.

Justification for Changes

The OTΔT trip provides core protection to prevent departure from nucleate boiling for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transit, thermowell, and resistance temperature device (RTD) response time delays from the core to the temperature detectors, and pressure is within the range between the high and low pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to the RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip setpoint is automatically reduced.

The OPΔT reactor trip provides assurance of fuel integrity, limits the required range for OTΔT protection, and provides a backup to the high neutron flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, dynamic compensation for transport, thermowell, and RTD response time delays from the core to the RTD output indication. The OPΔT provides protection to mitigate the consequences of various size steam breaks.

Westinghouse Electric Corporation has evaluated SQN's setpoints for OTΔT and OPΔT and has verified that sufficient margin exists in the SQN accident analyses to support relaxation of the lead/lag dynamic compensation. This evaluation addresses the Final Safety Analysis Report (FSAR) Chapter 6 and 15 accidents that are affected by a revised lead/lag dynamic compensation. These evaluated accidents only include the non-loss-of-coolant accidents (LOCA) because the LOCA and containment integrity accidents are not affected by the revised compensations. The results of the evaluation, which are included in Enclosure 4, indicate that the proposed revision of the τ_4 constant numerical value is acceptable in that all acceptance criteria of the current safety analyses described in the FSAR continue to be met.

The revision of the equality signs for the constant numerical values of the OTΔT and OPΔT equations does not change these functions. This change will clarify that the adjustment of the related setpoints for these values can be set more conservatively than assumed in the analysis. This change is consistent with the provisions in NUREG-1431.

Environmental Impact Evaluation

The proposed change 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 NRC'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 CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-95-11, REVISION 1)

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 revision of the τ_4 constant numerical value in the overtemperature delta temperature (OTΔT) and overpower delta temperature (OPΔT) equations have been analyzed by the enclosed Westinghouse Electric Corporation evaluation and have been found to have sufficient margin for the proposed change. This evaluation shows that the proposed changes are bounded by the existing analysis for Chapter 6 and 15 accidents. The setpoint change will continue to meet the applicable safety analysis acceptance criteria for the transients evaluated. The offsite dose rates for postulated accidents have not exceeded the values stated in the Updated Final Safety Analysis Report as a result of this change. The clarification of the equality signs for the constant numerical values does not change plant or accident mitigation functions. Therefore, the proposed changes will not increase the consequences of an accident.

   This change affects the OTΔT and OPΔT functions that are designed to mitigate the consequences of an accident and are not considered to be an accident initiating source. Therefore, the probability of an accident is not increased by the proposed change.

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

   The revision of lead/lag dynamic compensation for the OTΔT and OPΔT functions do not impact accident initiators because these functions are used for accident mitigation and are not postulated as a source. Therefore, the possibility of a new or different kind of accident is not created by the proposed revision.

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

   The proposed revision to the lead/lag compensation for the OTΔT and OPΔT functions does not invalidate the conclusions in the safety analysis. Margins provided for in the safety analysis are maintained with the proposed changes such that no reduction in the margin of safety is involved.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-95-11, REVISION 1)

WESTINGHOUSE ELECTRIC CORPORATION

EVALUATION OF REVISED TIME CONSTANTS

Customer Reference No(s).
N/A

Westinghouse Reference No(s).
N/A

WESTINGHOUSE
SAFETY EVALUATION CHECK LIST

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

2) CHECK LIST APPLICABLE TO: OPDT / OTDT Operating Margin Improvement Via Revised Lead / Lag Compensation (From 12/3 To 5/3)

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 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

3.1) Yes X No_ 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 (Appendix A to the Operating License)?

4) CHECK LIST - PART B (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 defined in the bases to any Technical Specification be reduced?

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

If the answer to any of the above questions in Part (3.4) or Part B cannot be answered in the negative, the change review requires an application for license amendment in accordance with 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.

Reference document(s):

N/A

FOR FSAR UPDATE

See separately transmitted UFSAR mark-ups [For 5/3 Lead/Lag: Section 15.2.2 pages 15.2-5 through 15.2-9, Table 15.2-1 (Sheet 1), and Figures 15.2.2-1 through 15.2.2-7]

SLB w/RWAP mark-ups (Ref. WCAP-12504)

See attached Tech Spec mark-ups [Table 2.2-1 (Page 2-7 and 2-8, Unit 1; Page 2-9, Unit 2)]

See attached PL&S mark-ups.[Pages 14 and 15 of 102]

SIGNATURES

Prepared by: L.V. Tomasic L V Tomasic Date: 7·18·95

Reviewed by: T. J. Kitchen T J Kitchen Date: 7/18/95

SUMMARY

The overall purpose of this safety evaluation is to increase the Overtemperature Delta-T and the Overpower Delta-T (OPDT / OTDT) operating margin to trip in order to increase the turbine runback margin (associated with OPDT / OTDT) by relaxing the lead / lag dynamic compensation.

This safety evaluation addresses revising the OPDT and OTDT lead / lag dynamic compensation from 12/3 to 5/3 and concludes that this revision does not represent an unreviewed safety question pursuant to 10CFR50.59,(a), (2), criteria, provides the basis for the attached marked up Technical Specifications, the attached marked-up PL&S document and the separately transmitted marked-up FSAR revisions, and supports a no significant hazards consideration pursuant to 10CFR50.92, (c), criteria.

Decreasing the lead/lag compensation from 12/3 to 5/3 is predicted to increase the OPDT /OTDT margin to trip from 1.0% to 3.6%. The turbine runback margin would increase from 1.0% to 2.6%.

This safety evaluation addresses the FSAR Chapters 6 & 15 accidents that are affected by a revised lead/lag dynamic compensation, which are the non-LOCA accidents described herein. The LOCA related accidents (large and small break LOCA, reactor vessel and loop LOCA blowdown forces, post-LOCA long term core cooling subcriticality, and hot leg switchover to prevent boron precipitation), and the Containment Integrity accidents are not affected by revised lead/lag compensations. The steam generator tube rupture (SGTR) accident is not adversely affected by the revised lead/lag compensation. The UFSAR SGTR analysis was performed to evaluate the radiological consequences resulting from the double ended rupture of a single tube in one steam generator. The major factors that affect the extent of radioactive release and the resultant offsite radiation doses for an SGTR are the amount of fuel defects (level of reactor coolant contamination), the primary-to-secondary mass transfer through the ruptured tube, and the steam released from the faulted steam generator to the atmosphere. The time of reactor trip is important for offsite doses since the amount of radioactivity discharged to the atmosphere is directly impacted by the duration of steam release via the steam generator safety and/or power operated relief valves (coincident loss of offsite power or failure of the condenser dump system assumption). Tube rupture recovery actions are assumed to be completed within 30 minutes of accident initiation, consequently, an earlier trip is conservative per current SGTR methods. For an SGTR event, reactor trip can occur either on OPDT/OTDT or low pressurizer pressure. The change in lead/lag compensation would effect the OPDT and OTDT setpoint algorithm and result in an increase in the time of reactor trip. However, this will not adversely effect the radiological consequences of a SGTR accident as the SGTR accident analysis trips upon low pressurizer pressure. The change in OPDT/OTDT lead lag compensation will not affect the pressurizer low pressure trip setpoint. The doses from the SGTR will remain well within the limits as defined in 10CFR100 (25 rem whole body and 300 rem thyroid). Additionally, in order to predict the margin to trip improvement for the lead / lag relaxation, an OPDT/OTDT margin parametric study was performed.

The curves showing the results of the parametric study for the margin to trip are included in Attachment 1, the PL&S mark-ups are included in Attachment 2, and the Technical Specification mark-ups are included in Attachment 3.

NON-LOCA

BACKGROUND

Overpower ΔT turbine runback alarms are intermittently occurring on Sequoyah Nuclear Plant (SQN) Unit 1 and Unit 2 during steady-state plant operation. A change is being made to increase the operating margin to the overpower ΔT turbine runback alarm.

The change is a decrease in the ratio of the time constants in the lead-lag compensator on the reactor coolant system (RCS) ΔT. The time constants in the lead-lag compensator on RCS ΔT are being relaxed from 12/3 to 5/3.

