ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-327

(TVA-SQN-TS-92-02)

LIST OF AFFECTED PAGES

Unit 1

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
13. Steam Generator Water Level--Low-Low			
a. RCS Loop ΔT Equivalent to Power $\leq$ 50% RTP	RCS Loop ΔT variable input $\leq$ 50% RTP	RCS Loop ΔT variable input $\leq$ trip setpoint + 2.5% RTP #	R145
Coincident with Steam Generator Water Level -- Low-Low (Adverse)	$\geq$ 15.0% of narrow range instrument span	$\geq$ 14.4% of narrow range instrument span	R155
and Containment Pressure - EAM	$\leq$ 0.5 psig	$\leq$ 0.6 psig	R145
or			
Steam Generator Water Level -- Low-Low (EAM)	$\geq$ 10.7% of narrow range instrument span	$\geq$ 10.1% of narrow range instrument span	R155
with			
A time delay (T_s) if one Steam Generator is affected	$\leq T_s$ (Note 5)	$\leq$ (1.01) T_s (Note 5)	
or			
A time delay (T_m) if two or more Steam Generators are affected	$\leq T_m$ (Note 5)	$\leq$ (1.01) T_m (Note 5)	R145
b. RCS Loop ΔT Equivalent to Power > 50% RTP			
Coincident with Steam Generator Water Level -- Low-Low (Adverse)	$\geq$ 15.0% of narrow range instrument span	$\geq$ 14.4% of narrow range instrument span	R155
and Containment Pressure (EAM)	$\leq$ 0.5 psig	$\leq$ 0.6 psig	R145
or			
Steam Generator Water Level -- Low-Low (EAM)	$\geq$ 10.7% of narrow range instrument span	$\geq$ 10.1% of narrow range instrument span	R155

THE REQUIREMENT FOR THE RCS LOOP ΔT VARIABLE INPUT ALLOWABLE VALUE OF $\leq$ TRIP SETPOINT + 2.5% RTP IS REDUCED TO $\leq$ TRIP SETPOINT + 2.4% RTP DURING UNIT 1 CYCLE 6 OPERATION BECAUSE OF INCREASED RTD UNCERTAINTIES. THIS ALLOWABLE VALUE REDUCTION EXPIRES AT THE END OF UNIT 1 CYCLE 6 OPERATION.

JUL 24 1991

SEQUOYAH - UNIT 1

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}. τ_3 = 10 secs. R145

K_6 = 0.0011 for T > T" and K_6 = 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

THE REQUIREMENT FOR THE OVERTEMPERATURE ΔT ALLOWABLE VALUE OF 1.9 PERCENT ΔT SPAN ABOVE THE COMPUTED TRIP SETPOINT IS REDUCED TO 1.6 PERCENT ΔT SPAN DURING UNIT 1 CYCLE 6 OPERATION BECAUSE OF INCREASED RTD UNCERTAINTIES. THIS ALLOWABLE VALUE REDUCTION EXPIRES AT THE END OF UNIT 1 CYCLE 6 OPERATION.

* THE REQUIREMENT FOR THE OVERPOWER ΔT ALLOWABLE VALUE OF 1.7 PERCENT ΔT SPAN ABOVE THE COMPUTED TRIP SETPOINT IS REDUCED TO 1.6 PERCENT ΔT SPAN DURING UNIT 1 CYCLE 6 OPERATION BECAUSE OF INCREASED RTD UNCERTAINTIES. THIS ALLOWABLE VALUE REDUCTION EXPIRES AT THE END OF UNIT 1 CYCLE 6 OPERATION.

Amendment No. 19, 141
May 16, 1990

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1) and R(9)	1, 2, and *
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	Q	1, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	Q	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	Q	1, 2
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S(7)	R(6)	M and S/U(1)	2, 3, 4, 5, and *
7. Overtemperature Delta T	S	R **	Q	1, 2
8. Overpower Delta T	S	R **	Q	1, 2
9. Pressurizer Pressure--Low	S	R	Q	1, 2
10. Pressurizer Pressure--High	S	R	Q	1, 2
11. Pressurizer Water Level--High	S	R	Q	1, 2
12. Loss of Flow - Single Loop	S	R	Q	1
13. Loss of Flow - Two Loops	S	R	N.A.	1
14. Main Steam Generator Water Level--Low-Low				
A. Steam Generator Water Level -- Low-Low (Adverse)	S	R	Q	1, 2
B. Steam Generator Water Level -- Low-Low (EAM)	S	R	Q	1, 2
C. RCS Loop ΔT	S	R **	Q	1, 2
D. Containment Pressure (EAM)	S	R	Q	1, 2

R145

R145

** FOR UNIT 1 CYCLE 6 OPERATION, IN LIEU OF A CHANNEL CALIBRATION A TECHNICAL EVALUATION HAS BEEN PERFORMED AS DELINEATED IN TECHNICAL SPECIFICATION CHANGE REQUEST 92-02.

