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WCAP 12614,
Rev. 2

RTD BYPASS ELIMINATION LICENSING REPORT
FOR
J. M. FARLEY NUCLEAR PLANT UNITS 1 and 2

Prepared by R. J. Morrison

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Pittsburgh, PA

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FOREWARD

Extensive studies were performed for Farley Units 1 and 2 for the effects of increased SG Tube Plugging and Reduced Thermal Design Flow (WCAP-12694 for Unit 1, WCAP-12659 for Unit 2). The purpose of this report is to show the effects of RTD bypass elimination on the Farley Units while also considering the effects of the increased SG Tube Plugging and Reduced Thermal Design Flow.

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1.0 INTRODUCTION

Westinghouse Electric Corporation has been contracted by Alabama Power Company (APCO) to remove the existing Resistance Temperature Detector (RTD) Bypass System 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 operation of the J. M. Farley Nuclear Plant Units 1 & 2 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:

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

- 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 its own obstacles such as:

- 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 <u>Hot Leg</u>

a) The hot leg temperature measurement on each loop will be accomplished with three fast response, narrow range, dual element RTDs mounted in thermowells. One element of the RTD will be considered active and the other element will be held in reserve as 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 the three existing RTD bypass manifold scoops wherever possible. A hole will be made through the end of each scoop so that water will flow in through the existing holes in the leading edge of the scoop, past the RTD, and out through the new hole (Figure 1.2-1). If plant interferences preclude the placement of a thermowell in a scoop, then the scoop will be capped and a new penetration made to accommodate the thermowell (Figure 1.2-2). These three RTDs will measure the hot leg temperature which is used to calculate the 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 Cold Leg

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). Temperature streaming in the cold leg is not a concern due to the mixing action of the RCP. For this reason, only one RTD is required. 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 1.2-3) 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.

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 as shown on Figure 1.2-4..

1.3 ELECTRICAL MODIFICATIONS

1.3.1 Control & Protection System

Figure 1.3-1 shows a block diagram of the modified protection system 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 protection system equipment. It is planned to wire the T_{hot} and T_{cold} spare RTD elements to the control room and terminate them at the 7300 rack input terminals. This arrangement will allow on-line accessibility to the spare elements for RTD cross calibrations and to facilitate connection of the spare RTD element in the event of an RTD element failure.

The present RCS loop temperature measurement system uses dedicated direct immersion RTDs for the control systems. This was done largely to satisfy the IEEE Standard 279-1971 which applied single failure criteria to control and protection system interaction. The new thermowell mounted RTDs will be used for both control and protection. In order to continue to satisfy the requirements of IEEE Standard 279-1971, the T_{avg} and ΔT signals generated in the protection system will be electrically isolated and transmitted to the control system into Median Signal Selectors for T_{avg} and ΔT , which will select the signal which is in between the highest and lowest values of the three loop inputs. This will preclude an unwarranted control system response that could be caused by a single signal failure.

1.3.2 Qualification

The 7300 Process System Electronics modifications will be qualified to the same level as the existing 7300 electronics. RTD qualification will be verified to support APCO's compliance to 10CFR50.49.

The Westinghouse qualification program entailed a review of the WEED Instrument Company's qualification documentation for testing performed on these RTDs. It was concluded that the equipment's qualification was in compliance with IEEE Standards 344-1975 and 323-1974 with one exception. Specifically, requirements relative to flow induced vibration were not addressed. To demonstrate that flow induced vibration would not result in significant aging mechanisms that could cause common mode concerns during a seismic event, Westinghouse performed flow induced vibration tests followed by pipe vibration aging and a simulated seismic event. These tests confirmed that the WEED RTDs do comply with the above IEEE standards.

1.3.3 RTD Operability Indication

Existing control board ΔT and T_{avg} indicators and alarms will provide the means of identifying RTD failures, although the now redundant indication for the T_{avg} and ΔT control signals will be removed. The spare cold leg RTD element provides sufficient spare capacity to accommodate a single cold leg RTD failure per loop. Failure of a hot leg RTD can be handled in two ways.

The first method disconnects the failed element and utilizes the second element of the same RTD. In the second, manual action initiated by the operator defeats the failed signal and rescales the electronics to average the remaining signals.

a,c

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

a,c

Figure 1.2-2 Cold Leg Pipe Nozzle Modification
for Fas Response RTD Installation

a,c

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

a,c

Figure 1.2-4 Crossover Leg Cap Installation

a,c

Figure 1.3-1 RTD Averaging Block Diagram
Typical for Each of 3 Protection Channels

a,c

Figure 1.3-2 Median Signal Selector Block Diagram

a,c

Figure 1.3-3 RTD Bypass Elimination Control System Schematic

2.0 TESTING

There are two specific types of tests which are performed to support the installation of the thermowell mounted fast-response RTDs in the reactor coolant piping: RTD response time tests and a hot leg temperature streaming test. The response time for the Farley Units 1 & 2 application will be verified by testing at the RTD manufacturer and by in-situ testing. Data from thermowell/RTD performance tests at operating plants provide additional support for the system by confirmation of RTD/thermowell response times and by confirmation of the magnitude of temperature streaming.