LICENSING BASIS

The lead-lag compensated RCS ΔT signals and lag compensated hot and cold leg temperature signals are modeled in SQN licensing-basis safety analyses which rely on the reactor protection system OPΔT and OTΔT reactor trip functions for primary protection. Table 1 lists the SQN safety analyses which rely on the OTΔT or OPΔT reactor trip functions for primary protection.

Table 1
SQN Safety Analyses which Trip on OTΔT or OPΔT

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 15.2.2)
Loss of External Electrical Load and/or Turbine Trip (UFSAR 15.2.7)
Uncontrolled Boron Dilution (UFSAR 15.2.4)
Accidental Depressurization of the Reactor Coolant System (UFSAR 15.2.12)
Steamline Break with Coincident Rod Withdrawal at Power (WCAP-12504)
Steamline Break Mass/Energy Outside Containment (WCAP-10961)

ANALYSES/EVALUATIONS

IMPACT OF THE TIME CONSTANT CHANGES TO THE RCS ΔT LEAD-LAG COMPENSATOR

Table 2 presents the analysis values of the RCS ΔT lead-lag compensator time constants in the SQN licensing-basis safety analyses.

Table 2

Lead-Lag Time Constants in the RCS ΔT Lead-Lag Compensator for the SQN Safety Analyses which Trip on OTΔT or OPΔT

Event	Lead (seconds)	Lag (seconds)
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	12	3
Loss of External Electrical Load and/or Turbine Trip	0	0
Uncontrolled Boron Dilution	12	3
Accidental Depressurization of the Reactor Coolant System	0	0
Steamline Break with Coincident Rod Withdrawal at Power	12	3
Steamline Break Mass/Energy Outside Containment	0	0

As shown in Table 2, three of the SQN safety analyses, Loss of External Electrical Load and/or Turbine Trip, Accidental Depressurization of the RCS, and Steamline Break Mass/Energy Outside Containment model a 0-0 lead-lag. Use of the 0-0 lead-lag compensation is more limiting than the 5-3 lead-lag compensation and provides a bounding analysis model. This is because the 0-0 compensation results in a slower reactor trip on OTΔT and OPΔT. Since these three events are analyzed with a conservative lead-lag model, there is no adverse impact on the analysis results with the implementation of the 5-3 lead-lag. Consequently, the UFSAR conclusions for the Loss of External Electrical Load and/or Turbine Trip, Accidental Depressurization of the Reactor Coolant System, and Steamline Break Mass/Energy Outside Containment events continue to remain valid.

The remaining Table 2 analyses, Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, Uncontrolled Boron Dilution, and Steamline Break with Coincident Rod Withdrawal at Power, assume a 12-3 lead-lag. Use of the 12-3 lead-lag compensation is less limiting than the 5-3 lead-lag compensation and does not provide a bounding analysis model. This is because the 5-3 lead-lag compensation results in a slower reactor trip on OTΔT and OPΔT. The three events which are adversely impacted by the change in lead-lag time constants are evaluated as follows.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The Rod Withdrawal at Power event is analyzed with a variety of reactivity insertion rates at 10%, 60%, and 100% of rated thermal power. Depending on the case analyzed, either the High Neutron Flux or OTΔT reactor trip occurs. This event was conservatively reanalyzed with 0-0 lead-lag time constant values. Based on the results of the analysis, all applicable safety analysis criteria continue to be satisfied and the UFSAR conclusions remain valid.

Uncontrolled Boron Dilution

The Boron Dilution event is analyzed to identify the amount of time available for operator or automatic mitigation of an inadvertent boron dilution prior to complete loss of shutdown margin. This transient is required to be considered for Sequoyah for refueling (Mode 6), startup (Mode 2) and power operation (Mode 1). The subcritical modes of operation: hot standby (Mode 3), hot shutdown (Mode 4), and cold shutdown (Mode 5) are not affected by the lead-lag time constant because Sequoyah adheres to the Westinghouse interim operating procedures (Reference 5) which do not rely on the OTΔT or OPΔT reactor trips for event mitigation. Following these administrative procedures ensures that 15 minutes of operator action time is available for mitigating an uncontrolled boron dilution event while in Mode 4 or 5 and operating on the residual heat removal (RHR) system. Operating in Mode 3 with at least one reactor coolant pump in operation is less limiting than operating in Mode 4 and 5 on RHR due to increased mixing and decreased localized dilution. The Mode 6 analysis is unaffected by the lead-lag time constant change because it is administratively precluded by the limitations of the Technical Specification 3/4.9.1 The Mode 2 analysis is unaffected as it relies on a source range reactor trip to initiate operator action to mitigate a boron dilution transient. The source range reactor trip setpoint is not affected by the OPDT/OTDT lead/lag constant revision.

The Mode 1 event is analyzed in two separate cases which assume either that the control rods are in manual mode of operation or automatic. If the control rods are in automatic, the operator would be alerted to the occurrence of a boron dilution by the rod insertion limit alarms. If the rods are in manual, the first indication may be the OTΔT reactor trip. The Mode 1 rods in manual analysis is impacted by the lead-lag time constant change because the time of reactor trip on OTΔT is taken from the Uncontrolled RCCA Bank Withdrawal at Power (RWAP). The time of reactor trip is taken from the RWAP case which has a reactivity insertion rate equal to or less than that calculated for the boron dilution event and then subtracted from the amount of time available between start of the event and loss of shutdown margin. Acceptable results were obtained when the RWAP results from the previously discussed reanalysis were used to calculate the amount of time for operator action. Consistent with the SQN UFSAR, there are over 40 minutes available between the start of the event and the complete loss of shutdown margin. Based on the results of the analysis, all applicable safety analysis criteria continue to be satisfied and the UFSAR results and conclusions remain valid.

Steamline Break with Coincident Rod Withdrawal at Power

In September of 1979, IE Information Notice 79-22 was issued by the NRC addressing a potential unreviewed safety question resulting from Control and Protection Systems interaction. One of the postulated scenarios identified was the operation of the rod control system following an inside containment steamline rupture (refer to Reference 4).

This analysis is simulated by modeling a steamline rupture and coincident withdrawal of control bank D at full power conditions. A spectrum of steamline break sizes was analyzed to determine the limiting condition. The following reactor trip functions may actuate during this postulated steamline rupture with a consequential rod withdrawal transient depending on the break size:

a. OPΔT

b. A reactor trip is generated subsequent to safety injection system and steamline isolation actuation caused by low steamline pressure.

This event was reanalyzed with the change to the 5-3 lead-lag time constant values. Based on the results of the analysis, all applicable safety analysis criteria continue to be satisfied and the UFSAR conclusions remain valid.

ASSESSMENT OF UNREVIEWED SAFETY QUESTION

Operation of Sequoyah Nuclear Plant Units 1 and 2 with the revised RCS ΔT lead-lag compensator time constants do not constitute an unreviewed safety question. This conclusion is based on the responses to the seven questions below.

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

No. As addressed in this safety evaluation, all transients affected by the proposed modifications were reanalyzed or evaluated and found to adhere to the safety analysis acceptance criteria. The proposed modification does not adversely affect the integrity of the RCS or Main Steam System pressure boundary. The operation of the plant with the revised RCS ΔT lead-lag compensator time constants do not impose any new performance requirements on the RCS. The probability of such accidents occurring remains unaffected.

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

No. Per the discussion presented in the Analyses/Evaluations section, all the applicable acceptance criteria are still met for the transients evaluated and for the events reanalyzed. Additionally, no new limiting single failure is introduced by the proposed change. Therefore, there is no potential for an increase in the consequences of an accident previously evaluated in the FSAR. The revised RCS ΔT lead-lag compensator time constants do not result in a challenge to the fission product boundaries, i.e., fuel cladding, pressure vessel and containment. Offsite doses for all events do not exceed the values reported in the FSAR.

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

No. Revising the RCS ΔT lead-lag compensator time constants do not introduce a new accident initiator mechanism. Thus, no new accident will be created.