Amendment No. 54, 141
MAY 16 1990

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
6. AUXILIARY FEEDWATER			
a. Manual	Not Applicable	Not Applicable	
b. Automatic Actuation Logic	Not Applicable	Not Applicable	
c. Main Steam Generator Water Level--Low-Low			
i. RCS Loop ΔT Equivalent to Power $\leq$50% RTP	RCS Loop ΔT variable input $\leq$50% RTP	RCS Loop ΔT variable input $\leq$ trip setpoint +2.5% RTP#	R145
Coincident with Steam Generator Water Level--Low-Low (Adverse) and	$\geq$15.0% of narrow range instrument span	$\geq$14.4% of narrow range instrument span	R155 R145
Containment Pressure-EAM or	$\leq$0.5 psig	$\leq$0.6 psig	
Steam Generator Water Level--Low-Low (EAM) with	$\geq$10.7% of narrow range instrument span	$\geq$10.1% of narrow instrument span	R155
A time delay (T_S) if one Steam Generator is affected or	$\leq T_S$ (Note 5, Table 2.2-1)	$\leq$ (1.01) T_S (Note 5, Table 2.2-1)	R145
A time delay (T_m) if two or more Steam Generators are affected	$\leq T_m$ (Note 5, Table 2.2-1)	$\leq$ (1.01) T_m (Note 5, Table 2.2-1)	

THE REQUIREMENT FOR THE RCS LOOP ΔT VARIABLE INPUT ALLOWABLE VALUE OF $\leq$ TRIP SETPOINT +2.5% RTP IS REDUCED TO $\leq$ TRIP SETPOINT +2.4% RTP DURING UNIT 1 CYCLE 6 OPERATION BECAUSE OF INCREASED RTD UNCERTAINTIES. THIS ALLOWABLE VALUE REDUCTION EXPIRES AT THE END OF UNIT 1 CYCLE 6 OPERATION.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Main Steam Generator Water Level-Low-Low				
1. Steam Generator Water Level -- Low-Low (Adverse)	S	R	Q	1, 2, 3
2. Steam Generator Water Level -- Low-Low (EAM)	S	R	Q	1, 2, 3
3. RCS Loop ΔT	S	R *	Q	1, 2, 3
4. Containment Pressure (EAM)	S	R	Q	1, 2, 3
d. S.I.	See 1 above (all SI surveillance requirements)			
e. Station Blackout	N.A.	R	N.A.	1, 2, 3
f. Trip of Main Feedwater Pumps	N.A.	N.A.	R	1, 2
g. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	M	1, 2, 3
h. Auxiliary Feedwater Suction Transfer Time Delays	N.A.	R	N.A.	1, 2, 3
7. LOSS OF POWER				
a. 6.9 kv Shutdown Board - Loss of Voltage				
1. Start Diesel Generators	S	R	M	1, 2, 3, 4
2. Load Shedding	S	R	N.A.	1, 2, 3, 4

R145

R145

* FOR UNIT 1 CYCLE 6 OPERATION, IN LIEU OF A CHANNEL CALIBRATION A TECHNICAL EVALUATION HAS BEEN PERFORMED AS DELINEATED IN TECHNICAL SPECIFICATION CHANGE REQUEST 92-02.

R145

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-327

(TVA-SQN-TS-92-02)

DESCRIPTION AND JUSTIFICATION FOR
REVISION OF OVERTEMPERATURE DIFFERENTIAL TEMPERATURE,
OVERPOWER DIFFERENTIAL TEMPERATURE, AND REACTOR COOLANT
SYSTEM (RCS) LOOP DIFFERENTIAL TEMPERATURE ALLOWABLE
VALUES AND CALIBRATION REQUIREMENT FOR RCS
RESISTANCE TEMPERATURE DETECTORS