2.1 RESPONSE TIME TEST

The RTD manufacturer, WEED Instruments Inc., will perform time response testing of each RTD and thermowell prior to installation at the Farley Units 1 & 2. These RTD/thermowells must exhibit a response time bounded by the values shown in Table 2.1-1. The revised response time 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 can 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 minimum to maximum of [$]^{b,c,e}$. A 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 hot leg pipes. 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 data taken from other Westinghouse designed plants to determine an appropriate temperature error for use in the safety analysis and calorimetric flow calculations. Section 3 will discuss the specifics of these uncertainty considerations.

The test data was reduced and characterized to answer the three objectives of the test program. First, it is conservative to state that the streaming pattern [ ]^{b,c,e}. Steady state data taken at 100% power for a period of four months indicated that 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. 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 has been provided to APCO which specifies how these [ ]^{b,c,e} are to be determined (Appendix A). This significantly reduces the error introduced by a failed RTD.

Both the test data and the operating data support previous calculations of streaming errors determined from tests at other Westinghouse plants. The temperature gradients defined by the recent plant operating data are well within the upper bound temperature gradients that characterize the previous test data. Differences observed in the operating data compared with the previous test data indicate that the temperature gradients are smaller, so the measurement uncertainties are conservative. The measurements at the operating plants, obtained from thermowell RTDs installed inside the bypass scoops, were expected to be, and were found to be, consistent with the measurements obtained previously from the bypass loop RTDs.

TABLE 2.1-1

RESPONSE TIME PARAMETERS FOR RCS TEMPERATURE MEASUREMENT

	RTD Bypass System	Fast Response Thermowell RTD System
	[ ]a,c	[ ]a,c
RTD Bypass Piping and Thermal Lag (sec)		
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. Tables 3.1-1 through 3.1-10 were generated specifically for APCO and reflect plant specific measurement uncertainties and operating conditions.

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 recommendations of the Westinghouse RTD cross-calibration procedure (resulting in low RTD calibration uncertainties at the beginning of a fuel cycle), the Farley Units 1 & 2 RCS Flow Calorimetric uncertainty is estimated to be [$]^{a,c}$ including use of cold leg elbow taps (see Tables 3.1-2, 3, 4 and 8). This estimate 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.

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 very effective in determining the average hot leg temperature. The most recent calculations for the thermowell RTD system have established an overall streaming uncertainty of [ ]^{b,c,e} for a hot leg measurement. Of this total, [ ]^{b,c,e}. This overall temperature streaming uncertainty determined for plants with similar or symmetrical temperature distributions is conservative when applied to 3 loop plants such as Farley Units 1 & 2 since the 3 loop temperature distributions are not symmetrical. This non-symmetric distribution results in a smaller systematic uncertainty for 3 loop plants..

The new method of measuring hot leg temperatures, with the three hot leg thermowell RTDs, is at least as effective as the existing RTD bypass system, [ ]^{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 opposite the center hole of the scoop and therefore measures the equivalent of the average scoop sample if a linear radial temperature gradient exists in the pipe. The thermowell measurement may have a small error relative to the scoop measurement if the temperature gradient over the 5-inch scoop span is nonlinear. Assuming that the maximum inferred temperature gradient of [ ]^{b,c,e} exists from the center to the end of the scoop, the difference between the thermowell and scoop measurement is limited to [ ]^{b,c,e}. Since three RTD measurements are averaged, and the nonlinearities at each scoop are random, the effect of this error on the hot

leg temperature measurement is limited to [ ]^{b,c,e}. On the other hand, imbalanced scoop flows can introduce temperature measurement uncertainties of up to
[
]^{a,c}. In all cases, the scoop flow imbalance uncertainty will equal or exceed the [ ]^{b,c,e} sampling uncertainty for the thermowell RTDs, so the new measurement system tends to be a more accurate measurement with respect to streaming uncertainties.

Temperature streaming measurements have been obtained from tests at 2, 3 and 4-loop plants and from thermowell RTD installations at 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.
[
]^{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. In addition, the uncertainty calculations assumed that two T_{hot} RTD's were utilized to determine T_{hot}. 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.

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. Methodology for these calculations has been accepted in Reference 6. Table 3.1-1 (Rod Control System Accuracy) notes that an acceptable value for control is calculated. Table 3.1-2, 3.1-3 and 3.1-4 provide the uncertainties, sensitivities and final result of the Precision RCS Flow Calorimetric. Table 3.1-5 provides the uncertainty breakdown for Overtemperature ΔT. As noted on this table, TA is greater than CSA, thus acceptable results are calculated for this function. Table 3.1-6 provides the breakdown for Overpower ΔT, with the same conclusions as for Overtemperature ΔT. Table 3.1-7 notes the uncertainty breakdown for Tavg Low-Low. Again acceptable results are calculated. Table 3.1-9 is concerned with the RCS Low Flow reactor trip. Based on the earlier calculations for the RCS Flow Calorimetric and the Rod Control System Accuracy, acceptable results are determined. Finally, Table 3.1-10 notes the changes necessary to the J. M. Farley Nuclear Plant Units 1 & 2 Technical Specifications. As noted, relatively minor changes are necessary to reflect the modified calculation results, primarily the Allowable Values. Appendix B contains a listing of acronyms for those used in the uncertainty calculations.