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

No. Revised RCS ΔT lead-lag compensator time constants will not adversely affect operation of the Reactor Protection System, any of the protection setpoints, or any other device required for accident mitigation. Operation with the revised RCS ΔT lead-lag compensator time constants will not affect the probability of safety related equipment malfunctions currently evaluated in the FSAR.

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

No. As discussed in the responses to questions 2 and 4, there is no possibility of increasing consequences of a malfunction of equipment for the revised RCS ΔT lead-lag compensator time constants.

4.6 May the possibility of a malfunction of equipment important to safety different than that evaluated in the FSAR be created?

No. As discussed in question 4, revised RCS ΔT lead-lag compensator time constants will not impact any other equipment important to safety. Operation of the plant with the revised RCS ΔT lead-lag compensator time constants do not introduce any new failure modes for any equipment which are credible and have not been previously considered in the FSAR. No new performance requirements are imposed on any system or component such that any design criterion is exceeded.

4.7 Will the margin of safety as defined in the bases to any technical specification be reduced?

No. As discussed in the safety evaluation, the proposed RCS ΔT lead-lag compensator time constants will not invalidate any of the conclusions presented in the UFSAR accident analyses. Therefore, the margin of safety, as defined in the bases to the Technical Specifications will not be decreased.

CONCLUSION

Based on the preceding evaluation, it is concluded that the acceptance criteria for the SQN Unit 1 and Unit 2 licensing-basis safety analyses continue to be satisfied and the UFSAR conclusions remain valid for the revised RCS ΔT lead-lag compensator time constants.

REFERENCES

1. Tennessee Valley Authority Sequoyah Nuclear Plant Updated Final Safety Analysis Report

2. WCAP-12504, Summary Report Process Protection System EAGLE 21 Upgrade, RTDBE, NSLB, MSS, EAM and TTD Implementation, Sequoyah Units 1 & 2, March 1990

3. WCAP-10961 Rev. 1, Steamline Break Mass/Energy Releases for Equipment Qualification Outside Containment - Report to the Westinghouse Owners Group High Energy Line Break/ Superheated Blowdowns Outside Containment Subgroup, October 1985

4. SECL-86-451, Sequoyah Steamline Rupture with Consequential Rod Withdrawal-Core Response Analysis for Control and Protection System Interaction, Sequoyah Units 1 and 2, November 1986.

5. TVA-92-226, "Improved Interim Operating Procedure for Boron Dilution in Modes 4 and 5," December 1, 1992, TVA-89-582, "Uncontrolled Boron Dilution Event Reanalysis and Interim Operating Procedure, FSAR Revision Section 15.2.4, "February 22, 1989, TVA-84-162, "Boron Dilution Concerns at Hot and Cold Shutdown, " August 22, 19984, and NS-TMA-2273, "Boron Dilution Concerns at Cold and Hot Shutdown, "July 8, 1980.

CONTROL SYSTEM

PARAMETRIC STUDY ON OPΔT / OTΔT DYNAMIC COMPENSATIONS

Background

Sequoyah Unit 2 is experiencing sporadic hot leg temperature fluctuations. These fluctuations are postulated to originate in the reactor vessel upper plenum. The hot leg temperatures are measured using fast-response RTDs installed in thermowells located 120 degrees apart in the same plane. Three RTDs are used to account for temperature streaming to provide an average temperature in the hot leg. Since the average of the three RTDs is used to represent the hot leg temperature, temperature fluctuations at any RTD can adversely affect the calculation of the average T-hot temperature. This, in turn, impacts the average temperature, Delta-T (ΔT) and the margin to the overtemperature and overpower Delta-T (OTΔT and OPΔT) trips/turbine runbacks. Since there have been OPΔT turbine runback alarms at 100%, Sequoyah Unit 2 has been operating at a reduced power (98-99.5%).

In order to recover some of OPΔT/OTΔT margin, the dynamic compensation used in the OPΔT/OTΔT setpoint margin calculations are being revised. TVA requested Westinghouse to predict the margin recovery if: the lead/lag compensations is reduced from 12/3. This report describes the results of a parametric study on the OPΔT/OTΔT margins.

Data Analysis

Coincident plant data from all three hot-leg RTDs and cold-leg RTDs are required to reproduce the OPΔT/OTΔT alarms. Since this data collection is very laborious, Westinghouse suggested using a hypothetical hot leg/cold leg RTD signal that would predict one percent OPΔT margin at 100% power. A simulation of the OPΔT/OTΔT trip system with current dynamic compensation would then be compared for several lead/lag and RTD filter time constants.

The hypothetical RTD signals are shown in Figure 1. It should be noted that there are several combinations of the temperature ramp rate and maximum amplitude that would result in a one percent OPΔT margin. The T-hot RTD signals shown in Figure 1 have a rise time of four seconds, a typical RTD time constant. If the rate of increase is higher than that simulated, a smaller temperature increase would result in OPΔT alarms/trips. Also, if the actual rate of temperature increases are lower than that simulated, the margin recovered will be more than that determined with a four second rise time.

The hypothetical T-hot signature is shown in Figure 1. The T-hot average (Figure 2) and T-avg (Figure 3) are then determined using a corresponding T-cold. The delta-T calculated from the above T-hot/T-cold variations are shown in Figure 4 along with the lead/lag compensated delta-T for a lead/lag ratio of 12/3. This lead/lag delta-T is compared with the OPΔT/OTΔT setpoints (Figures 5 and 6) to calculate the margin to OPΔT/OTΔT (Figures 7 and 8). As shown in Figure 7, the current OPΔT margin is about 1%. The same temperature fluctuations are used to calculate OPΔT margin for a lead/lag ratio of 8/3 and 5/3 (Figure 9).

Results

The results are shown in Table 1. If the lead/lag ratio is reduced from 12/3 to 5/3, the OPΔT margin increased from 1.0% to 3.6%.