Description of Change

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Unit 1 Technical Specifications (TSs) to revise the reactor trip allowable values for overtemperature (OT) differential temperature (ΔT) and overpower (OP) ΔT from 1.9 and 1.7 percent ΔT span respectively to 1.6 percent ΔT span. These changes affect Items 7 and 8 of Table 2.2-1 through the associated notes 3 and 4 of this table. In addition, the reactor coolant system (RCS) loop ΔT allowable value of +2.5 percent reactor thermal power (RTP) for Item 13.a in Table 2.2-1 and Item 6.c.i in Table 3.3-4 will be revised to +2.4 percent RTP. The channel calibration requirements for the RCS resistance temperature detectors (RTDs) associated with OTΔT, OPΔT, and RCS loop ΔT functions, Items 7, 8, and 14.C respectively of Table 4.3-1 and Item 6.c.3 of Table 4.3-2, will be revised to utilize the technical evaluation presented in this TS change request in lieu of an RTD cross-calibration. These changes are requested for Unit 1 Cycle 6 operation only and will expire at the end of this fuel cycle.

Reason for Change

During the start-up of Unit 1 for Cycle 6 operation, TVA performed RTD cross-calibrations to verify accuracy. At the upper-temperature plateaus, the data for this calibration effort appeared to be skewed. TVA investigated and identified errors in the data as a result of test-instrumentation application. The use of a data logger on the Eagle 21 analog test points had introduced a random bias on the temperature reading. This bias skewed the data such that the intent of the cross-calibration to verify a tolerance of ±0.5 degree Fahrenheit (F) could not be achieved. Westinghouse Electric Corporation was requested to provide an evaluation of the RTD accuracy based on Cycle 5 factory RTD data and maximum expected RTD uncertainties. This evaluation concluded that the RTDs would meet an accuracy of ±1.2 degrees F for the beginning of the Cycle 6 fuel cycle. With the RTD accuracy of ±1.2 degrees F, Westinghouse determined that the effect would require a slight reduction in the OTΔT, OPΔT, and RCS loop ΔT allowable values to support the safety analysis. TVA discussed these changes with NRC and agreed to administratively control these values while processing this TS change for the reasons described above. The Westinghouse evaluation also serves as the basis that SQN Unit 1 has met the intent of the 18-month channel calibration requirement for OTΔT, OPΔT, and RCS loop ΔT functions that utilize RCS RTD inputs. This evaluation will be used for Unit 1 Cycle 6 operation instead of the RTD cross-calibration data that was skewed as previously described.

The proposed changes are needed to properly reflect more conservative allowable values that are consistent with the safety analysis and to prevent an unnecessary midcycle unit shutdown to perform an additional RTD cross-calibration.

Justification for Change

The trip setpoint and allowable value for the OTΔT reactor trip function have been designed to protect the reactor from reaching an unacceptably low departure from nucleate boiling ratio. The RCS hot- and cold-leg RTD temperatures are inputs to the process circuitry that calculates this trip setpoint on a continuous basis. The RTD inputs are provided to the circuitry with one set of temperature measurements for each RCS loop. The OPΔT reactor trip function protects against excessive power that could exceed the power per foot rating of the fuel rods. This setpoint is also calculated on a continuous basis and utilizes RTD temperature inputs from each RCS loop.

The reactor trip function of the low-low steam generator (S/G) level protects against the loss of reactor heat sink in the event of loss of feedwater to the S/Gs. With the reactor protection system upgrade that included the installation of the Westinghouse Eagle 21 system, SQN implemented a trip time delay (TTD) feature associated with the low-low S/G level reactor trip. This TTD feature provides a variable time delay for actuation of the reactor trip based on reactor power levels below 50 percent RTP. The RCS hot- and cold-leg RTD temperatures are utilized to determine this power level by calculations performed with process circuitry. This logic also applies to the engineered safety feature that initiates auxiliary feedwater on low-low S/G level. This function is designed to protect against the loss of reactor heat sink as well.

The functions described above utilize the RCS hot- and cold-leg RTD temperature measurements to perform their design functions. The channel calibration requirements for these functions ensure that the RTDs have not experienced instrumentation drift or failures that would invalidate the assumptions used in the safety analysis. In addition, control and protection functions for low T_{avg} signal for feedwater isolation, Permissive P-12 for steam dump, pressurizer level control, RCS T_{avg} measurements for rod control system automatic operation, and calculated value of RCS flow measurement uncertainty also utilize these RTD temperature measurements. Enclosure 4 provides Westinghouse's safety evaluation that addresses the effect on each of these functions in relationship to the safety analysis for SQN. This evaluation provides the justification for the proposed TS changes needed as a result of the increased RCS RTD calibration uncertainty.

