WESTINGHOUSE CLASS 3

WCAP 13193

RTD BYPASS ELIMINATION LICENSING REPORT
FOR
SEABROOK NUCLEAR STATION

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1.0 INTRODUCTION

Westinghouse Electric Corporation has been contracted by New Hampshire Yankee to remove the existing Resistance Temperature Detector (RTD) Bypass System described in Final Safety Analysis Report (FSAR, Ref. 1) Section 5.4.3.2 and replace this hot leg and cold leg temperature measurement method with fast-response thermowell mounted RTDs installed in the reactor coolant loop piping. This report is submitted for the purpose of supporting the four loop operation of Seabrook utilizing the new thermowell mounted RTDs.

1.1 HISTORICAL BACKGROUND

Prior to 1968, PWR designs had been based on the assumption that the hot leg temperature was uniform across the pipe. Therefore, placement of the temperature instruments was not considered to be a factor affecting the accuracy of the measurement. The hot leg temperature was measured with direct-immersion RTDs extending a short distance into the pipe at one location. By the late 1960s, as a result of accumulated operating experience at several plants, the following problems associated with direct immersion RTDs were identified:

- o Temperature streaming conditions - the incomplete mixing of the coolant leaving regions of the reactor core at different temperatures - produces significant temperature gradients within the pipe.

- o The reactor coolant loops required cooling and draining before the RTDs could be replaced.

The RTD bypass system was designed to resolve these problems; however, operating plant experience has now shown that operation with the RTD bypass loops has created it's own obstacles such as:

- o Plant shutdowns caused by excessive primary leakage through valves, flanges, etc., or by interruptions of bypass flow due to valve stem failure.

- Increased radiation exposure due to maintenance on the bypass line and to crud traps which increase radiation exposure throughout the loop compartments.

The proposed temperature measurement modification has been developed in response to both sets of problems encountered in the past. Specifically:

- Removal of the bypass lines eliminates the components which have been a major source of plant outages as well as Occupational Radiation Exposure (ORE).

- Three thermowell-mounted hot leg RTDs provide an average measurement (equivalent to the temperature measured by the bypass system) to account for temperature streaming.

- Use of thermowells permits RTD replacement without draining the reactor coolant loops.

Following is a detailed description of the effort required to perform this modification.

1.2 MECHANICAL MODIFICATIONS

The individual loop temperature signals required for input to the Reactor Control and Protection System will be obtained using RTDs installed in each reactor coolant loop.

1.2.1 Hot Leg

a) The hot leg temperature measurement on each loop will be accomplished using three fast response, narrow range, dual element RTDs mounted in thermowells. These RTD's as well as those on the cold leg, will be provided with a connection head. Both elements of each hot leg RTD are wired to the appropriate process protection rack where the second RTD input is a spare. To accomplish the sampling function of the RTD bypass manifold system and minimize the need for additional hot leg piping penetrations, the thermowells will be located within two of the three existing RTD bypass manifold scoops (Figure 1.2-1). Due to a structural interference, the third RTD will be located in an independent boss (Figure 1.2-2). On loops A, B, and D the independent boss is located in the same cross-sectional plane as the existing scoops, but offset 30° from the unused location. On loop C, the boss will be relocated to a position approximately 12 inches upstream of the existing scoops at approximately 105° from TDC. The unused scoops (the 120° location on loops A & C and the 240° location in loops B & D) will be capped. These 3 RTDs will be used to obtain the hot leg temperature used for generation of reactor coolant loop differential temperature (ΔT) and average temperature (T-avg).

b) This modification will not affect the single wide range RTD currently installed near the entrance of each steam generator. This RTD will continue to provide the hot leg temperature used to monitor reactor coolant temperature during startup, shutdown, and post accident conditions.

1.2.2 <u>Cold Leg</u>

a) One fast response, narrow range, dual-element RTD will be located in each cold leg at the discharge of the reactor coolant pump (as replacements for the cold leg RTDs located in the bypass manifold). This RTD will measure the cold leg temperature which is used to calculate reactor coolant loop ΔT and T_{avg}. The existing cold leg RTD bypass penetration nozzle will be modified (Figure 1.2-2) to accept the RTD thermowell. One element of the RTD will be considered active and the other element will be held in reserve as a spare. Both elements of the cold leg RTDs will be wired to the appropriate process protection rack where the second RTD input is a spare.

b) This modification will not affect the single wide range RTD in each cold leg currently installed at the discharge of the reactor coolant pump.

This RTD will continue to provide the cold leg temperature used to monitor reactor coolant temperature during startup, shutdown, and post accident conditions.

1.2.3 Crossover Leg

The RTD bypass manifold return line will be capped at the nozzle on the crossover leg.

1.3 ELECTRICAL MODIFICATIONS

1.3.1 Function

Figure 1.3-1 shows a block diagram of the modified electronics. The hot leg RTD measurements (three per loop) will be electronically averaged in the process protection system. The averaged T_{hot} signal will then be used with the T_{cold} signal to calculate reactor coolant loop ΔT and T_{avg} which are used in the reactor control and protection system. This will be accomplished by additions to the existing process control equipment.

1.3.2 Qualification

The 7300 Process Electronics modifications has been qualified to the same level as the existing 7300 electronics. RTD qualification has been verified to support New Hampshire Yankee's compliance to 10CFR50.49.

1.3.3 RTD Operability Indication

Existing control board ΔT and T_{avg} indicators and alarms provide the means of identifying RTD failures. Should the failure of a hot leg RTD be diagnosed, two methods are available for addressing the failed RTD. The preferred method is to utilize the second element of the RTD. Since both elements of each dual element RTD are wired to the appropriate process protection rack, I&C personnel can disconnect the failed element from the rack terminal strip and connect the other RTD element.

The second method is for the I&C personnel to defeat the failed hot leg RTD and rescale the electronics to average the remaining two signals and incorporate a bias based upon the hot leg streaming measured in the loop.

Should a failure of a cold leg RTD be diagnosed, the I&C personnel would disconnect the failed element from the rack terminal strip and connect the other RTD element.

a,c

Figure 1.2-1
Hot Leg RTD Scoop Modification for Fast Response RTD Installation

a,c

Figure 1.2-2
Hot Leg RTD Located in Independent Boss

a,c

Figure 1.2-3
Cold Leg Pipe Nozzle Modification for Dual Element
Fast Response RTD Installation