Table 1

Comparison Of OPDT Margin

Lead / Lag Ratio

12/3	8/3	5/3
1.0	2.5	3.6

Attachment 1

Parametric Study Curves
For Margin To Trip

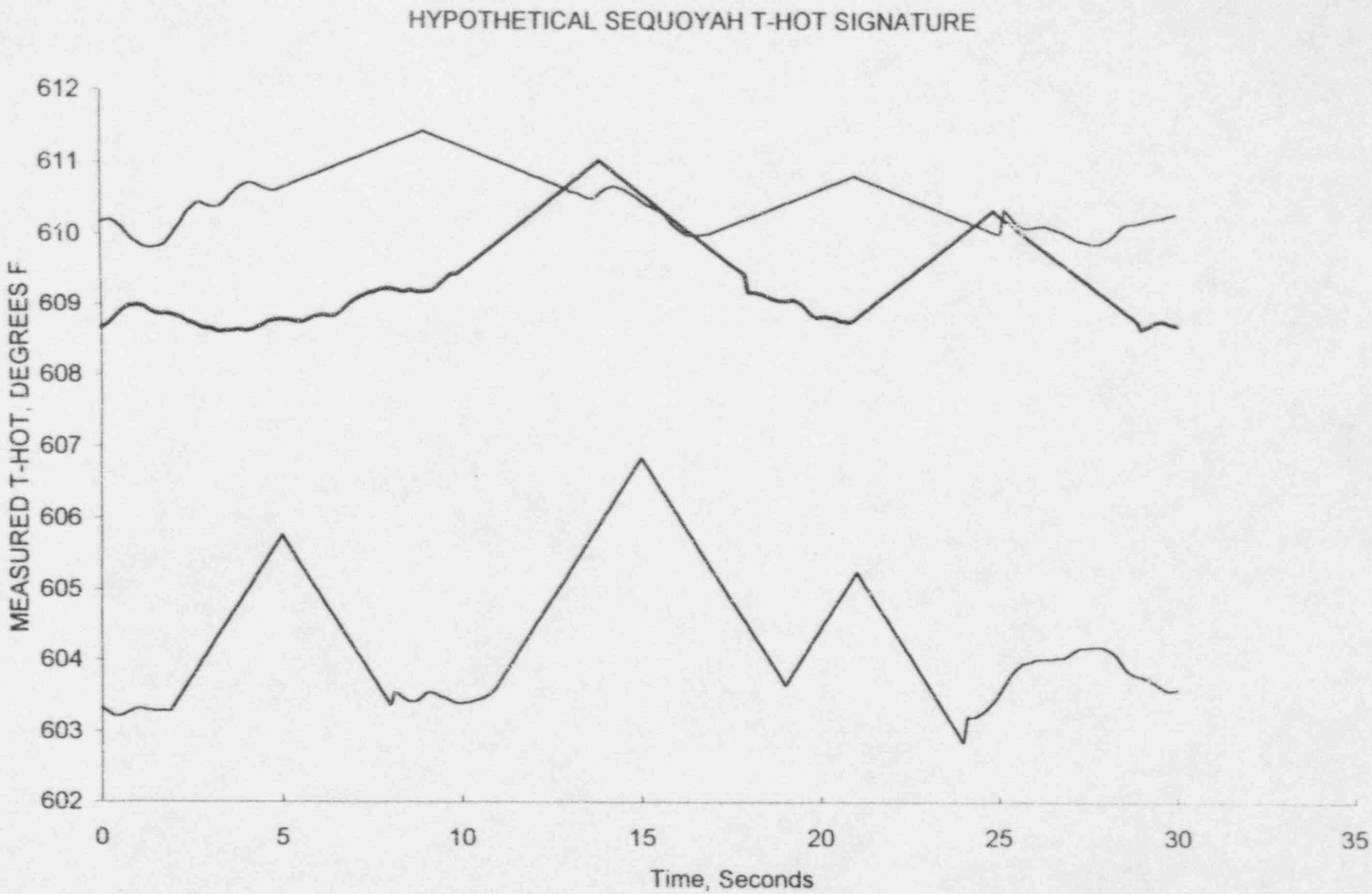

HYPOTHETICAL SEQUOYAH T-HOT SIGNATURE
MEASURED T-HOT, DEGREES F
612
611
610
609
608
607
606
605
604
603
602
0
5
10
15
20
25
30
35
Time, Seconds