The conclusion of the Westinghouse safety evaluation is that with the proposed TS changes and the normalization of appropriate ΔT and T_{avg} values, the RTD performance will be acceptable with the original RTD factory calibration constants for the remainder of the current Unit 1 Cycle 6 fuel cycle. Additionally, the Westinghouse evaluation provides the basis for the conclusion that the RCS RTDs are accurate to ±1.2 degrees F at the beginning of the Unit 1 Cycle 6 operation. These combine to provide the technical basis for meeting the intent of the 18-month channel calibration for OTΔT, OPΔT, and RCS loop ΔT functions. This evaluation will be used in lieu of an RCS RTD cross-calibration. TVA operates the SQN units within the constraints specified by Westinghouse for ΔT and T_{avg} normalization and therefore ensures the validity of this evaluation.

Environmental Impact Evaluation

The proposed change request revises the OTΔT, OPΔT, and RCS loop ΔT allowable values and associated channel calibration requirements to ensure the operability of the associated reactor trip and engineered safety feature functions. This change does not involve an unreviewed environmental question because operation of SQN Unit 1 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 CHANGE

SEQUOYAH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-327

(TVA-SQN-TS-92-02)

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.

   As documented in the attached Westinghouse Electric Corporation evaluation (Enclosure 4), this TS change will provide different allowable values for overtemperature (OT) differential temperature (ΔT), overpower (OP) ΔT, and reactor coolant system (RCS) loop ΔT and documents the technical evaluation used in lieu of the associated channel-calibration requirements. These changes ensure that the accident analysis for SQN remains valid and that the associated surveillances remain in frequency. The impact on control and protection functions considering these changes is shown not to increase the probability of any accident because no accident initiator is affected. With these changes, the consequences of an accident have been evaluated to ensure no increase in the radiological consequences would result. Control and protection functions will continue to operate acceptably to maintain all the assumptions in the SQN safety analysis for the remainder of the Unit 1 Cycle 6 operation. The proposed TS changes will compensate for the increased calibration uncertainty of the RCS resistance temperature detectors (RTD).

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

   These TS changes only affect protection functions that required additional conservatisms to support the SQN safety analysis because of the increase in RTD calibration uncertainty. All other effects resulting from the calibration uncertainty have been evaluated by Westinghouse and verified not to impact the intended functions or operability of control or protection features. Accordingly, no new accident scenarios have been created by these changes to the TSs or the calibration uncertainty.

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

   The increase in the RTD calibration uncertainty did not adversely impact the safety-analysis limits or nominal trip setpoints of any protection function. To accommodate the increase in RTD uncertainty, the TS allowances for OTΔT, OPΔT, and RCS loop ΔT setpoints are reallocated. The SQN safety analysis remains valid with these changes and does not involve a reduction in the margins of safety.

Enclosure 4

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-327

(TVA-SQN-TS-92-02)

WESTINGHOUSE ELECTRIC CORPORATION
SAFETY EVALUATION CHECK LIST (SECL) 91-459, REVISION 2

SECL 91-459 Rev 2
Customer Reference No(s).

____

Westinghouse Reference No(s).

____

WESTINGHOUSE
SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) SEQUOYAH UNIT 1

2) CHECK LIST APPLICABLE TO: INCREASED RTD CALIBRATION UNCERTAINTY FROM
(Subject of Change) +/-0.5°F to +/-1.2°F

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 - 10CFR50.59(a)(1)

(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

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 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 question (3.4) of Part A or any of the questions in Part B cannot be answered in the negative, based on the written safety evaluation, the Change review requires an application for license amendment as stated in 10CFR50.59(c) and must be submitted to the NRC pursuant to 10CFR50.90.

5) REMARKS:

The answers given in Sections 3 and 4, Parts A and B of the Safety Evaluation Checklist are based on the attached safety evaluation.

SEE ATTACHED SAFETY EVALUATION

FOR FSAR UPDATE

Section:________ Pages:________ Tables:__________ Figures:__________

Reason for / Description of Change:

Changes to Technical Specifications: See attachment

Table 2.2-1	Allowable values for RCS Loop Delta-T Equivalent to Power, Overtemperature Delta-T, and Overpower Delta-T
Table 3.3-4	Allowable Value for RCS Loop Delta-T Equivalent to Power

Approvals

Prepared by: U L Brown Date: 2/18/92
U. L. Brown

Group Manager: R J Sterdis Date: 2/19/92.
R. J. Sterdis

Sequoyah Unit 1
Increase in RTD Calibration Uncertainty
From +/-0.5°F to +/-1.2°F

SAFETY EVALUATION

1.0 INTRODUCTION

Reference 1 provides documentation of the Westinghouse calculations performed to determine the setpoints for the Sequoyah units with the Eagle-21 protection system process racks installed. One of the assumptions made in the Westinghouse calculations is that the Hot Leg and Cold Leg Narrow Range RTDs are calibrated to within +/-0.5°F. This calibration is normally performed during plant heat-up after refueling via the cross-calibration process.