TABLE 3.1-1

ROD CONTROL SYSTEM ACCURACY

Tavg TURB PRESS**

+a,c

PMA =	
SCA =	
SMTE =	
STE =	
SD =	
BIAS =	
RCA =	
RMTE =	
RMTE =	
RTE =	
RD =	
CA =	
BIAS =	

NO. RTDs USED TH = 2 TC = 1

+a,c

ELECTRONICS CSA =	
ELECTRONICS SIGMA =	
CONTROLLER SIGMA =	
CONTROLLER BIAS =	
CONTROLLER CSA =	

*

**

TABLE 3.1-2

FLOW CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

(% SPAN)	FW TEMP	FW PRES	FW DP	STM PRESS	TH	TC	PRZ PRESS
							+a,c
SCA =							
SMTE=							
SPE =							
STE =							
SD =							
R/E =							
RDOT=							
BIAS=							
CSA =							
NO. OF INST USED					3	1	2
	DEG F	PSIA	% DP	PSIA	DEG F	DEG F	PSIA
INST SPAN =	500	2000	120	1200	120	120	800
							+a,c
INST UNC. (RANDOM) =							
INST UNC. (BIAS) =							
NOMINAL =							

These calculations were performed assuming that

a,c

TABLE 3.1-3

FLOW CALORIMETRIC SENSITIVITIES

FEEDWATER FLOW

Fa		+a,c
TEMPERATURE	=	
MATERIAL	=	
DENSITY		
TEMPERATURE	=	
PRESSURE	=	
DELTA P	=	
FEEDWATER ENTHALPY		
TEMPERATURE	=	
PRESSURE	=	

hS	= 1199.9	BTU/LBM
hF	= 416.4	BTU/LBM
Dh(SG)	= 783.5	BTU/LBM

STEAM ENTHALPY

		+a,c
PRESSURE	=	
MOISTURE	=	
HOT LEG ENTHALPY		
TEMPERATURE	=	
PRESSURE	=	

hH	= 629.7	BTU/LBM
hC	= 538.6	BTU/LBM
Dh(VESS)	= 91.1	BTU/LBM
Cp(TH)	= 1.495	BTU/LBM-DEGF

COLD LEG ENTHALPY

		+a,c
TEMPERATURE	=	
PRESSURE	=	

Cp(TC)	= 1.227	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 (% FLOW)
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

SINGLE LOOP UNCERTAINTY (WITHOUT BIAS VALUES)

N LOOP UNCERTAINTY (WITHOUT BIAS VALUES)

N LOOP UNCERTAINTY (WITH BIAS VALUES)

TABLE 3.1-5

OVERTEMPERATURE DELTA-T TRIP

	DELTA-T	Tavg	PRESS	DELTA-I
PMA =				
SCA =				
SMTE =				
STE =				
SD =				
BIAS =				
RCA =				
RMTE =				
RMTE =				
RCSA =				
RTE =				
RD =				
SA =				

+a,c

NO. OF RTD USED TH = 2 TC = 1

INSTRUMENT SPAN = 102.3 DEGF

SAFETY ANALYSIS LIMIT (SAL) = [ ] +a,c

ALLOWABLE VALUE = 2.60% DELTA-T SPAN

NOMINAL SETPOINTS K1 = 1.1800 K3 = 0.000635

VESSEL DELTA-T = 68.2 DEGF DELTA-I GAIN = 1.75

PRESSURE GAIN = [ ] +a,c

Z = [ ] +a,c S = [ ] +a,c T = [ ] +a,c

TA = CSA = MAR =

TABLE 3.1-6

OVERPOWER DELTA-T TRIP

	DELTA-T	Tavg
PMA =	[	]+a,c
SCA =		
SD =		
BIAS =		
RCA =		
RMTE =		
RMTE =		
RCSA =		
RTE =		
RD =		]

NO. OF RTD USED TH = 2 TC = 1

INSTRUMENT SPAN = 102.3 DEGF

SAFETY ANALYSIS LIMIT = [ ]+a,c

ALLOWABLE VALUE = 2.93% DELTA-T SPAN

NOMINAL SETPOINT = 1.0800

VESSEL DELTA-T = 68.2 DEGF

Z = [ ]+a,c S = [ ]+a,c T = [ ]+a,c

TA = [ ] CSA = [ ] MAR = [ ]

TABLE 3.1-7

Tavg Low-Low TRIP

PMA = [ ]$^{+a,c}$
SCA =
SD =
BIAS =
RCA =
RMTE =
RCSA =
RTE =
RD =

NO. OF RTD USED TH = 2 TC = 1

INSTRUMENT SPAN = 100.0 DEGF

SAFETY ANALYSIS LIMIT = [ ]$^{+a,c}$

ALLOWABLE VALUE = 540.2 DEGF

NOMINAL TRIP SETPOINT = 543.0 DEGF

Z = [ ]$^{+a,c}$ S = [ ]$^{+a,c}$ T = [ ]$^{+a,c}$

TA = [ ] CSA = [ ] MAR = [ ]

*

TABLE 3.1-8

COLD LEG ELBOW TAP FLOW UNCERTAINTY

INSTRUMENT UNCERTAINTIES

+a,c

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

+a,c
% FLOW

SINGLE LOOP ELBOW TAP FLOW UNC =

N LOOP ELBOW TAP FLOW UNC =

N LOOP RCS FLOW UNCERTAINTY =
(WITH BIAS VALUES)

+a,c

TABLE 3.1-9

LOW FLOW REACTOR TRIP

INSTRUMENT UNCERTAINTIES

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

FLOW SPAN = 120.0 % FLOW

SAFETY ANALYSIS LIMIT = [ ] +a,c

ALLOWABLE VALUE = 88.5 % FLOW

NOMINAL TRIP SETPOINT = 90.0 % FLOW

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

TABLE 3.1-10

TECHNICAL SPECIFICATION MODIFICATIONS

Overtemperature ΔT

K_1 = 1.18
Z = 4.76
S = 1.47(ΔT) + 0.64(pressure)
Allowable Value ≤ 2.6% ΔT span
Response Time ≤ 6 sec
ΔI penalty = 1.75%