FIGURE 1

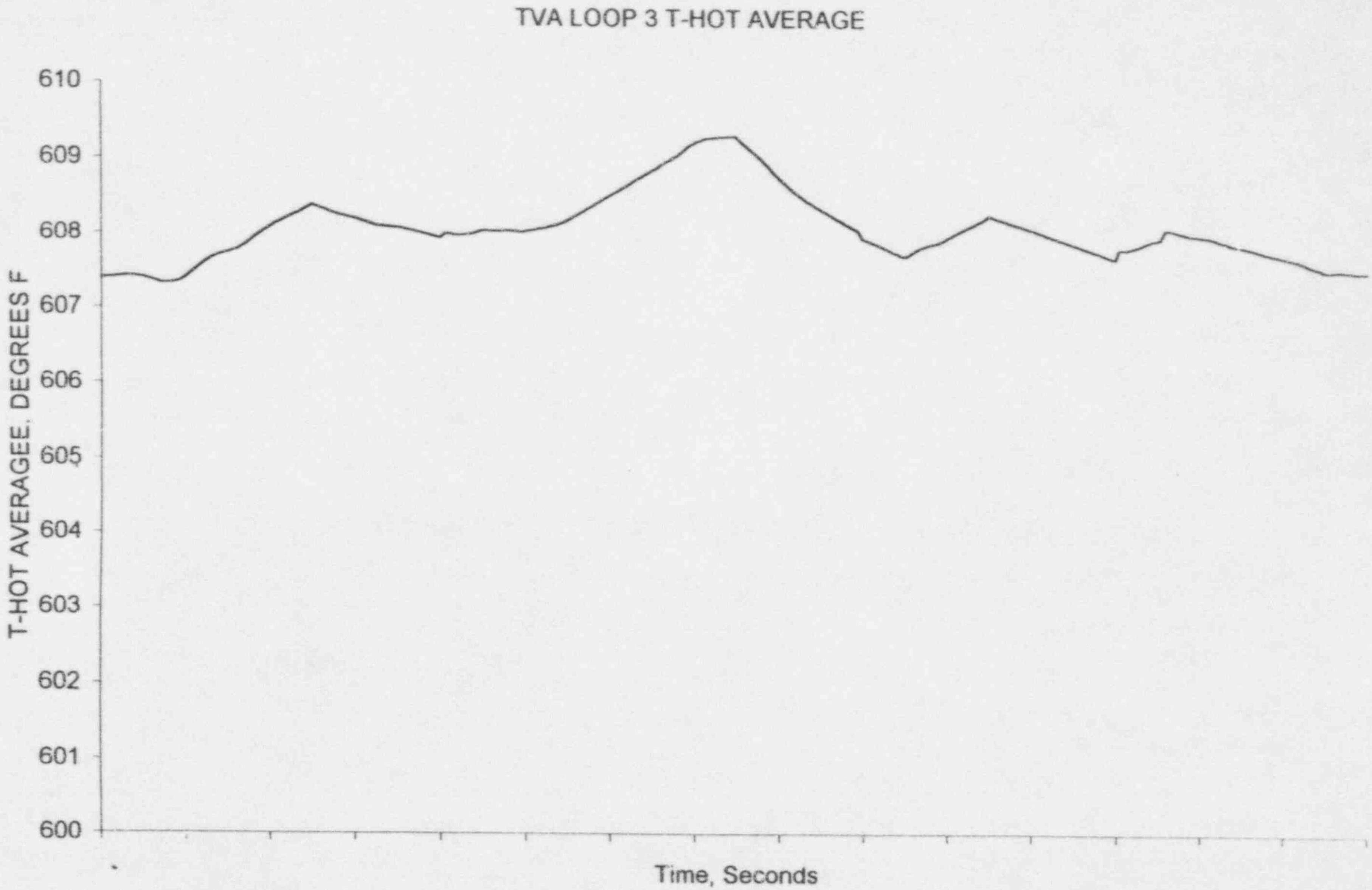

TVA LOOP 3 T-HOT AVERAGE
610
609
608
607
606
605
604
603
602
601
600
T-HOT AVERAGEE, DEGREES F
Time, Seconds

FIGURE 2

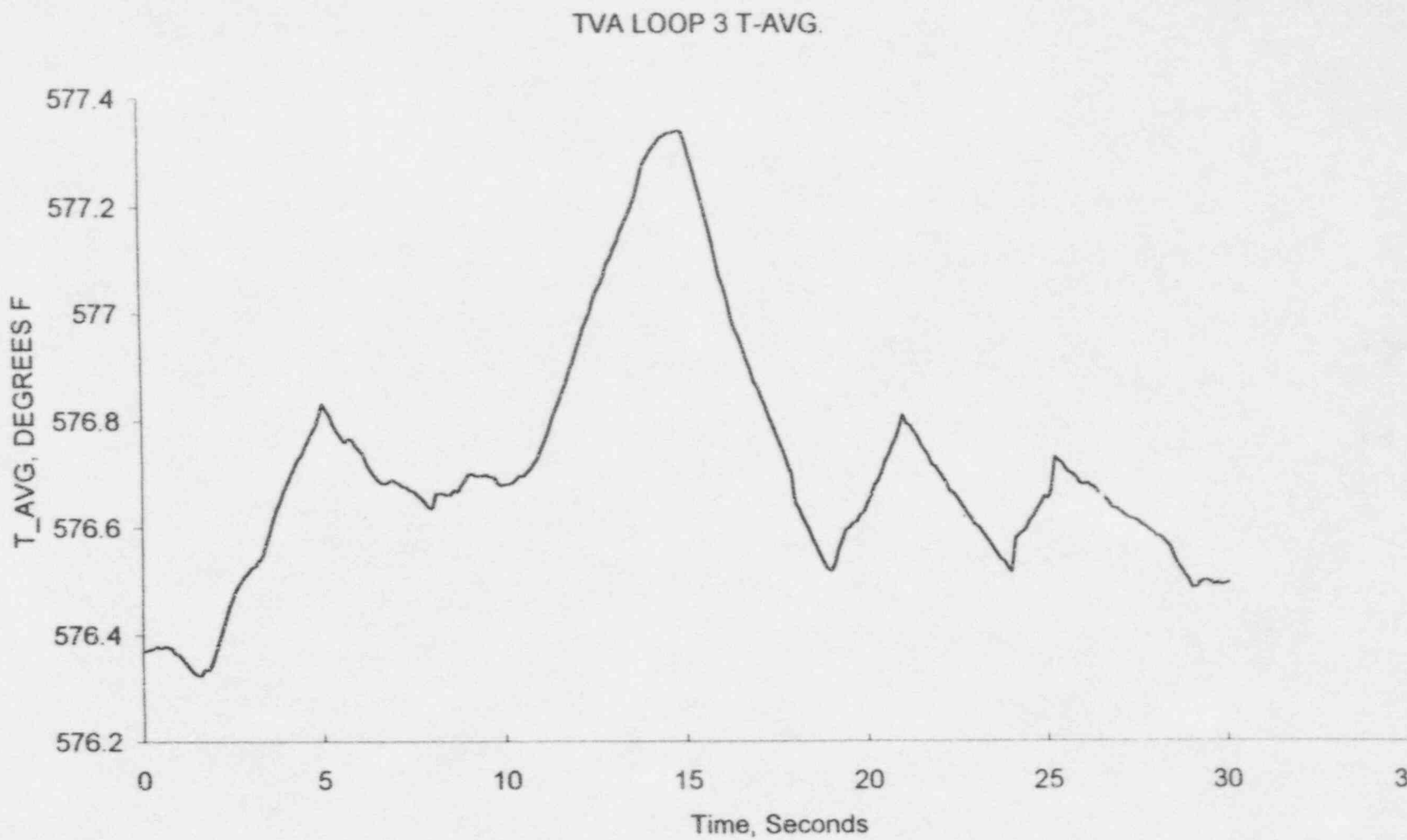

TVA LOOP 3 T-AVG.
577.4
577.2
577
576.8
576.6
576.4
576.2
T_AVG, DEGREES F
0
5
10
15
20
25
30
35
Time, Seconds

FIGURE 3

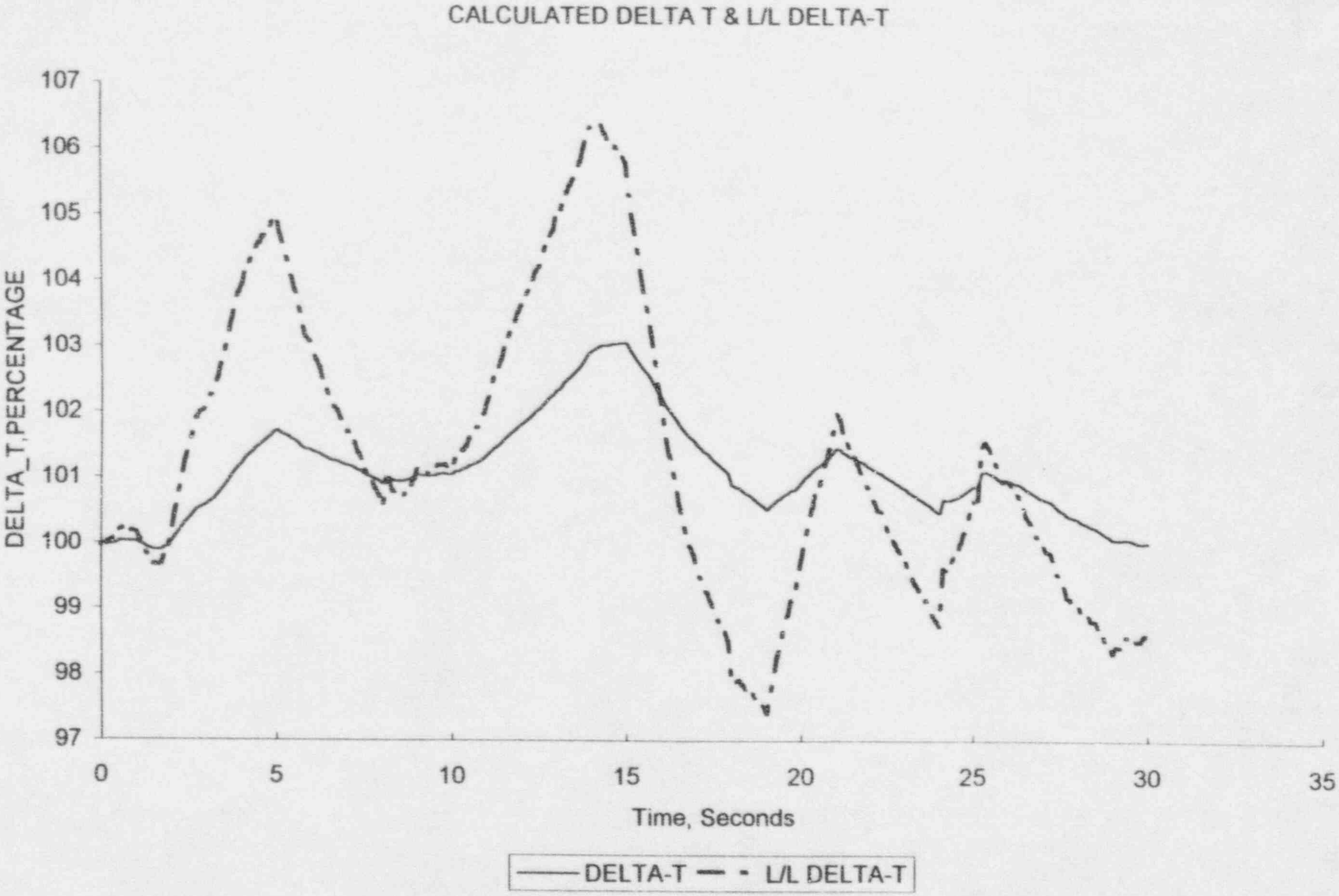
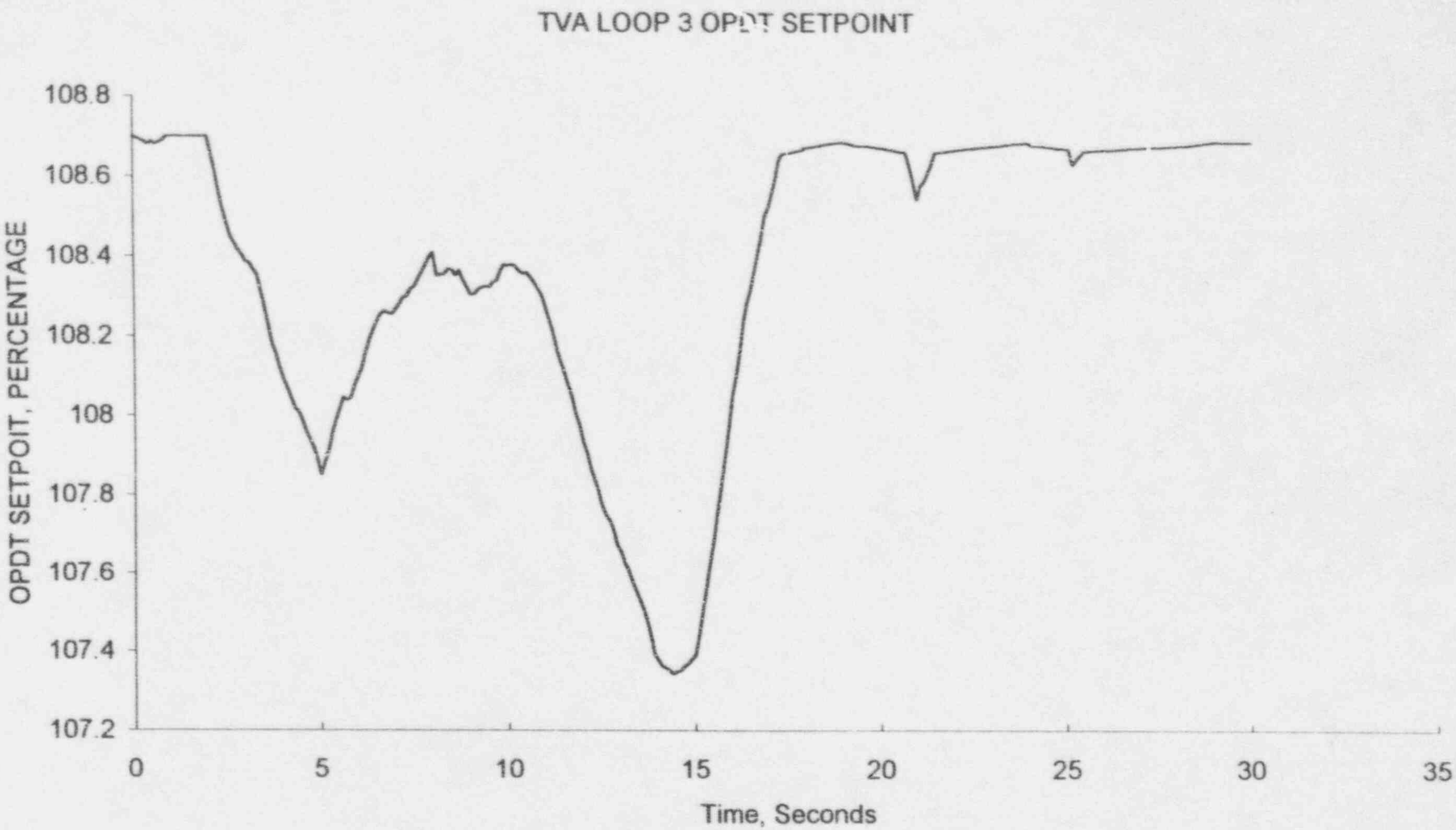