This safety evaluation addresses the impact on the safety analysis performed for Sequoyah Unit 1 Cycle 6 of increasing the RTD calibration uncertainty from +/-0.5°F to +/-1.2°F. The total RTD uncertainty included in the protection system setpoints is +/-1.2°F (calibration) plus +/-0.7°F (drift over the cycle) for a total of +/-1.9°F. The Reference 1 analysis was performed assuming an RTD calibration uncertainty of +/-0.5°F and drift of +/-0.7°F for a total RTD uncertainty of +/-1.2°F.

This safety evaluation provides the basis for the changes recommended to the Technical Specifications provided in the attachment and supports a "No Significant Hazards" determination pursuant to 10 CFR 50.92.

2.0 LICENSING BASIS

The evaluation performed by Westinghouse is based on comparison of the results of calculations performed with the revised RTD calibration uncertainties to results and limits noted in Reference 1 and the Sequoyah Unit 1 Technical Specifications. The Total Allowance noted in Reference 1 was compared with the Channel Statistical Allowance values calculated for the revised RTD calibration uncertainties. Changes in the Allowable Values noted in the Technical Specifications were made where necessary.

The analyses given in Chapters 6 and 15 of the Sequoyah Unit 1 FSAR make explicit allowances for instrumentation errors for some of the reactor protection system setpoints. In addition, an allowance is made on the initial average reactor coolant system (RCS) temperature, pressure, and power as described in FSAR Section 15.1.2.2. The following protection and control functions are affected by increases in the Narrow Range RTD calibration accuracy: Overtemperature Delta-T (OTDT) reactor trip, Overpower Delta-T (OPDT) reactor trip, Vessel Delta-T Equivalent to Power (input to Steam Generator Level Trip Time Delay), Low Tavg signal for feedwater isolation, Permissive P-12 for steam dump, Pressurizer Level for pressurizer level control, the rod control system, and the calculated value of RCS flow measurement uncertainty.

3.0 EVALUATION

3.1 Instrumentation and Control Systems Evaluation

While +/-0.7°F drift is utilized in this analysis, industry experience with RTDs show that little drift occurs. From the readings of the MMI of T-hot and T-cold, no anomalous indications are noted and a reasonable assurance that the Sequoyah Unit 1 RTDs have not experienced excessive drift during the time period can be made. Therefore, the +/-0.7°F assumed for drift uncertainty can be considered conservative and would still be valid. The original factory calibration is +/-0.2°F and assures that +/-0.5°F could be verified through cross calibration. Due to the vintage of the RTDs at TVA, and industry experience with other RTDs of the same vintage, recent installation with RTDs calibrated in place using the cross-calibration process have shown that the factory calibration uncertainty in addition to any installation errors have typically yielded RTDs within the +/-0.5°F range. As discussed above, it is reasonably expected that Sequoyah Unit 1 RTD performance is consistent with industry experience. Therefore, given that the factory calibration constants are utilized, +/-0.5°F uncertainty along with the additional +/-0.7°F is judged to be sufficient to cover any potential deviation from average temperature for any RTD as an initial condition for beginning of Cycle 6 of Unit 1.

3.2 Setpoint Study Evaluation

Westinghouse performed uncertainty calculations identical to those performed for Reference 1 with one change, the SCA (Sensor Calibration Accuracy) was modified to reflect the change from +/-0.5°F to +/-1.2°F. After evaluation of the results of the calculations, the following conclusions were reached:

1) The rod control uncertainty increases from +/-4.5°F to +/-4.6°F.
2) The Tavg - Low, Low uncertainty increased by +/-0.2% of span, but because of the process rack change-out to Eagle instrumentation and the use of the New Steam Line Break Protection System, there is no protection system impact.
3) The uncertainty of the baseline RCS Flow Calorimetric remains unchanged since TVA is basing the Sequoyah RCS flow uncertainty on the use of an earlier cycle RCS flow calorimetric.
4) The RCS flow indicated value on the control board or process computer is affected by a small increase in the Tcold RTD uncertainty and its subsequent affect on cold leg density. The affect of this increase is noted and accounted for within the round-off for the indicated flow uncertainty.
5) The uncertainty for the Loss of Flow Reactor Trip function is increased slightly due to the affect of the increased RTD error on Tcold. Subsequent effects on cold leg density are accounted for within the round-off of this function's total uncertainty value.