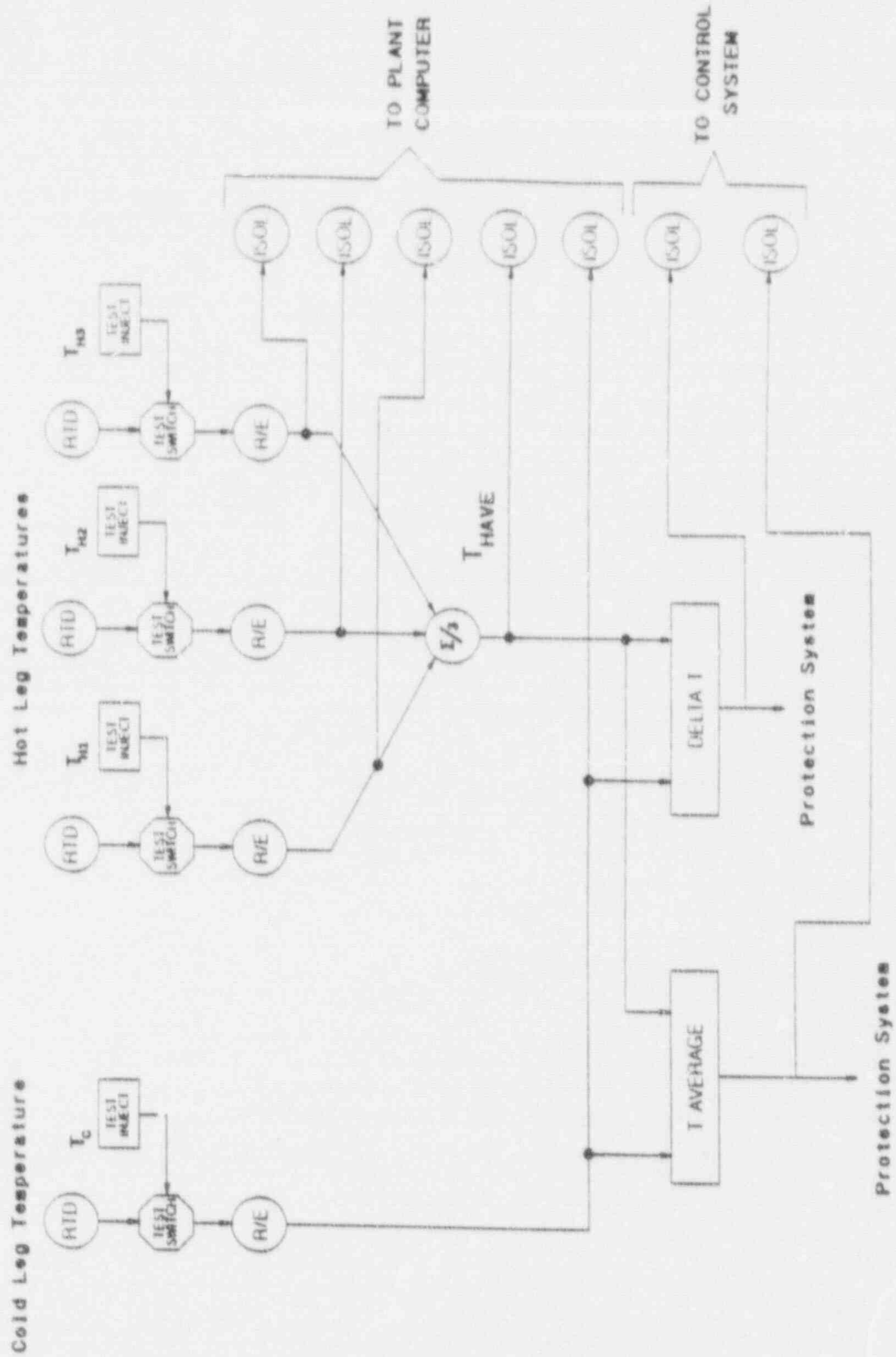

Figure 1.3-1
RTD Averaging Block Diagram
Typical for Each of 4 Channels

2.0 TESTING

There are two specific tests which have been performed to support the installation of the fast-response RTDs in the reactor coolant piping: a response time test and a hot leg temperature streaming test.

2.1 RESPONSE TIME

The RTD manufacturer, WEED Instruments, Inc., will perform time response testing of each RTD and thermowell prior to installation at Seabrook. These RTD/thermowells must exhibit a response time bounded by the values shown in Table 2.1-1. The response time for thermowell mounted RTDs has been factored into the transient analyses discussed in Section 4.0.

In addition, response time testing of the WEED RTDs will be performed in-situ. This testing will demonstrate that the WEED RTDs satisfy the response time requirement when installed in the plant.

2.2 STREAMING TEST

Past testing at Westinghouse PWRs has established that temperature stratification exists in the hot leg pipe with a temperature gradient from top to bottom of approximately [ ]^{b,c,e}. A special test program was implemented at an operating plant to confirm the temperature streaming magnitude and stability with measurements of the RTD bypass branch line temperatures on two adjacent reactor coolant loops. Specifically, it was intended to determine the magnitude of the differences between branch line temperatures, confirm the short-term and long-term stability of the temperature streaming patterns and evaluate the impact on the indicated temperature if only 2 of the 3 branch line temperatures are used to determine an average temperature. This plant specific data is used in conjunction with thermowell mounted RTD data taken from eight Westinghouse designed 4 loop plants similar in size and configuration to Seabrook to determine an appropriate temperature error for use in the safety analysis and calorimetric flow calculations. This data, as well as data taken in the ensuing years since the original test, have been used to calculate the appropriate hot leg streaming uncertainty for Seabrook.

The special test data was reduced and characterized to answer the three objectives of the test program. First, RTD bypass branch line temperature differences in the two adjacent loops were established. Also, data taken during power escalation indicated that branch line temperature differences [ ]^{b,c,e} as observed during the earlier streaming tests. Second, the streaming pattern [ ]^{b,c,e}.

In other words, the temperature gradient [ ]^{b,c,e}. This is inferred by [ ]^{b,c,e} observed between branch lines. Third, since the [ ]^{b,c,e} into the RTD averaging circuit if a hot leg RTD fails and only 2 RTDs are used to obtain an average hot leg temperature. The operator can review temperatures recorded prior to the RTD failure and determine an [ ]^{b,c,e} into the "two RTD" average to obtain the "three RTD" expected reading. A generic procedure (see Appendix B) has been provided to New Hampshire Yankee which specifies how these [ ]^{b,c,e} are to be determined. This significantly reduces the error introduced by a failed RTD. This long term stability associated with two out of three RTDs in service has been confirmed by observations at other plants.

This special test data also supports previous calculations of streaming errors determined from previous tests at other Westinghouse plants. In addition, more recent temperature data (provided by other 4 loop plants similar to Seabrook) is also consistent with the upper bound temperature gradients that characterize the special test data. Data from operating plants recently equipped with the thermowell-mounted RTDs is within the gradients used in the streaming error calculations. Data obtained during power escalation from these plants continued to indicate that the streaming differences [ ].^{b,c,e} There were no new discoveries from either the special test data or more recent operating plant data, but the new data did add a dimension previous tests did not have. The test sampled temperatures from the pipe interior while all previous tests investigated temperature gradients at the pipe surface. The pipe internal temperature data has greatly strengthened the assumptions and inferences made with previous test data.

The streaming test and response time test have both provided valuable information needed to support the design of the fast-response RTDs installed in the reactor coolant piping. The insight provided by the above data has been factored into the Seabrook design. The impact of the offset RTD configuration at Seabrook has been evaluated and appropriate streaming uncertainties are incorporated in the statistical setpoint calculations.

TABLE 2.1-1

RESPONSE TIME PARAMETERS FOR RCS TEMPERATURE MEASUREMENT

	RTD Bypass System	Fast Response Thermowell RTD System
RTD Bypass Piping and Thermal Lag (sec)	[ ][a,c]	[ ][a,c]
RTD Response Time (sec)		
Electronics Delay (sec)		
Total Response Time (sec)	6.0 sec	6.0 sec

3.0 UNCERTAINTY CONSIDERATIONS

This method of hot leg temperature measurement has been analyzed to determine the magnitude of the two uncertainties included in the safety analysis: calorimetric flow measurement uncertainty and hot leg temperature streaming uncertainty.

3.1 CALORIMETRIC FLOW MEASUREMENT UNCERTAINTY

Reactor coolant flow is verified with a calorimetric measurement performed after the return to power operation following a refueling shutdown. The two most important instrument parameters for the calorimetric measurement of RCS flow are the narrow range hot leg and cold leg coolant temperatures. The accuracy of the RTDs has, therefore, a major impact on the accuracy of the flow measurement.

With the use of three T_{hot} RTDs (resulting from the elimination of the RTD bypass lines) and the latest Westinghouse RTD cross-calibration procedure (resulting in low RTD calibration uncertainties at the beginning of a fuel cycle), the Seabrook Unit 1 RCS flow calorimetric uncertainty is determined to be $\leq$ 2.3% Flow including the use of cold leg elbow taps (see Tables 3.1-2, 3, 4 and 5). This calculation is based on the standard Westinghouse methodology previously approved on earlier submittals of other plants associated with RTD Bypass Elimination or the use of the Westinghouse Improved Thermal Design Procedure. Tables 3.1-1 through 3.1-8 were generated specifically for Seabrook and reflect plant specific measurement uncertainties and operating conditions.

3.2 HOT LEG TEMPERATURE STREAMING UNCERTAINTY

The safety analyses incorporate an uncertainty to account for the difference between the actual hot leg temperature and the measured hot leg temperature caused by the incomplete mixing of coolant leaving regions of the reactor core at different temperatures. This temperature streaming uncertainty is based on an analysis of test data from other Westinghouse plants, and on calculations to evaluate the impact on temperature measurement accuracy of numerous possible temperature distributions within the hot leg pipe. The test data has shown that the circumferential temperature variation is no more than [

]+b,c,e, and that the inferred temperature gradient within the pipe is limited to about [ ]+b,c,e. The calculations for numerous temperature distributions have shown that, even with margins applied to the observed temperature gradients, the three point temperature measurement (scoops or thermowell RTDs) is effective in determining the average hot leg temperature. Plant specific calculations performed for the Seabrook RTD system have established an overall streaming uncertainty of [ ]+b,c,e for a hot leg measurement. Of this total, [

]+b,c,e. The remaining 0.5°F is considered to be a random uncertainty for the four loop plant. Both the systematic and random uncertainty contain components attributable to the spacing of the hot leg RTDs. The 120° RTD has been moved to the 90° location on loop A, and the 240° RTD has been moved to the 270° location in loops B and D. On loop C, the 90° RTD has been moved approximately 12 inches upstream at 105°.