CALCULATED DELTA T & L/L DELTA-T
DELTA_T, PERCENTAGE
107
106
105
104
103
102
101
100
99
98
97
0
5
10
15
20
25
30
35
Time, Seconds
DELTA-T
L/L DELTA-T

FIGURE 4



TVA LOOP 3 OPDT SETPOINT
108.8
108.6
108.4
108.2
108
107.8
107.6
107.4
107.2
OPDT SETPOIT, PERCENTAGE
0
5
10
15
20
25
30
35
Time, Seconds

FIGURE 5

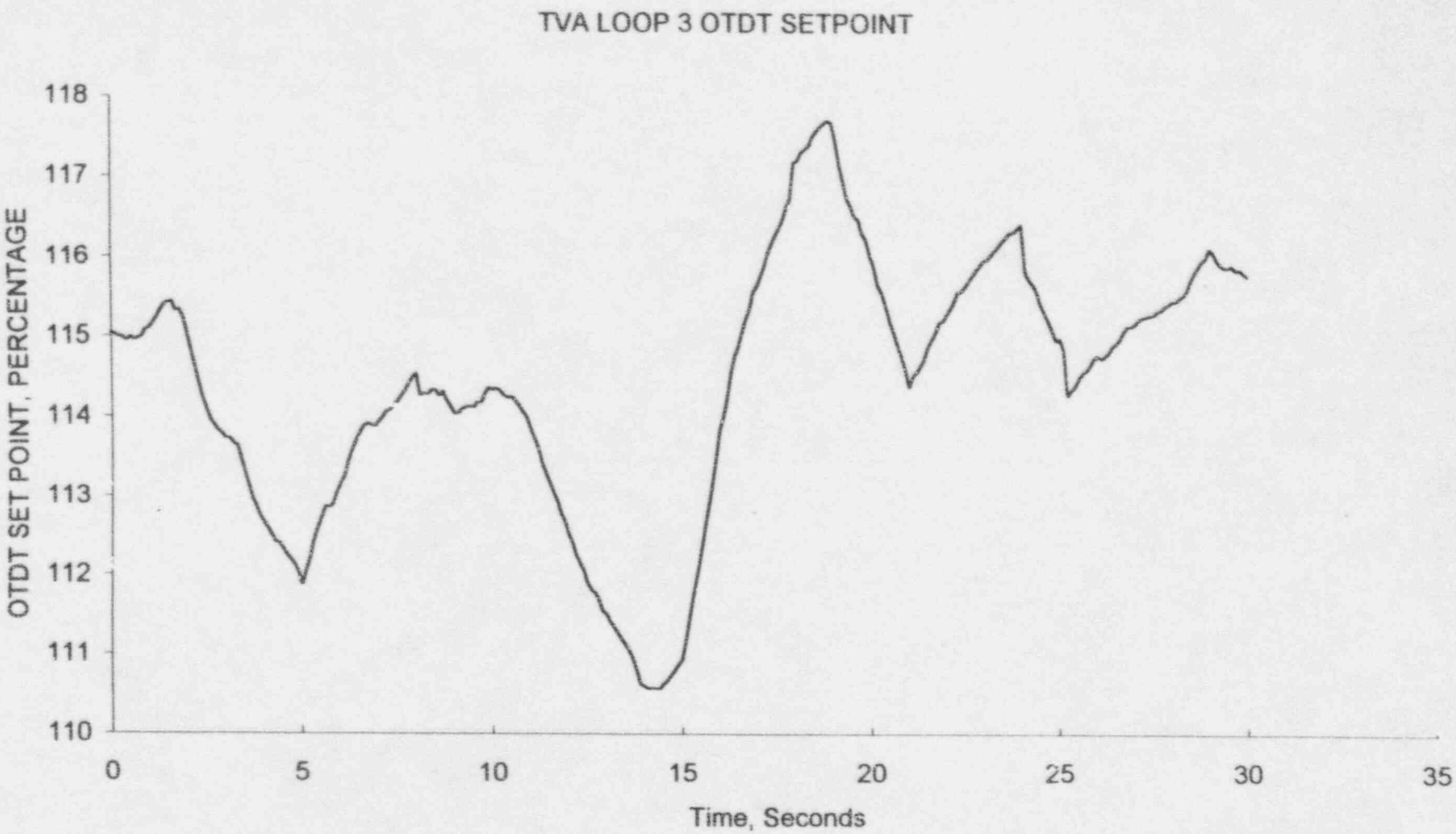

TVA LOOP 3 OTDT SETPOINT
OTDT SET POINT, PERCENTAGE
118
117
116
115
114
113
112
111
110
0
5
10
15
20
25
30
35
Time, Seconds