Overpower ΔT

Z = 1.10
S = 1.47
Allowable Value ≤ 2.9% ΔT span

Loss of Flow

Z = 1.71
S = 0.60
Allowable Value ≥ 88.5% of Loop Design Flow

DNB Parameters

RCS Tavg = 581.5°F
RCS Total Flow Rate ≥ 267,400 gpm

Tavg Low-Low

Allowable Valve = 540°F
P-12 ≤ 547°F (Increasing)
≥ 540°F (Decreasing)

*Includes approximately 1.5% TDF Reduction and 2.2% Increase for Uncertainty.

4.0 SAFETY EVALUATION

The primary impact of the RTD Bypass Elimination on the FSAR Chapter 15 (Reference 1) safety analyses are the differences in response time characteristics and instrumentation uncertainties associated with the fast response thermowell RTD system. The effects of these differences are discussed in the following sections.

4.1 RESPONSE TIME

The response time parameters of the J. M. Farley Nuclear Plant Units 1 & 2 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 faster than the maximum allowed time for the old bypass piping transport, thermal lag and direct immersion RTD. This response time is factored into the Overtemperature ΔT trip performance. Therefore, those transients that rely on the above mentioned trip must be evaluated for the modified response characteristics. Section 4.3 includes a discussion of the evaluations performed for these events.

4.2 RTD UNCERTAINTY

The proposed fast response thermowell 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. These allowances are explicitly applied to the initial conditions for the transients.

The following protection and control system parameters were evaluated (with respect to accident analysis assumptions) for 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 loop T_{avg} measurements used for input to the rod control system, steam dump system, feedwater isolation, steam line isolation, safety injection; 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, noted in 3.3, indicate sufficient margin exists to account for known instrument uncertainties for all of the above except the rod control system accuracies and the low RCS flow reactor trip. Therefore, these items are addressed in Section 4.3.1 and 4.3.2.

4.3 NON-LOCA EVALUATION

As discussed in WCAP-12659 (Reference 7) and WCAP-12694 (Reference 8), the evaluations presented in this section have conservatively considered an operating configuration of 15% average steam generator tube plugging, with a maximum plugging level in one steam generator of 20% and with an analyzed minimum average thermal design flow of 87200 gpm/loop. The evaluation results are applicable to this level of tube plugging or any lower level.

The RTD response time discussed in Section 2.1 and the instrumentation uncertainties calculated in Section 3.3 have been considered for the J. M. Farley Nuclear Plant non-LOCA safety analysis design basis. Only those transients which assume OTΔT protection are potentially affected by changes in the 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 piping 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 total OTΔT channel response time of 6.0 seconds remains valid. Evaluation of the effects of the RTD Bypass Elimination on the uncertainties associated with these setpoints supports the continuing validity of the non-LOCA safety analyses (References 1, 7 and 8).

Instrumentation uncertainties can affect the non-LOCA transient initial condition assumptions and those transients which assume protection from low primary coolant flow reactor trip. These effects are discussed in the following sections.

4.3.1 EFFECTS OF ROD CONTROL SYSTEM ERRORS

As noted in Section 3.0, the RTD Bypass Elimination affects the Rod Control System accuracies. These accuracies affect the initial RCS Tavg assumed in the non-LOCA safety analyses. The current analysis assumptions are based on a ±4°F allowance as discussed in the J. M. Farley Nuclear Plant Units 1 and 2 FSAR, Section 15.1.2. For the RTD Bypass Elimination, the allowance is increased to ±4.3°F.

The initial Tavg assumed for the non-LOCA transients initiated from full or partial power includes an error allowance for the rod control system. The allowance is not assumed for transients initiated from zero power conditions. Therefore, the following zero power transients are not affected by the increase in the Tavg allowance from ±4°F to ±4.3°F:

RCCA Bank Withdrawal from a Subcritical Condition ((15.2.1) and the new analysis presented in References 7 and 8).

Excessive Heat Removal due to Feedwater System Malfunctions [zero power case] (15.2.10);

Accidental Depressurization of the Main Steam System (15.2.13)

RCCA Ejection [zero power cases] (15.4.6);

Rupture of a Main Steam Line (15.4.2.1)

The conclusions of these analyses as well as the conclusions in Reference 7 and Reference 8 remain valid.

For transients analyzed to confirm that the DNB design basis is met, generic plant DNB margin has been allocated to offset the DNB penalty of the additional 0.3°F in the initial Tavg. Therefore, the conclusions of the following DNB transients remain valid:

Uncontrolled RCCA Bank Withdrawal at Power (FSAR Section 15.2.2)

RCCA Misalignment (15.2.3)

Partial Loss of Forced Reactor Coolant Flow (15.2.5, and the new analysis presented in References 7 and 8)

Startup of an Inactive Reactor Coolant Loop (15.2.6)

Loss of External Electrical Load (15.2.7)

Excessive Heat Removal Due to Feedwater System Malfunctions (15.2.10)

Excessive Load Increase Incident (15.2.11)

Accidental Depressurization of the RCS (15.2.12)

Inadvertent Operation of ECCS During Power Operation (15.2.14)

Complete Loss of Forced Reactor Coolant Flow (15.3.4)

A number of non-LOCA transients are analyzed to demonstrate acceptability for criteria other than DNB. A discussion of the effects of increasing the Tavg uncertainty by 0.3°F for these transients follows.