6) The Pressurizer Level Control uncertainty is determined to be +/-5.01% of level span which is essentially the same as that assumed in the safety analysis as an initial condition (+/-5.00%).
7) Overtemperature Delta-T was evaluated and it was determined that the Allowable Value decreases from 1.9% of Delta-T span to 1.6% of Delta-T span, as noted on the attached Technical Specification page markups.
8) Overpower Delta-T was evaluated and it was determined that the Allowable Value decreases from 1.7% of Delta-T span to 1.6% of Delta-T span, as noted on the attached Technical Specification page markups.
9) Vessel Delta-T Equivalent to Power was evaluated and it was determined that it is appropriate to reduce the Allowable Value from 2.5% RTP (1.7% span) to 2.4% RTP (1.6% span). This is noted on the attached Technical Specification page markups.

The probability and confidence level of the results of the protection function uncertainty calculations reported in Reference 1 have been previously identified to be accurate with a 95% probability at a 95% confidence level. One assumption which supported this assertion was that a valid RTD cross calibration is performed which confirms an RTD uncertainty. While cross calibration data in this case can not be used to confirm an uncertainty, Westinghouse has no reason to believe that the present installation is not accurate with a 95% probability at a 95% confidence level, given the discussion under Section 3.1.

3.3 Transient Analysis Evaluation

The Chapter 6 and Chapter 15 safety analyses were performed assuming that, at steady state full power, the average RCS temperature was equal to the nominal value plus 5.5°F. This allowance is currently described in the FSAR as +/-4.0°F for rod controller accuracy and +/-1.5°F for accident evaluation. Based on the increase in the rod control uncertainty (+/-4.5°F to +/-4.6°F) due to the increased RTD uncertainty, a portion of the accident evaluation margin can be allocated to maintain the same RCS temperature assumption in the safety analyses. Since the average temperature remains unchanged, the conclusions in the FSAR would remain valid when the effects of the estimated RTD errors are included for the Chapter 6 and Chapter 15 events which do not rely on the protection functions identified above.

To accommodate the increase in the RTD uncertainty, the Technical Specifications Total Allowances for the OTDT and OPDT reactor trip setpoints are reallocated. The reallocation ensures consistency with the FSAR Chapter 6 and 15 safety analysis assumptions for the OTDT and OPDT reactor protection functions. The modifications to the existing Technical Specifications which include the increased RTD calibration errors are provided in Attachment 1.

The Chapter 6 and Chapter 15 safety analyses do not take credit for the Low Tavg feedwater isolation or Permissive P-12 steam dump functions. Therefore, the impact of increased RTD calibration uncertainty on these functions does not affect the conclusions in the FSAR.

Pressurizer Level uncertainty for pressurizer level control is increased to +/-5.01% due to the increased RTD calibration uncertainty. An increased pressurizer level can affect those analyses which use overfill of the pressurizer as an acceptance criterion. Overfill of the pressurizer is used as an acceptance criterion for the Loss of Normal Feedwater (FSAR Section 15.2.8), Loss of Off-Site Power to the Station Auxiliaries (FSAR Section 15.2.9), and Rod Withdrawal at Power (FSAR Section 15.2.2) events. These analyses have been reviewed and it has been determined that the increased pressurizer level would not cause the pressurizer to overfill. Thus, the increased pressurizer level uncertainty is acceptable with respect to the non-LOCA accident analyses.

3.4 Operational Evaluation

3.4.1 Delta-T Normalization at Power

One of the requirements for operability for the OTDT and OPDT protection functions is the normalization to loop specific, indicated vessel Delta-T values. It has been recently noted that with low leakage cores, the indicated vessel Delta-T values change in the non-conservative direction with increasing burnup (i.e., the indicated Delta-T decreases as the cycle progresses). This is attributed to the change in radial power distribution with burnup. Renormalization of the protection channels should be performed if any indicated loop Delta-T is more than 1% (0.6°F) smaller than the calibration value. Likewise, if the indicated loop Delta-T is more than 2% (1.2°F) larger than the calibration value, the channels should be renormalized. The process of renormalization is complicated by feedwater venturi fouling which has been observed in the past at Sequoyah. One of the more significant inputs to compensate for the magnitude of the fouling has been indicated vessel Delta-T. Since feedwater venturi fouling results in a conservative full power Delta-T, Westinghouse recommends the use of either, a value of feedwater flow which is not compensated for the affects of venturi fouling, or feedwater flow which has been compensated by other means in order to renormalize the Delta-T used for the protection channels.