FIGURE 6

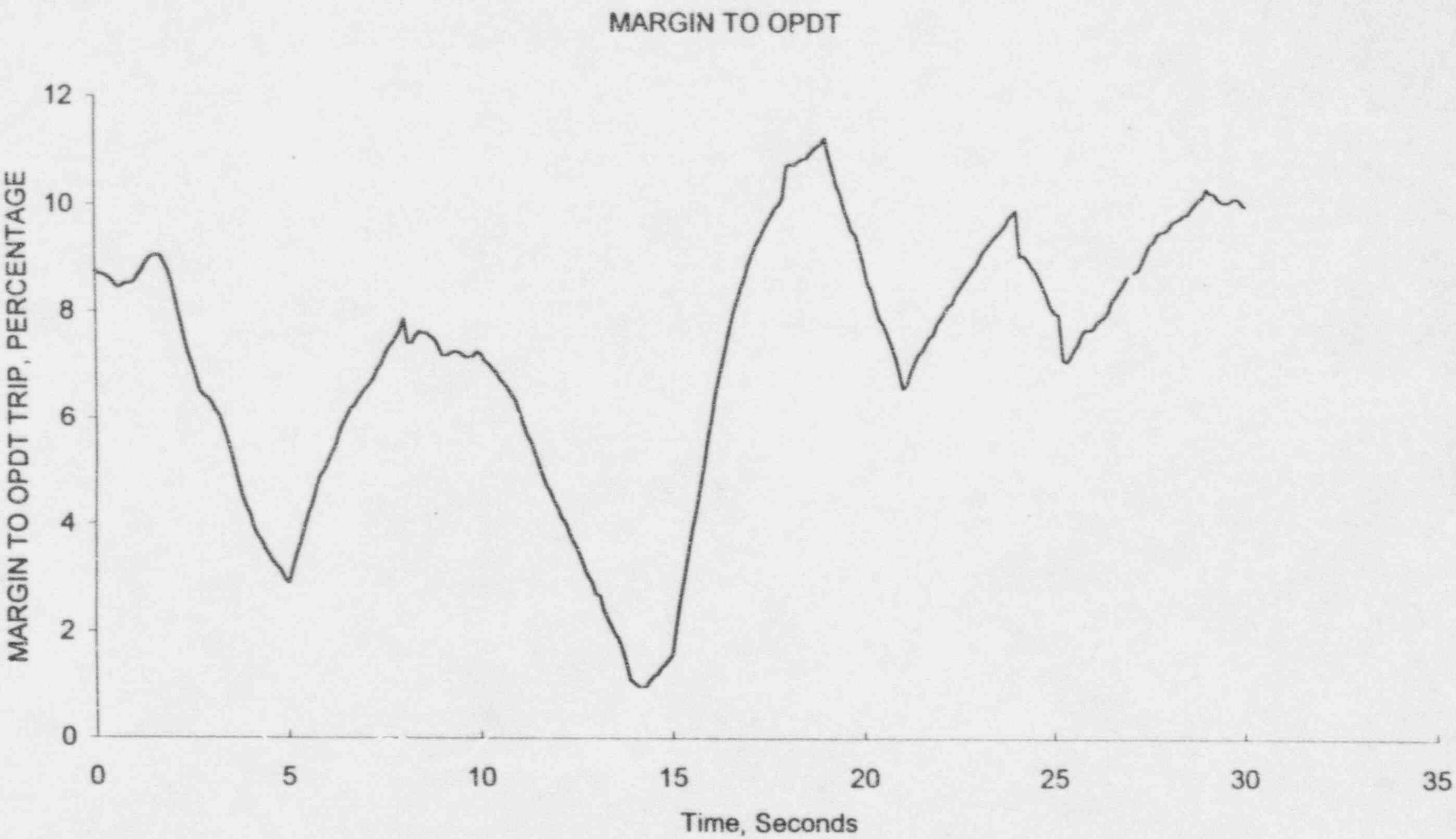

MARGIN TO OPDT
MARGIN TO OPDT TRIP, PERCENTAGE
0
2
4
6
8
10
12
0
5
10
15
20
25
30
35
Time, Seconds

FIGURE 7

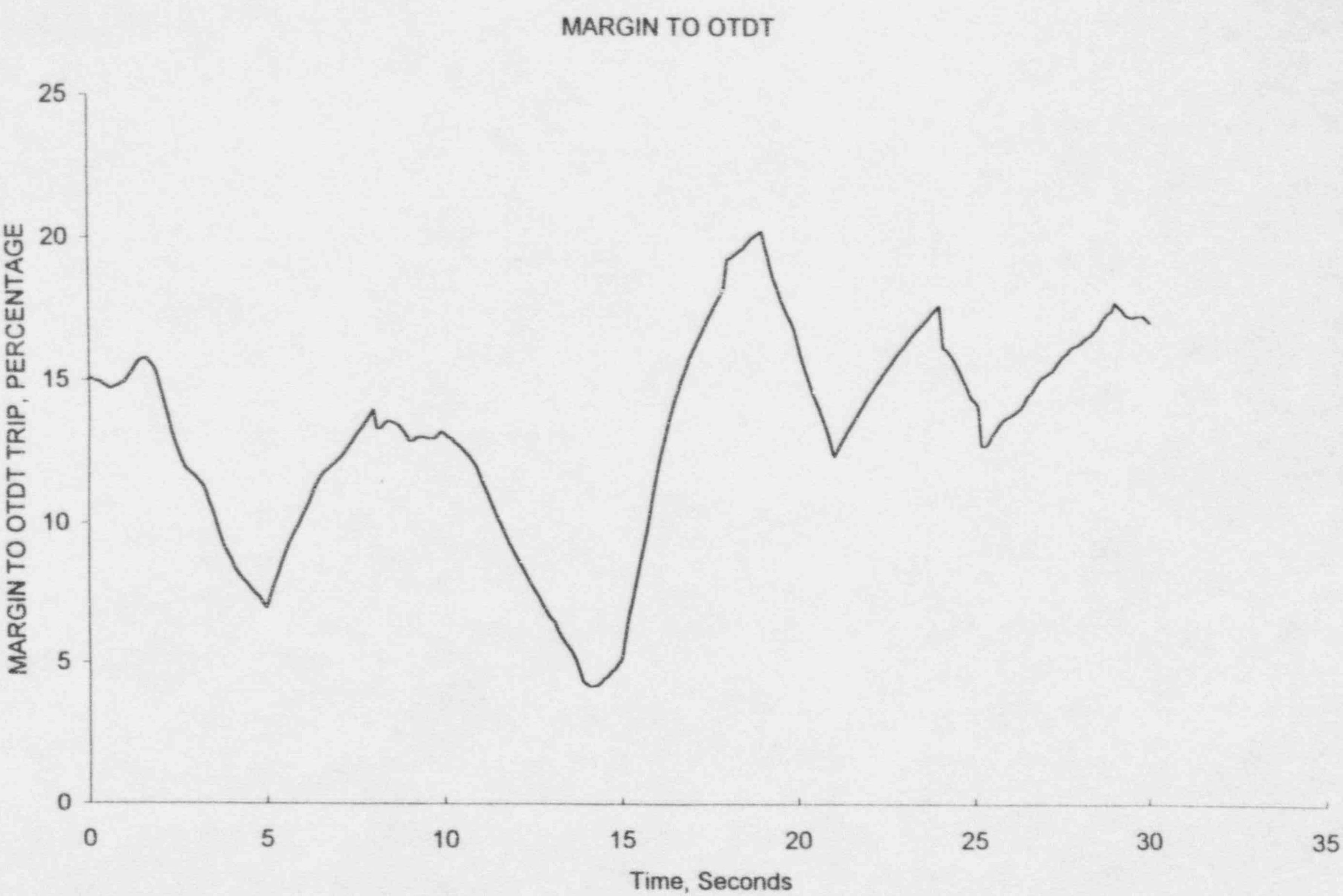

MARGIN TO OTDT
MARGIN TO OTDT TRIP, PERCENTAGE
0
5
10
15
20
25
0
5
10
15
20
25
30
35
Time, Seconds

FIGURE 8

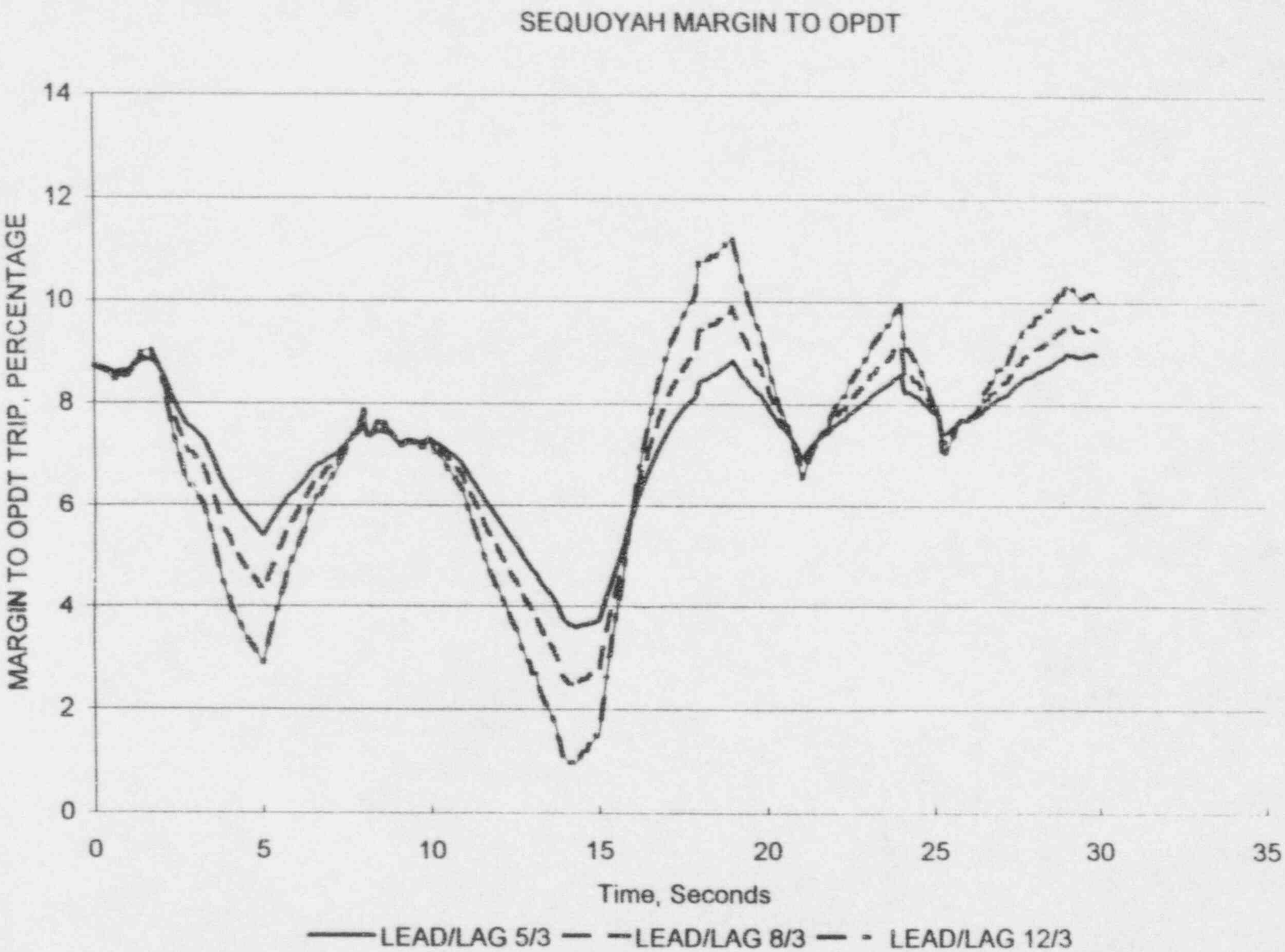

SEQUOYAH MARGIN TO OPDT
MARGIN TO OPDT TRIP, PERCENTAGE
14
12
10
8
6
4
2
0
0
5
10
15
20
25
30
35
Time, Seconds
LEAD/LAG 5/3
LEAD/LAG 8/3
LEAD/LAG 12/3

FIGURE 9

ATTACHMENT 2

PLS CHANGES

P'(PO OTSP)	P' = 2235 psig Nominal RCS operating pressure
T'(TAVGO FULL OTSP)	T' = ≤ 578.2°F Nominal T_{avg} at rated thermal power
ΔT_0 (Delta T_0)	ΔT_0 = indicated ΔT at rated thermal power (See Appendix A)

S = Laplace transform operator

T = Average Temperature, °F

P = Pressurizer Pressure, psig

$f_1(\Delta I)$ = See Item 4 below

ΔT turbine runback setpoint (OT TRB RBK) (C-3)
= ΔT reactor trip
(TB-411D, TB-421D, TB-431D, TB-441D) setpoint - 3%

Turbine runback time delay relay	on 1.5 sec
	off 28.5 sec
Turbine load reference reduction rate	200%/minute

Parameter	Four-Loop Operation
K_1 (K1 OTSP)	1.15 (115%)
K_2 (K2 OTSP)	0.011 (1.1%/°F)
K_3 (K3 OTSP)	0.00055 (0.055%/psi)
τ_1 (TAU1 Lead TAVG)	33 seconds
τ_2 (TAU2 LAG TAVG)	4 seconds
τ_4 (TAU4 Lead Delta T)	~~12~~ 5 seconds
τ_5 (TAU5 Lag Delta T)	3 seconds

2. Overpower ΔT trip and turbine runback/Block automatic rod withdrawal

ΔT reactor trip setpoint (OP Trip):

(TB-411G, TB-421G, TB-431G, TB-441G)

$$\text{Overpower } \Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_0 \left(K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 (T - T'') - f_2(\Delta I)\right)$$

where,

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = Lead-Lag compensator on measured ΔT

T = Average temperature, °F

(TAVG1 Full OPSP) T'' = Indicated average temperature at rated thermal power (calibration temperature for ΔT instrumentation, $\leq$ 578.2°F)

S = Laplace transform operator

(Delta T_0) ΔT_0 = indicated ΔT at rated thermal power

(TAU3 Rate TAVG) τ_3 = 10 seconds

(K4 OPSP) K_4 = 1.087 (108.7%)

(TAU4 Lead Delta T) τ_4 = ~~12~~ 5 seconds

(TAU5 Lag Delta T) τ_5 = 3 seconds

(K5 OPSP) K_5 = 0.02/°F for increasing T
0.0/°F for decreasing T

(K5 OPSP) K_6 = 0.0011/°F for $T > T''$
0.0/°F for $T \leq T''$

$f_2(\Delta I)$ = 0 for all ΔI

ΔT turbine runback setpoint (OP TRB RBK) (C-4)
= ΔT reactor trip
(TB-411H, TB-421H, TB 431H, TB-441H) setpoint -3%

3. Nuclear calibration for ΔT trips