The new method of measuring hot leg temperatures, with the three hot leg thermowell RTDs, is more effective than the existing RTD bypass system [the streaming error caused by imbalances in the scoop sample flows is]a,c. Although the new method measures temperature at one point (at the RTD/thermowell tip), compared to the five sample points in a 5-inch span of the scoop measurement, the thermowell measurement point is the point used to establish the streaming uncertainty and thus accounts for a range of possible temperature gradients in the hot leg. Since the thermowell tip is at the same radius as, or opposite the center hole of the scoop, the two systems measure the same average temperature [
]a,c.

Temperature streaming measurements have been obtained from tests at 2, 3, and 4-loop plants and from thermowell RTD installations at 3 and 4-loop plants.

Although there have been some differences observed in the orientation of the individual loop temperature distributions from plant to plant, the magnitude of the differences have been [

]$^{+b,c,e}$.

Over the testing and operating periods, there were only minor variations of less than [ ]$^{+b,c,e}$ in the temperature differentials between scoops, and smaller variations in the average value of the temperature differentials. The three RTDs measure a reasonably accurate hot leg temperature, which at most may have a positive (conservative) bias due to the effect of extreme reactor core power distributions. Changes that may occur during the fuel cycle are not considered to do more than reduce the positive bias. [

]$^{+b,c,e}$.

Provisions were made in the RTD electronics for operation with only two hot leg RTDs in service. The two-RTD measurement will be biased to correct for the difference compared with the three-RTD average. Based on test data, the bias value would be expected to range between [ ]$^{+b,c,e}$. Data comparisons show that the magnitude of this bias varied less than [ ]$^{+b,c,e}$ over the test period. Appendix A provides a procedure for utilizing the actual plant bias data. Note that this procedure only allows the use of positive (or zero) bias values, since the biases as established in the electronics remain constant as power is reduced. A negative bias would result in a nonconservatively low hot leg temperature measurement at reduced power because the actual temperature gradient is reduced as power is reduced.

3.3 CONTROL AND PROTECTION FUNCTION UNCERTAINTIES

Calculations were performed to determine or verify the instrument uncertainties for the control and protection functions affected by the RTD Bypass Elimination. Table 3.1-1, Rod Control System Accuracy, provides an acceptable value for control since the safety analyses assume a larger value as an initial condition. Tables 3.1-2 to 3.1-4 provide a breakdown of the uncertainties associated with the performance of a precision RCS flow calorimetric measurement. Table 3.1-5 notes the uncertainties for process computer indication of RCS flow via the Cold Leg Elbow Tap. Table 3.1-6 lists the uncertainties associated with the Low Flow Reactor Trip from the Cold Leg Elbow Taps. The current Nominal Trip Setpoint is verified to be acceptable because the Total Allowance (TA) is larger than the combination of uncertainties (CSA). Table 3.1-7, Overtemperature ΔT, notes the uncertainties for the protection function based on the failure of one Hot Leg RTD. Corresponding uncertainties for the resulting bias correction factor for the remaining operable RTDs has been included in this calculation. A comparison of TA and CSA results in the conclusion that margin exists, thus the Nominal Trip Setpoint of $K_1 = 1.0995$ is acceptable. Table 3.1-8, Overpower ΔT, notes the uncertainties for the protection function based on the failure of one Hot Leg RTD. Corresponding uncertainties for the resulting bias correction factor for the remaining operable RTDs has been included in this calculation. A comparison of TA and CSA results in the conclusion that margin exists, thus the Nominal Trip Setpoint of $K_4 = 1.0900$ is acceptable. The values shown in Technical Specification Table 3.3-4 (Engineered Safety Features Actuation System Instrumentation Trip Setpoints) for Low RCS Tavg coincident with reactor trip (Functional Unit 6.b) should be eliminated since this sequence is not credited in any accident analyses. Table 3.1-9 lists the Technical Specification which are affected by RTD Bypass Elimination. However, based on the calculations performed, the changes in uncertainties are acceptable with minimal modifications necessary, primarily Allowable Values. Please note that these calculations were performed using plant specific instrument uncertainties for Pressurizer Pressure Sensor Calibration Accuracy, Pressurizer Pressure Sensor Drift, and Feedwater Temperature Sensor Drift. These calculations were performed at a 95% probability with a high (but undefined) confidence level. The streaming values noted in this document are based on available data, are bounding, and are treated in a conservative manner.

TABLE 3.1-1

ROD CONTROL SYSTEM ACCURACY

Tavg TURB PRESS (% SPAN)

+a,c

PMA =

SCA =

M&TE=

STE =

SD =

BIAS=

RCA =

M&TE=

M&TE=

RTE =

RD =

CA =

BIAS=

RTDs USED - TH = 2 TC = 1

+a,c

ELECTRONICS CSA =

ELECTRONICS SIGMA =

CONTROLLER SIGMA =

CONTROLLER BIAS =

CONTROLLER CSA = 5.0 DEG. F

TABLE 3.1-2

FLOW CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

(% SPAN)	FW TEMP	FW PRES	FW DP	STM PRESS	TH	TC	PRZ PRESS
							+a,c
SCA =							
M&TE=							
SPE =							
STE =							
SD =							
R/E =							
RDOT=							
BIAS=							
CSA =							
# OF INST USED					3	1	4
	DEG F	PSIA	% DP	PSIA	DEG F	DEG F	PSIA
INST SPAN =	720.	1500.	100.	1300.	100.	100.	800.
							+a,c
INST UNC. (RANDOM) =							
INST UNC. (BIAS) =							
NOMINAL =	440.	1180.		1000.	618.2	560.6	2250.

TABLE 3.1-3

FLOW CALORIMETRIC SENSITIVITIES

FEEDWATER FLOW

+a,c

FA
TEMPERATURE =
MATERIAL =

DENSITY
TEMPERATURE =
PRESSURE =

DELTA P =

FEEDWATER ENTHALPY

TEMPERATURE =
PRESSURE =

h_s = 1192.9 BTU/LBM
h_f = 419.6 BTU/LBM
Dh(SG) = 773.3 BTU/LBM

STEAM ENTHALPY

+a,c

PRESSURE =
MOISTURE =

HOT LEG ENTHALPY

TEMPERATURE =
PRESSURE =

h_h = 640.2 BTU/LBM
h_c = 560.5 BTU/LBM
Dh(VESS) = 79.7 BTU/LBM
Cp(TH) = 1.548 BTU/LBM-DEGF

COLD LEG ENTHALPY

+a,c

TEMPERATURE =
PRESSURE =

Cp(TC) = 1.270 BTU/LBM-DEGF

COLD LEG SPECIFIC VOLUME

+a,c

TEMPERATURE =
PRESSURE =

TABLE 3.1-4

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTIES

+a,c

COMPONENT	INSTRUMENT ERROR	FLOW UNCERTAINTY
FEEDWATER FLOW		
VENTURI		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE		
MATERIAL		
DENSITY		
TEMPERATURE		
PRESSURE		
DELTA P		
FEEDWATER ENTHALPY		
TEMPERATURE		
PRESSURE		
STEAM ENTHALPY		
PRESSURE		
MOISTURE		
NET PUMP HEAT ADDITION		
HOT LEG ENTHALPY		
TEMPERATURE		
STREAMING, RANDOM		
STREAMING, SYSTEMATIC		
PRESSURE		
COLD LEG ENTHALPY		
TEMPERATURE		
PRESSURE		
COLD LEG SPECIFIC VOLUME		
TEMPERATURE		
PRESSURE		

TABLE 3.1-4 (Continued)

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTIES

		+a,c
BIAS VALUES		
FEEDWATER PRESSURE	DENSITY	
	ENTHALPY	
STEAM PRESSURE	ENTHALPY	
PRESSURIZER PRESSURE	ENTHALPY - HOT LEG	
	ENTHALPY - COLD LEG	
	SPECIFIC VOLUME - COLD LEG	
FLOW BIAS TOTAL VALUE		

*,**,+,++ INDICATE SETS OF DEPENDENT PARAMETERS

	+a,c
SINGLE LOOP UNCERTAINTY (WITHOUT BIAS VALUES)	
N LOOP UNCERTAINTY (WITH BIAS VALUES)	

TABLE 3.1-5

COLD LEG ELBOW TAP FLOW UNCERTAINTY

+a,c

	% DP SPAN	% FLOW
PMA =		
PEA =		
SCA =		
SPE =		
STE =		
SD =		
RCA =		
M&TE=		
RTE =		
RD =		
ID =		
A/D =		
RDOT=		
BIAS=		
FLOW CALORIM. BIAS =		
FLOW CALORIMETRIC =		
INSTRUMENT SPAN =		120.