3.4.2 Tavg Input to OTDT and OPDT

A secondary effect (which is not necessarily small in magnitude) is on the Tavg input to these protection functions. Generally, no effort is made to revise the T' and T" values to reflect the indicated, loop specific Tavg values at 100% RTP. However, Westinghouse makes the following recommendations in order to assure that the protection functions will respond appropriately, in light of the fact that the indicated Tavg has a possibility of being in error in the non-conservative direction and that the indicated Thot value will decrease with increasing burnup resulting in a change in indicated Tavg in the non-conservative direction.

It is recommended that T' and T" be modified to reflect the indicated, loop specific value for Tavg at 100% RTP at the beginning of the cycle. In addition, the T' and T" values should be modified to reflect loop specific, indicated values when the indicated Delta-T values are redetermined and the protection functions rescaled. This is a conservative position with respect to T' and is a restatement of the existing requirement for T" (see page 2-10 of the Sequoyah Unit 1 Technical Specifications).

When the plant is in automatic rod control with the reference Tavg set to the design nominal full power Tavg, the T' and T" values would not need modification. This is due to the control system maintaining the Tavg value to the same as that noted for T' and T" in the plant Technical Specifications. When in manual rod control, the operation of the plant should emulate the auto controller, i.e., maintain the plant about the same reference Tavg. Indicated temperature may vary above and below the reference Tavg as in auto control. A variance of +/-1.5°F about the reference Tavg would be consistent with the auto control system deadband.

3.5 Other

Other safety related areas within the Westinghouse scope of supply have been reviewed and it was determined that none of these are affected by the increased RTD calibration uncertainty. These areas include:

- mechanical components and systems integrity
- containment response
- radiological consequences
- LOCA and LOCA related transients including,
  - large and small break LOCA
  - LOCA hydraulic forces
  - post-LOCA long term core cooling
  - rod ejection mass releases
  - hot leg switchover time to prevent boron precipitation
- steam generator tube rupture
- probabilistic risk assessment
- emergency operating procedures

4.0 <u>ASSESSMENT OF UNREVIEWED SAFETY QUESTION</u>

The use of an increased RTD calibration uncertainty is evaluated below in accordance with the criteria of 10 CFR 50.59 as required to demonstrate that no unreviewed safety question is involved.

4.1 The probability of an accident previously evaluated in the FSAR will not be increased.

The increase in the RTD calibration uncertainty does not adversely impact any control and protection functions. The setpoint study calculations performed confirmed that in all cases the nominal trip setpoints were preserved. Only a reallocation of Total Allowances was necessary to accommodate the increased calibration uncertainty. No initiators of any accidents are affected.

4.2 The consequences of an accident previously evaluated in the FSAR will not be increased.

The setpoint study calculations showed that the nominal trip setpoints for the protection functions were unaffected by the increase in RTD calibration uncertainty. Also, the increased uncertainty on pressurizer level was shown to be acceptable with respect to the consequences of events which use overfill of the pressurizer as an acceptance criterion. Thus, system performance with respect to the control of radiological consequences is not adversely impacted.

4.3 The possibility of an accident which is different than any already evaluated in the FSAR has not been created.

No new limiting single failures are introduced due to the increase in the RTD calibration uncertainty. No previously incredible event is now made credible as a result of this change. All control and protection functions continue to be operable and capable of performing their intended functions.

4.4 The probability of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

Neither system performance nor safety system functions are adversely impacted by the increase in the RTD calibration uncertainty. This activity has no affect on non-safety related equipment or functions which could in turn affect safety related equipment performance. This change does not affect the initiators of any event.

4.5 The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

The increase in the RTD calibration uncertainty does not adversely impact any contol or protection functions. The setpoint calculations demonstrated that nominal trip setpoints are preserved. The affect on pressurizer control can be accommodated in the analysis of those events which use overfill of the pressurizer as an acceptance criterion without any increase in radiological consequences. System performance is not compromised by this change.

4.6 The possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR will not be created.

No new failure modes for any equipment are created by the increase in the RTD calibration uncertainty. The affected control and protection functions continue to remain operable and capable of performing their intended functions.