Uncontrolled Boron Dilution (15.2.4)

The boron dilution event is an uncontrolled addition of unborated reactor makeup water into the RCS via the Chemical and Volume Control System. The boron dilution event is analyzed to demonstrate that, prior to total loss of shutdown margin, there is sufficient operator action time available to recognize the event and terminate the dilution. The increased temperature uncertainty does not change the critical parameters assumed in the analysis: the maximum dilution rate, RCS boron concentrations, or the dilution volume for any of the operational modes. Therefore, the current boron dilution analysis and the evaluation of this event in References 7 and 8 remain valid.

Loss of External Electrical Load (15.2.7)

The loss of external electrical load is a complete loss of steam load from full power without a direct reactor trip. Four cases are analyzed which are based on two different primary side pressure control strategies (automatic and no mitigating control) and two sets of core physics characteristics (minimum and maximum reactivity feedback). The key acceptance criteria for this transient besides DNB are primary and secondary pressures remaining below 110% of design. The RCS design pressure is 2485 psig (2500 psia) and the steam generator design pressure is 1085 psig (1100 psia). As shown in FSAR Figures 15.2-19 through 15.2-26, the peak pressurizer pressure for all of the cases all remains below 110% of RCS Design Pressure. The peak calculated RCS and secondary side pressures are not sensitive to the initial temperature assumed. Considering the temperature increase is only 0.3°F and given the margin in the current analysis, it is concluded that the increase in temperature uncertainty will not change the conclusions of the FSAR or the conclusions of References 7 and 8.

Loss of Normal Feedwater (15.2.8)

The loss of normal feedwater is the simultaneous loss of feedwater flow to all three steam generators. The FSAR analysis assumes that the reactor coolant pumps coastdown due to an assumed loss of offsite power. As stated in the FSAR, this event is analyzed to demonstrate that the pressurizer does not become water solid during the transient. The analysis assumes a power level corresponding to the 102% of the engineered safeguards power rating (a conservative assumption because Farley Units 1 and 2 are not licensed to operate at engineered safeguard power). The initial Tavg is assumed to be the engineered safeguards Tavg minus 4°F. The temperature uncertainty is subtracted from the nominal Tavg for this analysis because a lower temperature results in more initial RCS mass. The larger RCS mass is more conservative when verifying that the pressurizer does not fill. For a change of 0.3°F, however, this effect is small.

FSAR Figure 15.2-27 shows that the peak pressurizer water level is less than 1200 cubic feet. The pressurizer has an internal volume of 1400 cubic feet. The resulting margin to pressurizer filling has been compared to the effects of a 0.3°F change in Tavg and increased steam generator tube plugging (Reference 7). The analysis margin is sufficient to accommodate the effects of both changes and still maintain margin for filling the pressurizer. It should be noted that the decay heat model used in the current analysis is based on the ANS-1971 decay heat model. Additional analysis margin would be gained if the ANS-1979 model was used because the total energy released into the RCS is lower. Therefore, the conclusion that the pressurizer does not fill for this event remains valid.

Loss of All AC Power to the Station Auxiliaries (15.2.9)

This event represents a complete loss of power to the plant auxiliaries (i.e., the reactor coolant pumps, feedwater pumps, condensate pumps, etc.). The conclusions section in the FSAR states that the loss of forced flow and loss of normal feedwater results show that the acceptance criteria will be met for this transient. Both of these transients have been evaluated and found acceptable; therefore, the conclusions of the FSAR and References 7 and 8 remain valid.

Single RCCA Withdrawal at Full Power (15.3.6)

This event represents the accidental withdrawal of a single RCCA from the inserted bank at full power operation. An evaluation for this transient was performed for the increased Tavg uncertainty and it was demonstrated that the conclusions in the FSAR (i.e., less than 5% of the fuel rods are below the DNB limit value) and References 7 and 8 remain valid.

Major Rupture of a Main Feedwater Pipe (15.4.2.2)

The rupture of a main feedwater pipe is a break in the feedwater pipe large enough to prevent the addition of feedwater to the steam generators. There are two cases for feedline break presented in the FSAR. The primary acceptance criterion in the FSAR is that the core remains covered. Case A was analyzed assuming the initial temperature was 6.5°F above the engineered safeguards Tavg. Therefore, the current analysis bounds the increase in Tavg uncertainty.

Case B assumed that the initial temperature was 4°F above the engineered safeguards Tavg. This analysis is performed at 102% engineered safeguards power with an ANS 1971 decay heat model. The plant is not licensed to operate at engineered safeguard power and the results would be less limiting using an ANS 1979 decay heat model. Taking credit for the conservative power level assumption and the 1979 decay heat model, it is concluded that the FSAR conclusions remain valid (i.e. a feedline break at the licensed plant power level will have acceptable results) for the increase of 0.3°F in Tavg uncertainty.

In addition, this transient was analyzed for the steam generator tube plugging program. As discussed in Reference 7 and Reference 8, the analysis assumed a Tavg uncertainty of 6°F. This provides a bounding analysis for a Tavg uncertainty of 4.3°F. Therefore, the analysis described in References 7 and 8 is not affected by this increase in Tavg uncertainty.