+a,c

SINGLE LOOP ELBOW TAP FLOW UNC = [ ]

N LOOP ELBOW TAP FLOW UNC = [ ]

N LOOP RCS FLOW UNCERTAINTY
(WITH BIAS VALUES) = 2.3

TABLE 3.1-6

LOW FLOW REACTOR TRIP

	% DP SPAN	% FLOW SPAN
		+a,c
PMA1 =		
PMA2 =		
PEA =		
SCA =		
SPE =		
STE =		
SD =		
BIASF =		
BIAS1 =		
BIAS2 =		
RCA =		
M&TE =		
RCSA =		
RTE =		
RD =		
BIAS =		

INSTRUMENT RANGE = 0 TO 120.0 % FLOW

FLOW SPAN = 120.0 % FLOW

SAFETY ANALYSIS LIMIT = 87.0 % FLOW

ALLOWABLE VALUE = 89.3 % FLOW

NOMINAL TRIP SETPOINT = 90.0 % FLOW

	+a,c		+a,c		+a,c
Z = 1.85		S = 0.60		T = [ ]	
TA = 2.5		CSA = [ ]		MAR = [ ]	

TABLE 3.1-7

OVERTEMPERATURE DELTA-T TRIP

	DELTA-T	Tavg	PRESS	DELTA-I	(% SPAN) $^{+a,c}$
PMA =					
SCA =					
M&TE=					
STE =					
SD =					
BIAS=					
RCA =					
M&TE=					
M&TE=					
RCSA=					
RTE =					
RD =					
SA =					

\# OF RTDs USED TH = 2 TC = 1 (1 T_H RTD assumed failed)

INSTRUMENT SPAN = 86.4 DEGF

SAFETY ANALYSIS LIMIT = [ ]+a,c

ALLOWABLE VALUE = 2.49 % DELTA-T SPAN

NOMINAL SETPOINTS K1 = 1.0995 K3 = 0.000519

VESSEL DELTA-T = 57.6 DEGF DELTA-I GAIN = 1.00

PRESSURE GAIN = [ ]+a,c

Z =	3.50	S =	2.20	T =	[	]$^{+a,c}$
TA =	6.5	CSA =	[ ]a,c	MAR =	[	]

TABLE 3.1-8

OVERPOWER DELTA-T TRIP

	DELTA-T	Tavg	(% SPAN)
PMA =			+a,c
SCA =			
SD =			
BIAS=			
RCA =			
M&TE=			
M&TE=			
RCSA=			
RTE =			
RD =			

OF RTDs USED TH = 2 TC = 1 (1 T_H RTD assumed failed)

INSTRUMENT SPAN = 86.4 DEGF

SAFETY ANALYSIS LIMIT = [ ]+a,c

ALLOWABLE VALUE = 2.03 % DELTA-T SPAN

NOMINAL SETPOINT = 1.0900

VESSEL DELTA-T = 57.6 DEGF

Z = 2.19	S = 1.74	T = [ ] +a,c
TA = 4.9	CSA = [ ]a,c	MAR = [ ]

TABLE 3.1-9

TECHNICAL SPECIFICATION MODIFICATIONS

Overtemperature ΔT

TA = 6.5% ΔT span

Z = 3.5

S = 1.74 (Temperature) + 0.46 (Pressure)

Nominal Values K1 = 1.0995, K3 = 0.000519

Allowable Value ≤ 2.49% ΔT span

Overpower ΔT

TA = 4.9% T span

Z = 2.2

S = 1.74

Nominal Value K4 = 1.0900

Allowable Value ≤ 2.03% ΔT span

Loss of Flow

TA = 2.5% span

Z = 1.9

S = 0.60

Nominal Trip Setpoint ≥ 90.0% Loop Design Flow

Allowable Value ≥ 89.3% Loop Design Flow

DNB Parameters

Reactor Coolant Flow ≥ 3.92×10^5 gpm**

**Includes a 2.3% Flow Measurement Uncertainty

Engineered Safety Features Actuation System Trip Setpoints (Table 3.3-4)

Delete Low RCS Tavg coincident with reactor trip (Functional Unit 6.b)

4.0 SAFETY EVALUATION

RTD Bypass Elimination can potentially influence the results of the FSAR Chapter 15 safety analyses if changes in response time characteristics and instrumentation uncertainties associated with the fast response thermowell mounted RTD system are significant. These issues are discussed in the following sections.

4.1 RESPONSE TIME

The current response time parameters of the Seabrook RTD bypass system assumed in the safety analyses are shown in Table 2.1-1. For the fast response thermowell RTD system, the overall response time will consist of [ ]^{a,c} (as presented in Section 2.1 and as given in Table 2.1-1).

The new thermowell mounted RTDs have a response time equal to or better than the old bypass piping transport, thermal lag and direct immersion RTD. This then allows the total RCS temperature measurement response time to remain unchanged at 6.0 seconds (Reference Table 2.1-1). The channel response time is a factor in the Overtemperature Delta-T and Overpower Delta-T trip performance. Section 4.3 includes a discussion of the evaluation performed for those transients which rely on the above-mentioned trips.

4.2 RTD UNCERTAINTY

The proposed fast response thermowell mounted RTD system will make use of RTDs, manufactured by Weed Instruments Inc., with a total uncertainty of [ ]^{a,c} assumed for the analyses.

The FSAR analyses make explicit allowances for instrumentation errors for some of the reactor protection system setpoints. In addition, allowances are made for the average reactor coolant system (RCS) temperature, pressure and power as described in FSAR Section 15.0. These allowances are made explicitly to the initial conditions.

The following protection and control system parameters were affected by the change from one hot leg RTD to three hot leg RTDs; the Overtemperature ΔT (OTΔT), Overpower ΔT (OPΔT), and Low RCS Flow reactor trip functions, RCS average temperature measurements used for control board indication and input to the rod control system, and the calculated value of the RCS flow uncertainty. System uncertainty calculations were performed for these parameters to determine the impact of the change in the number of hot leg RTDs. The results of these calculations indicate sufficient margin exists to account for all known instrument uncertainties.

Changes have been made in the reactor protection system setpoints to account for the new thermowell mounted RTDs. In general, the current values of the nominal setpoints as defined by the Seabrook Technical Specifications remain valid, with a change in the corresponding Allowable Values.

In addition, the DNB Parameters spec will be affected. The Technical Specification LCO on Reactor Coolant System Total Flow Rate will increase due to the change in the flow calorimetric uncertainty to 2.3%. Administrative limits on Tavg are affected by the increase in uncertainty from +/-4°F to +/-5°F and will be changed accordingly. The uncertainty on the indication of pressurizer pressure is not affected by RTD bypass elimination.

4.3 NON-LOCA EVALUATION

The RTD response time discussed in Section 2.1 and the instrument uncertainties have been considered for the Seabrook non-LOCA safety analysis design basis. These effects are discussed separately in the following paragraphs.

Only those transients which assumed OTΔT/OPΔT protection are potentially affected by changes in RTD response time. As noted in Section 4.1, the new thermowell mounted RTDs have a response time equal to or better than the old bypass transport, thermal lag and direct immersion RTD. On the basis of the information documented in Table 2.1-1, it is concluded that the safety analysis assumption for the total OTΔT/OPΔT channel response time of 6.0

seconds remains valid. Additionally, evaluation of the effects of the RTD bypass elimination on the uncertainties associated with these setpoints supports the continued validity of the current non-LOCA safety analyses.

RTD instrumentation uncertainties can affect the non-LOCA transient initial condition assumptions and those transients which assume protection from the low primary coolant flow reactor trip. Although, as noted previously, the uncertainty on T_{avg} increased from +/-4°F to +/-5°F, this is still less than the uncertainty assumed in the non-LOCA accident analysis. Also, since the non-LOCA accident analysis use Thermal Design Flow, the change in the flow uncertainty has no impact. It has been determined that the RTD bypass elimination does not increase any uncertainty that will affect any initial condition assumed in any non-LOCA transient or the low primary coolant flow reactor trip.