4.7 The margin of safety as defined in the bases to any Technical Specifications will not be reduced.

The increase in the calibration uncertainty of the RTDs did not adversely impact the Safety Analysis Limit or Nominal Trip Setpoint of any protection function. To accommodate the increase in the RTD uncertainty, the Technical Specifications Total Allowances for the OTDT and OPDT reactor trip setpoints and Vessel Delta-T Equivalent to Power setpoints are reallocated. The conclusions of the FSAR found in Chapters 6 and 15 continue to remain valid.

5.0 CONCLUSION

An increased RTD calibration uncertainty of +/-1.2°F plus +/-0.7°F for cycle drift has been evaluated to determine the effect on the Sequoyah Unit 1 Chapter 6 and Chapter 15 accident analyses. It is concluded that, with the attached changes to the Technical Specifications, the increased RTD calibration uncertainty does not constitute an unreviewed safety question.

Westinghouse finds that continued operation with the original factory calibration constants is acceptable for the remainder of the current Unit 1 Cycle 6 fuel cycle.

6.0 REFERENCES

1. WCAP-11239 Rev 5, "Westinghouse Setpoint Methodology for Protection Systems - Sequoyah Units 1 & 2 - Eagle-21 Version," March 1991.

7.0 ACKNOWLEDGEMENTS

The following personnel contributed to this safety evaluation:
R. B. Miller, J. F. Mermigos, W.H. Moomau, C. R. Tuley and J.T. Doman.

ATTACHMENT 1

TECH SPEC CHANGES

SEQUOYAH

SEQUOYAH - UNIT 1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
13. Steam Generator Water Level--Low-Low			
a. RCS Loop ΔT Equivalent to Power ≤ 50% RTP	RCS Loop ΔT variable input ≤ 50% RTP	RCS Loop ΔT variable input ≤ trip setpoint +2.4% ~~+ 2.5%~~ RTP	R145
Coincident with Steam Generator Water Level -- Low-Low (Adverse) and	≥ 15.0% of narrow range instrument span	≥ 14.4% of narrow range instrument span	R155
Containment Pressure - EAM	≤ 0.5 psig	≤ 0.6 psig	R145
or			
Steam Generator Water Level -- Low-Low (EAM)	≥ 10.7% of narrow range instrument span	≥ 10.1% of narrow range instrument span	R155
with			
A time delay (T_s) if one Steam Generator is affected	≤ T_s (Note 5)	≤ (1.01) T_s (Note 5)	
or			
A time delay (T_m) if two or more Steam Generators are affected	≤ T_m (Note 5)	≤ (1.01) T_m (Note 5)	R145
b. RCS Loop ΔT Equivalent to Power > 50% RTP			
Coincident with Steam Generator Water Level -- Low-Low (Adverse) and	≥ 15.0% of narrow range instrument span	≥ 14.4% of narrow range instrument span	R155
Containment Pressure (EAM)	≤ 0.5 psig	≤ 0.6 psig	R145
or			
Steam Generator Water Level -- Low-Low (EAM)	≥ 10.7% of narrow range instrument [illegible]	≥ 10.1% of narrow range instrument	R155

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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}, τ_3 = 10 secs. R145

K_6 = 0.0011 for T > T" and K_6 = 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~~ 1.6 percent ΔT span.

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than ~~1.7~~ 1.6 percent ΔT span. R145

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
6. AUXILIARY FEEDWATER			
a. Manual	Not Applicable	Not Applicable	
b. Automatic Actuation Logic	Not Applicable	Not Applicable	
c. Main Steam Generator Water Level--Low-Low			
i. RCS Loop ΔT Equivalent to Power $\leq$50% RTP	RCS Loop ΔT variable input $\leq$50% RTP	RCS Loop ΔT variable input < trip setpoint ~~+2.5% RTP~~ +2.4%	R145
Coincident with Steam Generator Water Level--Low-Low (Adverse) and	$\geq$15.0% of narrow range instrument span	~~$\geq$14.4%~~ of narrow range instrument span	R155 R145
Containment Pressure-EAM or	$\leq$0.5 psig	$\leq$0.6 psig	R145
Steam Generator Water Level--Low-Low (EAM) with	$\geq$10.7% of narrow range instrument span	$\geq$10.1% of narrow instrument span	R155
A time delay (T_S) if one Steam Generator is affected or	$\leq T_S$ (Note 5, Table 2.2-1)	$\leq$ (1.01) T_S (Note 5, Table 2.2-1)	R145
A time delay (T_m) if two or more Steam Generators are affected	$\leq T_m$ (Note 5, Table 2.2-1)	$\leq$ (1.01) T_m (Note 5, Table 2.2-1)	