Single Reactor Coolant Pump Locked Rotor (15.4.4)

This transient was reanalyzed to demonstrate acceptable results with a lower analysis value for the low reactor coolant loop flow setpoint. The increased uncertainty in Tavg was explicitly modeled in the analysis. Refer to subsection 4.3.2.

RCCA Ejection (15.4.6)

The RCCA ejection event is defined as the mechanical failure of a control rod drive mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. Beginning and end of cycle conditions are analyzed at full and hot zero power levels. As previously discussed, the hot zero power cases are not affected by the increase in Tavg uncertainty. The hot full power cases are analyzed assuming DNB conditions immediately following the RCCA ejection. This minimizes the heat transfer from the fuel to the coolant which maximizes the fuel rod temperature transient. Therefore a small change in the coolant temperature does not have a significant effect on the results.

There is sufficient margin in the current results to conclude that, with the increased temperature uncertainty and the steam generator tube plugging effects (Reference 7 and 8), the acceptance criteria will continue to be met. Therefore, the conclusions of the FSAR analyses and Reference 7 and 8 evaluations remain valid.

Steamline Break Mass and Energy Releases Outside Containment (15.4.2.1).

Steamline break mass and energy release data are calculated for several break sizes at different power levels for the purposes of equipment environmental qualification outside containment. Reference 2 contains the results of the outside containment steamline break mass and energy releases. Farley Units 1 and 2 were included in Reference 2 as part of Category 4. For this analysis, the Tavg uncertainty assumed was 6.5°F. Therefore, the new Tavg uncertainty of 4.3°F is bounded by the current analysis assumptions and the mass and energy release data remains applicable. Because the analysis assumptions do not change, the conclusions of References 7 and 8 remain valid.

Steamline Break Mass and Energy Releases Inside Containment (15.4.2.1).

Steamline break mass and energy release data are calculated for several break sizes at different power levels for the purposes of calculating the containment pressure and temperature response. The analysis to calculate mass and energy releases inside containment assumed a Tavg uncertainty of 4°F. The analysis also used a conservative ANS 1971 decay heat model. Raising the Tavg uncertainty to 4.3°F, and using the less limiting ANS 1979 decay heat model, results in mass and energy release data which is negligibly different from the current analysis. It is therefore concluded that the current mass and energy release data remains applicable for Farley Units 1 and 2, and the conclusions of Reference 7 remain valid.

4.3.2 CONCLUSION

The effects of the increase in Tavg uncertainty have been evaluated for all of the non-LOCA transients. The zero power transients are not affected by the change. The DNB related transients have been shown to be acceptable by using existing DNB margin. The preceding discussions for the remaining transients demonstrate that the conclusions of the FSAR and References 7 and 8 remain valid, and the steamline break mass and energy release data remains applicable for the Farley units.

4.3.3 EVALUATION OF THE LOSS OF FLOW REACTOR TRIP SETPOINT

The uncertainty for the loss of flow trip has increased with the RTD bypass elimination. In order to maintain the same Technical Specification trip setpoint, a lower analysis value was required. The current analysis value is 87% of nominal loop flow. The revised analysis value is 85% of nominal loop flow. Two transients rely on the low loop flow reactor trip; partial loss of flow and locked rotor. A discussion of the analysis performed for these transients follows.

Partial Loss of Forced Reactor Coolant Flow (15.2.5)

This analysis was performed similar to the analysis presented in the FSAR. Because Farley Units 1 and 2 are not licensed to operate at power with only two reactor coolant pumps in operation, a partial loss of flow with three reactor coolant pumps initially operating was analyzed. Three digital computer codes were used in the analysis. The LOFTRAN code (Reference 3) was used to calculate the flow coastdown, RCS transient conditions, the nuclear power transient, and the reactor trip on low loop flow. The FACTRAN code (Reference 4) was used to calculate the heat flux transient based on the nuclear power and flow data from LOFTRAN. Finally, the THINC code (described in Section 4.4 of the FSAR) was used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN.

Conservative initial conditions were assumed which included a 5.5°F uncertainty for Tavg. The low flow trip setpoint was assumed to be 85% of nominal flow. The effects of increased steam generator tube plugging were also modeled in the analysis (References 7 and 8).

The results of the analysis confirmed that the minimum DNBR during the transient remained above the limit value. Therefore, the revised low flow trip setpoint and the increased Tavg uncertainty and the effects of increased SG tube plugging have been shown to be acceptable for this transient. Refer to References 7 and 8 for the transient plots for this analysis.

Single Reactor Coolant Pump Locked Rotor (15.4.4)

This analysis was performed similar to the analysis presented in the FSAR. Because Farley Units 1 and 2 are not licensed to operate with only two reactor coolant pumps in operation, the analysis modeled three reactor coolant pumps initially operating. Two digital computer codes were used in the analysis. The LOFTRAN code (Reference 3) was used to calculate the flow coastdown, RCS transient conditions, the nuclear power transient, the reactor trip on low loop flow, and the peak RCS pressure. The FACTRAN code (Reference 4) was used to calculate the thermal behavior of the fuel at the core hot spot based on the nuclear power and flow data from LOFTRAN.

Conservative initial conditions were assumed which included a 5.5°F uncertainty for Tavg. The low flow trip setpoint was assumed to be 85% of nominal flow. The effects of increased steam generator tube plugging were also modeled in the analysis (References 7 and 8).