In conclusion, the non-LOCA safety analyses applicable to Seabrook have been evaluated with respect to the replacement of the existing RTD Bypass System with the fast response thermowell installed in the reactor coolant loop piping. It was determined that all safety analysis assumptions currently assumed in the non-LOCA analyses remain valid. The Reference 1 results and conclusions are unchanged and all applicable non-LOCA safety analysis acceptance criteria continue to be met.

4.4 LOCA EVALUATION

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The magnitude of the uncertainties are such that RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will not be affected. Past sensitivity studies have shown that the variation of the core inlet temperature (T_{in}) used in the Large Break LOCA analyses affects the predicted core flow during the blowdown period of the transient. The amount of flow into the core is influenced by the two-phase vessel-side break flow, and the core cooling is affected by the quality of the fluid. These sensitivity studies concluded that the inlet temperature effect on peak cladding temperature is dependent on break size but the magnitude of the sensitivity is small. Small Break LOCA transient results are typically much less limiting with respect to Large Break LOCA analyses but they exhibit a more stable sensitivity to changes in vessel Tavg. Again, the sensitivity to these changes in Tavg on Small Break LOCA are small.

As a result of these studies, the Large and Small Break LOCA analyses are performed at a nominal value of Tavg in conjunction with conservative Appendix-K required features. The steam generator secondary side temperature and pressure are also determined using the nominal loop average temperature (Tavg) output. Since nominal values are used, the inputs to the analyses would not be affected due to the RTD bypass elimination. However, the basis for utilizing nominal Tavg conditions is that the Tavg uncertainty range will be less than or equal to +/- 4°F. Since the Tavg uncertainty for Seabrook Unit 1 is now stated to be +/- 5°F, small PCT penalties will be applied to both the Large and Small Break LOCA analyses of record to address the Tavg uncertainty range increase. It is concluded that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses. However, small PCT increases will be assessed to account for the increased Tavg uncertainty range from +/- 4°F to +/-5°F. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint without requiring reanalysis.

4.5 STEAM GENERATOR TUBE RUPTURE (SGTR) EVALUATION

The FSAR SGTR analysis is performed to evaluate the radiological consequences of an SGTR accident. An SGTR event results in a depressurization of the Reactor Coolant System (RCS) due to the continued primary to secondary leakage. As a result of the RCS depressurization, automatic reactor trip occurs on a low pressurizer pressure or overtemperature delta-T (OTΔT) signal, and safety injection (SI) actuation occurs automatically on a low pressurizer pressure signal shortly thereafter. Operator actions are required to equalize the RCS and ruptured SG pressures and stop primary to secondary break flow. The operator actions required for the SGTR recovery include identification and isolation of the ruptured SG, cooldown of the RCS to establish subcooling margin, depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage.

The SGTR analysis in the Seabrook FSAR was performed using the LOFTRAN program (Ref. 2). The primary to secondary break flow was assumed terminated at 30 minutes after initiation of an SGTR, although the operator actions to terminate the break flow were not explicitly modeled. The SGTR analysis assumed a conservatively high average RCS temperature (T_{avg}) to maximize the radioactivity released from the ruptured SG to the atmosphere. The analysis also credited the OTΔT trip for SGTR protection. Thus, the SGTR analysis could be affected by the RTD Bypass Elimination if either the uncertainty for the RCS temperature control or the temperature measurement response time is impacted. However, the removal of the RTD bypass system and substitution with fast response RTDs will not affect the overall temperature measurement response time. In addition, the increase in the uncertainty on RCS temperature control from +/-4°F to +/-5°F is still less than the uncertainty of 5.8°F assumed in the SGTR analysis. Thus, the SGTR transient response and the radioactivity release from the ruptured SG will not be affected by the RTD Bypass Elimination.

Based on the above evaluation, it is concluded that the RTD Bypass Elimination will not change the results and conclusions reported for the Seabrook FSAR SGTR analysis.

4.6 INSTRUMENTATION AND CONTROL (I&C) SAFETY EVALUATION

The RTD Bypass Elimination modification for Seabrook does not functionally change the $\Delta T/T_{avg}$ protection channels. The implementation of the fast response RTDs in the reactor coolant piping will change the inputs into the $\Delta T/T_{avg}$ Protection Sets I, II, III, and IV as follows:

1. The Narrow Range (NR) cold leg RTD in the cold leg manifold will be replaced with a fast response NR dual element RTD well mounted in the RCP pump discharge pipe. The signal from this fast response NR RTD will perform the same function as the existing RTD T_{cold} signal. One element of the RTD will be held in reserve as a spare.

2. The NR hot leg RTD in the bypass manifold will be replaced with 3 fast response NR dual element RTDs well mounted in the hot leg that are electronically averaged in the process protection system. The signal from this average T_{hot} circuit obtained from these 3 NR T_{hot} RTDs will perform the same function as the existing RTD T_{hot} signal.

3. Identification of failed signals will be by the same means as before the modifications, i.e., existing control board alarms and indications.

4. Signal process and the added circuitry to the protection system racks will be accomplished by additions to the process control (Westinghouse Model 7300) racks using 7300 technology.

   Existing control board ΔT and T_{avg} indicators and alarms will provide the means of identifying RTD failures. Upon identification of a failed hot leg RTD, the I&C personnel would disconnect the failed RTD element from the process rack terminal strip and connect the other element of the RTD. An alternate procedure would be to disconnect the failed RTD and rescale the summing amplifier for a two RTD input condition. If one hot leg RTD signal is removed from the averaging process, the electronics allow a bias to be manually added to a 2-RTD average in order to obtain a hot leg average temperature value comparable with a 3-RTD average.

   In the event of a cold leg RTD failure, the spare cold leg RTD element will be manually connected to the 7300 circuitry in place of the failed RTD. After this process, the channel would then be returned to service. During the RTD replacement or rescaling process, the plant will be in a partial trip mode and will therefore be in a safe condition.

   Other than the above changes, the instrumentation and control will remain the same and unchanged from what has previously been utilized. For example, two out of four voting logic continues to be utilized for protection functions, with the model 7300 process control bistables continuing to operate on a "de-energize to actuate" principle. Non-safety related control signals continue to be derived from electrically isolated protection channels.

The above principles of the modification have been reviewed to evaluate conformance to the requirements of IEEE-279-1971 criteria and associated 10CFR50 General Design Criteria (GDC), Regulatory Guides, and other applicable industry standards. IEEE 279-1971 requires documentation of a design basis. Following is a discussion of design basis requirements in conformance to pertinent I&C criteria:

a. Single failure criterion continues to be satisfied by this change because the independence of redundant protection sets is maintained.

b. Quality components and modules being added is consistent with use in a Nuclear Generating Station Protection System. For the Westinghouse Quality Assurance program, refer to Chapter 17 of the FSAR.

c. The changes will continue to maintain the capability of the protection system to initiate a reactor trip during and following natural phenomena credible to the plant site to the same extent as the existing system.

d. Channel independence and electrical separation is maintained because the Protection Set circuit assignments continue to be Loop 1 circuits input to Protection Set I; Loop 2, to Protection Set II; Loop 3, to Protection Set III; and Loop 4 to Protection Set IV, with appropriate observance of field wiring interface criteria to assure the independence.

e. The compliance of the hardware to IEEE 279-1971 Section 4.7 and GDC requirements concerning Control and Protection interaction has not been changed.

On the basis of the foregoing evaluation, it is concluded that the compliance of Seabrook to IEEE 279-1971, applicable GDCs, and industry standards and regulatory guides has not been changed with the I&C modifications required for RTD bypass elimination.

4.7 MECHANICAL SAFETY EVALUATION

The presently installed RTD Bypass Manifold System is to be replaced with fast acting narrow range thermowell mounted RTDs installed in the reactor coolant loop piping. This change requires modifications to the hot leg scoops, the crossover leg bypass return nozzle, the cold leg RTD bypass nozzle and one new thermowell penetration in Loops 1, 2, 3, and 4 hot leg piping.

All machining operations performed during modification of the hot and cold leg penetrations, as well as machining of the crossover leg bypass return nozzle and the new hot leg penetration will be done in a manner that minimizes debris escaping into the reactor coolant system.

The use of temporary seal plugs inserted into the hot, cold and crossover leg nozzles during machining operations minimizes the amount of debris entering the Reactor Coolant System (RCS). The use of seal plugs is precluded during Metal Disintegration Machining (MDM) operations that will be used to machine the additional flow hole in the scoop and to perform the breakthrough machining operations on the additional hot leg penetrations. Therefore, the MDM debris directly enters the Reactor Coolant System. However, the very fine particles produced by the MDM process are not of a size, form or quantity as to have a deleterious impact on plant safety.