During plant startup tests, all eight calibrated current signals from the power range nuclear channels are to be calibrated from core power distribution measurements such that the same signal (defined at 100% normal current) is obtained for the reference flat power condition. The reference flat power condition is defined as rated core power with nominal plant conditions and equal power in the top and bottom halves of the core (q_{top} = 50%, q_{bottom} = 50%).

ATTACHMENT 3

TECH SPEC CHANGES

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	< 10% Turbine Impulse Pressure Equivalent	< 12.4% Turbine Impulse Pressure Equivalent	R145
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Trip	< 35% of RATED THERMAL POWER	< 37.4% of RATED THERMAL POWER	R145
23. Power Range Neutron Flux - (P-10) - Enable Block of Source, Intermediate, and Power Range (low setpoint) Reactor Trips	> 10% of RATED THERMAL POWER	> 7.6% of RATED THERMAL POWER	R145
24. Reactor Trip P-4	Not Applicable	Not Applicable	
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	< 50% of RATED THERMAL POWER	< 52.4% of RATED THERMAL POWER	R145

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1+\tau_4 S}{1+\tau_5 S}\right) \leq \Delta T_o \{K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)[T-T'] + K_3(P-P') - f_1(\Delta I)\}$ R145

Where: $\frac{1+\tau_4 S}{1+\tau_5 S}$ = Lead-lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT, τ_4 ≥ 12 → 5 secs, τ_5 ≤ 3 sec

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.15

K_2 = 0.011

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1, & τ_2	=	Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 33 secs., ≥ τ_2 = 4 secs. ≤
T	=	Average temperature °F
T'	≤	578.2°F (Nominal T_{avg} at RATED THERMAL POWER)
K_3	=	0.00055
P	=	Pressurizer pressure, psig
P'	=	2235 psig (Nominal RCS operating pressure)
S	=	Laplace transform operator (sec^{-1})

R145

R145

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 29 percent and + 5 percent $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1 + \tau_4 S}{1 + \tau_5 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)[T - T'] + K_3(P - P') - f_1(\Delta I)\right\}$ R132

where: $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = Lag compensater on measured ΔT

≥

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT, τ_4 = ~~12~~ secs,
τ_5 = 3 secs ≤

5

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.15 R21

K_2 = 0.011

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation R132

τ_1, & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 33 secs.,
τ_2 = 4 secs. ≤

≥

T = Average temperature °F

T' $\leq$ 578.2°F (Nominal T_{avg} at RATED THERMAL POWER) R132

K_3 = 0.00055

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)