The results of the analysis confirmed that the peak RCS pressure remained below that which would cause stresses to exceed the faulted condition stress limits. In addition, the calculated zirconium-water reaction remained a small fraction, and the peak clad surface temperature was less than 2700°F. Therefore, the revised low flow trip setpoint and the increased Tavg uncertainty and the effects of increased SG tube plugging have been shown to be acceptable for this transient. Refer to References 7 and 8 for the transient plots for this analysis.

4.3.4 SUMMARY

In summary, non-LOCA safety analyses applicable to the Farley Units 1 and 2 have been evaluated for the replacement of the existing RTD Bypass System with fast response thermowell mounted RTDs installed in the reactor coolant loop piping. It is concluded that an increase in RCS temperature uncertainty can be accommodated by the margins in the safety analyses and allocation of generic DNB margin. In addition, it has been demonstrated by analysis that the revised analysis value for the loss of flow reactor trip setpoint is acceptable. All other safety analysis assumptions remain valid. The evaluations have also considered the tube plugging effects of References 7 and 8. The FSAR and References 7 and 8 conclusions applicable to the J. M. Farley Units 1 & 2 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 be slightly affected. The evaluation of the slight increase in the Tavg uncertainty has resulted in an estimated increase of 3°F for the Large Break LOCA Peak Cladding Temperature (PCT) and a 2°F increase for the Small Break LOCA PCT. There is sufficient margin to 2200°F for both the Large and Small Break LOCA analyses to offset the estimated increases due to RTD bypass elimination at the Farley Units. The analytical results represented in References 7 and 8 include the effect of these PCT increases.

4.5 INSTRUMENTATION AND CONTROL (I&C) SAFETY EVALUATION

The RTD Bypass Elimination modification for the J. M. Farley Units 1 & 2 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 to the $\Delta T/T_{avg}$ Protection Set I, II, and III, circuitry as follows:

1. The Narrow Range (NR) cold leg RTD (used in the protection system) in the cold leg manifold will be replaced with a fast response NR dual element well mounted RTD 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, well mounted RTDs in the hot leg that are electronically averaged in the process protection system.

3. Identification of failed signals will be by the similar means as before the modifications, i.e., existing control board alarms and protection channel indicators, except that the control systems will not be sensitive to RTD failures or protection channel failures due to MSS.

4. The NR cold leg RTD signals and the NR hot leg RTD signals are electronically processed in the plant 7300 series process protection racks to generate loop T_{avg} and delta T signals. These signals (one per loop) are electronically isolated and transmitted to the plant 7300 series process control racks. The T_{avg} and delta T signals are input to a Median Signal Selector, respectively, which selects the median signal for use in the plant control systems. By rejecting the high and low signals, the control system will not act on any single failed input channel. Since no adverse control system action therefore results from a single failed instrument channel, a second random failure is not required per IEEE 279-1971, section 4.7.

The existing protection channel control board T_{avg} and delta T indicators and alarms will provide the means of identifying RTD failures. As part of the RTD Bypass Elimination modification, the electronically isolated T_{avg} and delta T signals will be utilized for control grade signals and alarms which can also be utilized to detect failed RTD or a protection channel input signal.

Upon identification of a failed hot leg or cold leg RTD, the operator would request that I&C personnel place the failed protection channel in a tripped condition, identify the failed RTD, disconnect the failed RTD, connect the other RTD in the dual element device and rescale the applicable RTD amplifier. After this process, the channel would be returned to service.

If both RTDs in a dual element device are bad, the RTD input is removed from the averaging process and a bias is manually added to a 2-RTD average T_{hot} (as opposed to a 3-RTD average T_{hot}) in order to obtain a value comparable with the 3-RTD average T_{hot} prior to the failure of the dual element RTD.

The conversion to thermowell mounted RTDs will result in elimination of the control grade RTDs and their associated control board indicators. The protection grade channels will now be used to provide inputs to the control system through electrical isolators to prohibit faults in the control rack from propagating into the protection racks.

In order to satisfy the control and protection interaction requirements of IEEE Standard 279-1971, a Median Signal Selector (MSS) will be used in the control channels presently utilizing a high auctioneered T_{avg} or ΔT signal (there will be a separate MSS for each function). The Median Signal Selector will use as inputs the isolated protection grade T_{avg} or ΔT signals from all three loops, and will supply as an output the channel signal which is the median of the three signals. The effect will be that the various control grade systems will still use a valid RCS temperature in the case of a single signal failure.

To ensure proper action by the Median Signal Selector, the present manual switches that allow for defeating of a T_{avg} or ΔT signal from a single loop will be eliminated. The MSS will automatically select a valid signal in the case of a signal failure. Warnings that a failure has occurred will be provided by loop to median T_{avg} and ΔT deviation alarms.

Other than the above changes, the Reactor Protection System and Control System will remain the same, as that previously utilized. For example, two out of three voting logic continues to be utilized for the thermal overtemperature and overpower protection functions, with the model 7300 process control bistables continuing to operate on a "de-energize to trip" principle. Nonsafety-related control signals will now be derived from isolated protection channels.