Another form of debris is formed during reactor site machining operations. In this case the debris will come from pipe cutting processes utilizing a fine-toothed blade from a porta-band saw. The size of this debris is expected to be in the 200 - 400 micron range. Based on extensive experience in the disposition of loose parts and foreign objects in the Reactor Coolant System, it is clear that two components in the RCS represent the primary area of concern. First, the effects on the fuel must be assessed. Again, based on experience, it is concluded that the fine particles produced during the cutting process are not of a size or form as to produce fuel rod failures. The particles are slightly larger than MDM fines, but typically do not have the size or thickness of machining chips or turnings that may have caused fuel rod failures in the past. As previously evaluated, fuel rod failures are a potential commercial concern but do not represent an unreviewed safety question.

Second, the potential effects on the Reactor Coolant Pump (RCP) seal must be assessed. Debris entering the seal from the RCS during a loss of seal injection must be considered. There will be a small amount of debris in the system, which will disperse, and is unlikely to be sufficient in quantity to affect all four reactor coolant pumps. The largest of the 200 - 400 micron (.008" - .016") particulates are too large to migrate into the pump during a loss of injection. In the worst case, at the instant the seal injection is lost, the No. 1 seal leakage would be expected to be less than 6 gpm (typically 3.5 gpm). At this flow rate the velocities up the pump shaft alley are too low to keep particulates in suspension. It is not expected that these particulates will migrate into the pump bearing or seals.

Typically seals are more sensitive to gradual degradation due to small particulates 0.1 - 10 microns. There is considerable experience with debris in the 100 - 200 micron sizes being ingested by the seals during hot functional testing. At this time, injection piping may not be as clean as typically desired. This debris is ground up by the ceramic faceplates. Seal inspections after hot functionals routinely show imbedded ground metal in the seal faces. The seal performance is not measurably affected by this material.

However, the machining debris is much less than typically experienced during hot functionals. The machining debris is similar to that generated by the pump shaft impeller rubbing the labyrinth seals of the pump as the equipment breaks in. These particulates do not represent a significant safety issue. The utility should follow normal procedures with regard to inspections and replacement of seals.

Due to the expected size of the debris, there are no other components within the RCS that are a potential concern. The cutting operation itself will be controlled in such a way as to reduce the amount of fines that surround the cut at any one time. All welding and NDE will be performed per ASME Code Section XI requirements. Each of these modifications is evaluated as follows.

The original hot leg RTD bypass piping which feeds the bypass manifold must be removed and two of the three scoops will be modified to accept fast response RTD thermowells. A hole will be machined through the tip of each scoop which will provide the proper flow path. The unused (the 120° location on loops A and C, and the 240° location on loops B and D) scoops will be capped and a new penetration made in the same plane as the existing scoops but offset 30° from the unused location. A boss and thermowell will be installed at each of these new locations. The thermowells, bosses, and caps will be fabricated in accordance with Section III (class 1) of the ASME Code. The field machined surfaces will be examined prior to welding as required by ASME Code, Section XI. The installation described above will be performed using Gas Tungsten Arc Weld (GTAW) for the root pass and finished out with either GTAW or Shielded Metal Arc Weld (SMAW). All of the welds will be examined by Penetrant Test per ASME Code, Section XI.

The cold leg RTD bypass piping must be removed and the cold leg RTD bypass nozzle modified to accept the fast response RTD thermowell. The RTD thermowell will be installed into the nozzle and will extend approximately [ ]^{a,c} inches into the flow stream. The thermowell will be fabricated in accordance with Section III (class 1) of the ASME Code. The machined surfaces of the nozzle to be welded will be examined prior to welding as required by ASME Code, Section XI. The root weld joining the RTD thermowells to the modified nozzles will utilize GTAW for the rootpass and will be finished out with either GTAW or SMAW. The welds will be examined by Penetrant test per ASME Code, Section XI.

The cross-over bypass return piping must be removed and the nozzles will be modified and capped. The cap will be fabricated to meet the pressure boundary criteria of the ASME Code, Section III (class 1). The machined surfaces will be examined prior to welding as required by the ASME Code, Section XI. The cap will be root welded to the nozzles by GTAW and fill welded by either GTAW or SMAW. The welds will be examined by utilizing Penetrant Test and radiographs per ASME Code, Section XI.

Machining of the bypass nozzle, as well as any machining performed during modification of the penetration of the hot and cold legs, shall be performed such as to minimize debris escaping into the Reactor Coolant System.

During the welding of the 3" stainless steel butt welded nozzle caps, rice paper purge dams are used. The purge dams are made of Dissolvo WLD-60 weld dam paper. There should be no safety or corrosion concerns if the weld dam paper is used correctly - held in place without adhesive tape, sufficiently far from the weld to avoid overheating and scorching, and flushed to drain after completing the welding operations. The adhesive tape will not be used at Seabrook, and Westinghouse procedures, which have been verified based on mockup testing, preclude the overheating of the weld dam paper during the welding operations. However, the crossover leg nozzle at Seabrook cannot be flushed to drain following the welding operations, so that the dispersed weld dam paper at these four locations is not removed from the RCS system.

The soluble purge dam paper, Dissolvo WLD-60, is acceptable for application during inert gas welding of NSSS piping. The paper has low contaminant levels such as chloride and does not increase the risk of corrosion in NSSS components. The soluble purge dam paper, assuming it is dispersed and flushed into the RCS during startup operations, will not significantly increase the halogen concentration in the RCS or increase the risk of corrosion in NSSS components. The suspended solids in the RCS will not result in a safety concern as the Dissolvo WLD-60 dispersed solids do not contain contaminants which could be activated to significantly increase activity levels or increase the risk of corrosion when deposited on NSSS component surfaces. It should also be noted that the Dissolvo WLD-60 weld dam paper will not form a rough sided solid that could enter the pump seal. It is concluded that the dispersed material will not adversely affect the RCS chemistry nor the NSSS components.

The weld dam paper is not a concern for DNBR. This is based on the conclusions that the Dissolvo WLD-60 will disperse rapidly. Formation of a non-porous blockage in a fuel assembly of sufficient size to be a DNBR concern is not possible for this material.

Westinghouse has reviewed the impact on fuel perfor e of Dissolvo WLD-60 weld dam paper released into the Reactor Coolant System. Information on the Dissolvo WLD-60 paper, indicates that the paper is based on rice paper, a natural organic material, and does not contain significant amounts of metallic contaminants. Further, only 101 grams of material are estimated to be introduced into the system. Being organic in nature, any material (i.e. suspended solids) which may stay in the active core region would be expected to degrade rather rapidly due to the thermal and radiolytic effects. Thus the potential of the Dissolvo WLD 60 suspended solids forming a permanent crud residue on the fuel and adversely affecting the fuel performance is considered highly unlikely.

The analysis of the Dissolvo WLD-60 paper also indicated the presence of halogenated compounds. The analysis estimated the increase in the chloride concentration in the RCS due to these compounds in the paper to be 0.021 ppb, well within the 150 ppb specification limits of SIP 5-1. The impact of the halogen compound on fuel performance is negligible since the halogen limits of SIP 5-1 would be readily met.

In accordance with Article IWA-4000 of Section XI of the ASME Code, a hydrostatic test of new pressure boundary welds is required when the connection to the pressure boundary is larger than one inch in diameter. Since the cap for the crossover leg bypass return pipe is [ ]^{a,c} inches and the cold leg RTD connections are [ ]^{a,c} inches, a system hydrostatic test is required after bypass elimination. Paragraph IWB-5222 of Section XI defines this test pressure to be 1.02 times the normal operating pressure at a temperature of [ ].^{a,c}

The integrity of the reactor coolant piping as a pressure boundary component, is maintained by adhering to the applicable ASME Code sections and Nuclear Regulatory Commission General Design Criteria. The pressure retaining capability and fracture prevention characteristics of the piping is not compromised by these modifications.

5.0 CONTROL SYSTEM EVALUATION

A prime input signal to the various NSSS control systems is the RCS average temperature (T_{avg}). This is calculated electronically as the average of the measured hot leg and cold leg temperatures in each loop.

The effect of the new RTD temperature measurement system is to potentially change the time response of the T_{avg} channels in the various loops. This in turn could impact the response of [ ]^{a,c} However, as noted in Section 2.1, Table 2.1-1, the new RTD system will have a time response close to that of the present system. Therefore, there should be no significant effect in the T_{avg} channel response, and no apparent need to revise any of the control system setpoints. The need to modify control system setpoints will be determined during the plant startup following the installation cf the new RTD system by observing the response of the control systems. If necessary, signal compensators and function generators in the control systems could be adjusted to obtain a more optimum system response. In any case, the parameters listed in Table 2.1-1 would not require modification. Also, control system responses are not assumed in the FSAR transient analyses where reactor protection is provided by the Overtemperature and Overpower ΔT trips, hence changing rod control system setpoints will not impact the results of these analyses where the values of Table 2.1-1 are assumed.

6.0 CONCLUSIONS

The method of utilizing fast-response, dual element, well mounted RTDs installed in the reactor coolant loop piping as a means for RCS temperature indication has undergone extensive analyses, evaluation and testing as described in this report. The incorporation of this system into the Seabrook design meets all safety, licensing, Quality Assurance and control requirements necessary for safe operation of this unit. The analytical evaluation has been supplemented with Weed factory testing to further verify system performance and will be supplemented by in-situ testing at Seabrook. The fast-response RTDs installed in the reactor coolant loop piping adequately replace the present hot and cold leg temperature measurement system and enhances ALARA efforts as well as improve plant reliability.

The replacement material and components used in this modification were designed, procured, manufactured, tested and installed under the controls of the Weed Instrument Co. and Westinghouse NATD, 10 CFR 50 Appendix B Quality Assurance programs. The entire modification was controlled in accordance with the New Hampshire Yankee Design Control, Maintenance and Operational Quality Assurance programs.

7.0 REFERENCES

1. Seabrook Station Updated FSAR, Amendment 63.

2. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

APPENDIX A

HOT LEG RTD FAILURF COMPENSATION PROCEDURE

DEFINITION OF AN OPERABLE CHANNEL

The RTD Bypass Elimination modification uses the average of 3 RTDs in each hot leg to provide a representative temperature measurement. In the event one or more of the RTDs fails steps must be taken to compensate for the loss of that RTD's input to the averaging function.

Single RTD Failure

Hot Leg: All three hot leg RTDs must be operable during the period following refueling from cold to hot zero power and from hot zero power to full power. During the heat up period the plant operators will be [

]^{a,c}

Once [ ]^{a,c} any hot leg can then tolerate a single RTD failure and still remain operable. If the situation arises where a single hot leg RTD failure occurs a bias value must be applied to the averaging of the remaining two valid RTDs. [

]^{a,c} No reanalysis will be necessary to evaluate this situation. The plant will be allowed to operate for the balance of the fuel cycle with this single RTD failure in one of the hot legs. If another single RTD subsequently fails in a different hot leg the same bias application methodology will apply.

The plant may operate with a failed hot leg RTD at any power level during that same fuel cycle. It is permissible to shut down and start up during the cycle without requiring that the failed RTD be replaced. [

]^{a,c}

In order to eliminate any control system concerns, the Tavg and ΔT signal associated with the loop containing the failed hot leg RTD will be defeated as an input to the control system. This will prevent the control system from using a Tavg or ΔT at power levels less than 100% which may be offset due to the fixed bias. If another hot leg RTD fails in a different loop the utility should operate using manual control. Manual control is recommended because only one control channel at a time can be defeated. If automatic operation is continued the control system will most likely auctioneer the biased channel because it will be the highest Tavg due to the positive (or zero) bias application. This means the control system will perceive a higher Tavg than is real at reduced power and the plant will operate at depressed temperatures. While this is not necessarily undesirable it does reduce the total plant megawatt output. The use of automatic control can be considered based on utility power requirements.

Cold Leg: If the active cold leg RTD fails that RTD should be disconnected from the 7300 cabinets. The installed spare RTD should then be connected in the failed RTD's place.

<u>Double RTD Failure: Inoperable Channel</u>

Hot Leg or Cold Leg: If two or more of the three hot leg RTDs or both cold leg RTDs fail in the same protection channel then that channel is considered inoperable and should be placed in trip. Operation with a single valid hot leg RTD is not presently analyzed as part of the licensing basis.

PROCEDURE FOR OPERATION WITH A HOT LEG RTD OUT OF SERVICE

The hot leg temperature measurement is obtained by averaging the measurements from the three thermowell RTDs installed on the hot leg of each loop. [

]^{a,c}

In the event that one of the three RTDs fails, the failed RTD will be disconnected and the hot leg temperature measurement will be obtained by averaging the remaining two RTD measurements plus the bias. [

]^{a,c}

The bias adjustment corrects for [

]^{a,c} To assure that the measured hot leg temperature is maintained at or above the true hot leg temperature, and therefore, to avoid a reduction in safety margin at reduced power, [

]^{a,c}

An RTD failure will most likely result in an off scale high or low indication and will be detected through the normal means in use today (i.e., T_{AVG} and ΔT deviation alarms). Although unlikely, the RTD (or its electronics channel) can fail gradually, causing a gradual change in the loop temperature measurements. [

]a,c

The detailed procedure for correcting for a failed hot leg RTD is presented below:

a,c

[

]

a,c

a,c

APPENDIX B

a,c

a,c

EVALUATION OF THE EFFECTS OF REMOVAL OF THE RTD BYPASS SYSTEM ON CONTAINMENT RESPONSE OF SEABROOK STATION

SUMMARY

The RTD bypass system elimination and its replacement with thermowell mounted RTDs for measuring hot and cold leg loop temperatures is described in Reference 1. This modification will result in an increased loop temperature instrumentation uncertainty which in turn results in a higher possible Primary Coolant average temperature. The effects of the higher Primary Coolant average temperature on the containment design basis have been evaluated. The results of the evaluation confirm that sufficient margin exists in the current Seabrook Updated Final Safety Analysis Report (UFSAR) containment analysis to support this modification.

The bases for this conclusion are provided in the following discussion.

DISCUSSION

Two types of postulated accident events are used to establish the containment design basis. These are Loss of Coolant Accidents (LOCAs) and Main Steam Line Breaks (MSLB).

Main Steam Line Break

The increased RTD instrumentation uncertainty will have negligible impact on the containment response due to an MSLB. The main source of energy for such a scenario is the secondary system steam release which will not be affected significantly by a slight change in primary system stored heat.

Loss of Coolant Accident

In the UFSAR analysis the limiting event was defined as a double-ended guillotine rupture of the primary coolant system piping at the pump suction location. The calculated peak containment pressure of 49.6 psig occurred during the post-reflood phase of the LOCA transient at 3,601 seconds. The energy releases to the containment were conservatively calculated. The analysis assumed a core power level of 3657.8 MWt which is about 5% higher than required for the current licensed power level.

During the blowdown phase of the LOCA transient, the stored energy in the primary fluid is the principal source of energy released to the containment. Following the blowdown phase (reflood and post-reflood phases), additional heat is transferred to the containment via the ECCS injection water. This is mainly comprised of core decay heat and energy stored in the secondary system.

To address the impact of the proposed change, an increase of 2°F was assumed for the RCS average temperature resulting in a value of 594.7°F. This will result in an increase in blowdown energy release to the containment of 1.46×10^6 BTUs. The higher energy release will cause a slightly higher calculated containment pressure during the blowdown phase. During the reflood and post-reflood phases, the containment response will be strongly influenced by the core decay heat. The current UFSAR analysis assumed a 5% margin in the core power level and as a result a 5% margin in the decay heat energy release. This margin accumulates with time and is presented in Table 1. Beyond approximately 215 seconds into the LOCA event, the margin in energy release due to the power level assumption exceeds that associated with a 2°F increase in RCS average temperature. Since the peak containment pressure occurs beyond 215 seconds, the current UFSAR peak containment pressure is still conservative considering the proposed modification.

CONCLUSION

The impact of the RTD bypass elimination on the UFSAR containment design basis was evaluated. The evaluation determined that while the large break LOCA containment analysis event will be affected by this modification (the early pressure response during the blowdown phase of this event may increase slightly) the current UFSAR analysis results remain bounding. The basis for this assertion is the conclusion tht the longterm and peak containment pressures reported in the UFSAR for this event remain valid, due to the assumption of a conservative core power level in the analysis. The conservatism introduced by the power level assumption is sufficient to offset the effect of potentially higher initial primary stored energy resulting from the increased instrumentation uncertainty. Therefore, the proposed modification is bounded by the current containment analysis.

REFERENCES

1. WCAP-13181, RTD Bypass Elimination Licensing Report for Seabrook Power Station, G. A. Brassart and U. L. Brown, January 1992.

Table 1

Effect of RTD Bypass Elimination
on Containment Design Basis LOCA
for the Seabrook Station

Time After Pipe Rupture (sec)	Change in Primary Fluid Stored Energy (Million BTU)	Change in Decay Heat from UFSAR[a] Assumption to current licensed power level (million BTU)	Net Change in Energy Release to Containment (Million BTU)
0.00	0.00	0.00	0.0
24.20	1.46	-0.39	1.07
155.19	1.46	-1.20	0.26
~215[b]	1.46	-1.46[b]	~0.0
682.99	1.46	-3.51	-2.05
1636.86	1.46	-6.98	-5.52
3600.00	1.46	-13.28	-11.82

a. Seabrook UFSAR, Table 6.2-58

b. Values calculated by linear interpolation; this is only approximate because cumulative core decay heat release to containment is nonlinear.

EVALUATION OF THE EFFECTS OF REMOVAL OF THE RTD BYPASS SYSTEM ON YAEC-1698 AND THE SEABROOK STATION BORON DILUTION ANALYSIS

SUMMARY

This attachment provides an assessment of the potential effects of the planned removal of the RTD loop bypass system on the Steam Generator Tube Rupture and Boron Dilution safety analyses performed by Yankee to support operation of Seabrook Station.

The conclusions of this assessment are that the changes outlined in Reference 1 have negligible impact on the SGTR accident analysis for Seabrook Station performed by Yankee (Reference 2) and will have negligible impact on Boron Dilution Analyses conducted to support operation of Cycle 3 or later cycles.

The bases for this conclusion are provided in the following discussion.

DISCUSSION

The changes resulting from the RTD bypass system elimination which potentially affect the analyses performed by Yankee are:

1. Physical removal of the RTD bypass piping in each loop. This will reduce the total RCS free volume by approximately 10 cubic feet (0.1%). Reactor coolant loop temperature measurements will subsequently be provided by fast response, narrow range, dual-element RTDs installed in the hot legs and cold legs, to provide Narrow Range hot and cold leg temperatures and loop delta-T measurements. (As noted in Reference 1, the overall temperature measurement response time will be 6 seconds, the same as the existing RTD loop bypass temperature measurement arrangement).

2. An increase in the Calorimetric (RCS) Flow Measurement Uncertainty due to increased uncertainty associated with temperature streaming affects on the new loop temperature measurements. The increased uncertainty will be accounted for in determining compliance with the proposed revised "analysis" value of minimum Reactor Coolant System Flow specified in TS 3.2.5.

3. An increase in the uncertainty in the uncertainty associated with the RCS T_{avg} measurement used for control board indication and input to the rod control system from ±4°F to ±5°F.

4. Minor changes proposed for the values for Sensor Error (S), (Z), and Total Allowance (TA), specified in TS Table 2.2-1 for the Overtemperature ΔT and Overpower ΔT trip setpoints.

5. A proposed increase in the value of the K_6 coefficient of the Overpower ΔT trip setpoint.

The influence of these changes on the SGTR and Boron Dilution analyses are discussed below.

Steam Generator Tube Rupture Assessment

The SGTR analyses documented in Reference 2 were performed to quantify the radiological consequences of a design-basis SGTR at Seabrook. The potential effect of each change listed above on these analyses is discussed in the corresponding numbered paragraph below.

1. The severity of the radiological consequences is directly related to the amount of primary to secondary coolant leakage which occurs during the event. At any point in the event, the ruptured SG tube primary-to-secondary leakage flow rate is proportional to the pressure differential between the primary and secondary sides of the steam generator. The slight reduction in RCS volume due to the elimination of the RTD loop bypass piping would tend to result in a faster RCS depressurization for the same leakage flow rate, thereby reducing the primary-to-secondary pressure difference and reducing the leakage flow rate. The effects of such a small reduction in RCS volume are therefore conservative and negligible.

2. The analyses summarized in Reference 2 were performed assuming the Thermal Design RCS Flow Rate, which is the same as the proposed new value for TS 3.2.5. Thus there is no effect on the analyses.

3. The analyses summarized in Reference 2 assumed a conservatively high value for initial T_{avg} (594.5°F), 6F higher than the nominal T_{avg} (588.5°F), which conservatively maximizes the radiological release. Thus the uncertainty assumed in the analysis bounds the increased RCS temperature control uncertainty which will result from the RTD bypass elimination.

4. Although not explicitly addressed in Reference 2, reliance continues to be placed upon the action of the Overtemperature ΔT trip, to ensure that the thermal design limits on Departure from Nucleate Boiling (DNB) are not violated during the initial RCS depressurization resulting from a steam generator tube rupture. The changes proposed will assure that this continues to be true. Thus, there is no impact from the proposed changes.

5. The Overpower ΔT trip is not credited in the SGTR analysis. Thus, there is no impact from the proposed changes.

In summary, the proposed changes have negligible impact on the SGTR analyses documented in Reference 2.

Boron Dilution Analysis Assessment

The assessment of potential impact on the Boron Dilution analysis which follows is provided to supply a preliminary assessment of the probable effect on both the Westinghouse and Yankee portions of the Boron Dilution analysis for Cycle 3. Since the RTD loop bypass piping replacement will not occur until the Cycle 3 refueling outage, there can be no impact on the Boron Dilution Analysis supporting current plant operation (e.g. Cycle 2). The effects of the bypass piping removal will be considered by Westinghouse and Yankee in the performance of the Boron Dilution Analysis for operation of Cycle 3, which has yet to be performed.

The potential effect of each change listed above on these analyses is discussed in the corresponding numbered paragraph below.

1. Credit is taken for the liquid volume in the RTD loop bypass piping in the analyses of dilutions in Modes 1, 2, and 3. Elimination of the loop bypass piping will reduce the RCS mixing volume assumed in the analysis by less than 0.1%, a negligible amount.

2. RCS flow rate does not influence the boron dilution analysis, other than in a gross sense, in that the operation of one or more Reactor Coolant Pumps (RCPs) increases the assumed RCS mixing volume to include the complete reactor coolant loop volume in addition to the portions of the reactor vessel included in the mixing process. The RCS flow rate does influence the margin to DNB limits during a dilution event at power. As noted above, the Thermal Design flow rate assumed in DNB margin analyses will be assured by plant operation in compliance with the proposed changes to TS 3.2.5 Thus the margin to DNB for overpower transients due to Boron Dilution during power operation will be unaffected.

3. The only influence of the increase in uncertainty on T_{avg} measurement on the Boron Dilution analysis is on the initial margin to DNB limits for events occurring during power operation. As noted in Reference 1, the increased uncertainty remains within the value of uncertainty assumed in Chapter 15 FSAR analyses. Thus, the margin to DNB should be unaffected.

4. These changes will assure that an Overtemperature ΔT trip will occur during boron dilutions at power, prior to the existence of coolant conditions, core power distributions, and core power levels which could result in violation of the thermal design limits for DNB.

5. The Overpower ΔT trip provides protection against fuel damage resulting from overpower transients in which local power densities might otherwise exceed the value at which centerline fuel melting could occur. Figure 4-4 of WCAP-13022 (Reference 3) indicates that power distributions resulting from overpower Boration/Dilution events in which core power does not exceed 118% RTP, do not result in local power peaking which exceeds the centerline melt local power density limit. The proposed change to K_6 ensures that the Overpower ΔT trip will continue to assure a plant trip prior to power level exceeding 118% RTP.

In summary, the anticipated effects of these changes on the Cycle 3 Boron Dilution analysis are negligible.

CONCLUSION

The proposed changes to plant systems, instrumentation setpoints and measurement uncertainties, and Technical Specifications to allow removal of the RTD bypass piping have negligible effect on the SGTR analysis performed by Yankee (Reference 2) and will have negligible effect on the Boron Dilution analyses performed to support Cycle 3 operation when those analyses are performed.

REFERENCES

1. WCAP-13181, R?D Bypass Elimination Licensing Report for Seabrook Power Station, G. A. Brassart and U. L. Brown, January 1992.

2. YAEC-1698, Analysis of a Postulated Design-Basis Steam generator Tube Rupture for the Seabrook Nuclear Power Station, A. E. Ladieu et al, February, 1991.

3. WCAP-13022, The Nuclear Design and Core Physics Characteristics of the Seabrook Station Unit 1 Cycle 2, K. E. Bahr et al, September 1991.