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

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

b. The quality of the components and modules being added is consistent with use in a Nuclear Generating Station Protection System. For the Westinghouse Quality Assurance program, refer to Appendix 17C 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 RCS Loop 1 circuits input to Protection Set I; RCS Loop 2 to Protection Set II; and RCS Loop 3 to Protection Set III, with appropriate observance of field wiring interface criteria to assure the independence.

e. Due to the elimination of the dedicated control system RTD elements, temperature signals for use in the plant control systems must now be derived from the protection system RTDs. To eliminate any degrading control and protection system interaction mechanisms introduced as a consequence of the RTD Bypass Elimination modification, a Median Signal Selector has been introduced into the control system. The Median Signal Selector preserves the functional isolation of interfacing control and protection systems that share common instrument channels. The details of the signal selector implementation are contained in Section 1.3.1 and Section 4.5.

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

4.6 MECHANICAL SAFETY EVALUATION

The presently installed RTD bypass system is to be replaced with fast acting narrow range RTD thermowells. This change requires modifications to the hot leg scoops, the hot leg piping, the crossover leg bypass return nozzle, and the cold leg bypass manifold connection. All welding and NDE will be performed per ASME Code Section XI requirements. Each of these modifications is evaluated below.

The hot leg temperature measurement on each loop will be accomplished using three (3) fast response, narrow range single element RTDs mounted in thermowells. To accomplish the sampling function of the RTD bypass manifold system and minimize the need for additional hot leg piping penetrations, the RTD thermowell assemblies will be located within the existing RTD Bypass Manifold Scoops wherever possible.
[ ]^{a,c} to provide the proper flow path. If structural interferences preclude the placement of a thermowell in a given scoop, then the scoop will be capped and a new RCS penetration made to accommodate the relocated thermowell. The relocated thermowell will be located in an installation boss. A thermowell design will be used such that the thermowell will be positioned to provide an average temperature reading. The thermowell and installation boss will be fabricated in accordance with Section III (Class 1) of the ASME Code. The installation of the thermowell into the scoop or boss 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). The welding will be examined by penetrant test (PT) per the ASME Code Section XI. Prior to welding, the surface of the scoop or boss onto which welding will be performed will be examined as required by Section XI.

The cold leg RTD bypass line must also be removed. The nozzle must then be modified to accept the fast response RTD thermowell. The installation of the thermowell into the nozzle will be performed using GTAW for the root pass and finished with either GTAW or SMAW. Weld inspection by PT will be performed as required by Section XI. The thermowells will extend approximately [ ]^{a,c} inches into the flow stream. This depth has been justified based on [ ]^{a,c} analysis. The root weld joining the thermowells to the modified nozzles will be deposited with GTAW and the remainder of the weld may be deposited with GTAW or SMAW. Penetrant testing will be performed in accordance with the ASME Code Section XI. The thermowells will be fabricated in accordance with the ASME Section III (Class 1).

The cross-over leg bypass return nozzle will be modified and capped or the existing piping connection will be severed to leave a stub of pipe protruding from the nozzle and the stub will be capped. The cap design, including materials, will meet the pressure boundary criteria of ASME Section III (Class 1). The cap will be root welded to the nozzles by GTAW and fill welded by either GTAW or SMAW. Non-destructive examinations (PT and radiographs) will be performed per ASME Section XI. Machining of the bypass return nozzle (or piping), as well as any machining performed during modification of the penetrations in the hot and cold legs, shall be performed such as to minimize debris escaping into the reactor coolant system.

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 the bypass elimination modification is complete. Paragraph IWB-5222 of Section XI defines this test pressure to be 1.02 times the normal operating pressure at a temperature of 500°F or greater.

In summary, 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. Further, the pressure retaining capability and fracture prevention characteristics of the piping is not compromised by these modifications.

4.7 TECHNICAL SPECIFICATION EVALUATION

As a result of the calculations summarized in Section 3.0, several protection functions' Technical Specifications must be modified. The affected functions and their associated Trip Setpoint information, are noted on Table 3.1-10.

5.0 CONTROL SYSTEM EVALUATION

A prime input to the various NSSS control systems is the RCS average temperature, T(avg). This is calculated electronically as the average of the measured hot 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} As previously noted, the new RTD system (RTD + thermowell) will have a time response slightly longer than that of the current system (RTD + bypass line). The additional delay resulting from the Median Signal Selector (MSS) is small in comparison with the RTD time response
[(
)]^{a,c}. Therefore, there will be no significant impact on the T(avg) channel response and no need, as a result of implementing the new system, to revise any of the control system setpoints. However, APCO always has the option of making setpoint adjustments. If desired, system performance can be verified by performing a series of plant tests (e.g., step load changes, load rejections, etc.) following installation of the new RTD system. Control system setpoints can then be adjusted based on the results of the tests. It should be recognized that control systems do not perform any protective function in the FSAR accident analysis. With respect to accident analyses, control systems are assumed operative only in cases in which their action aggravates the consequences of an event, and/or as required to establish initial plant conditions for an analysis. The modeling of control systems for accident analyses is based on nominal system parameters as presented in the Precautions, Limitations, and Setpoint document.

6.0 CONCLUSIONS

The method of utilizing fast-response 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 J. M. Farley Nuclear Plants Units 1 and 2 design meets all safety, licensing and control requirements necessary for safe operation of these units. The analytical evaluation has been supplemented with in-plant and laboratory testing to further verify system performance. The fast response RTDs installed in the reactor coolant loop piping adequately replace the present hot and cold leg temperature measurement system and enhance ALARA efforts as well as improve plant reliability. In addition to the effects of the RTD Bypass Elimination, this evaluation also consider the effects of increased SG tube plugging and reduced RCS flowrate as described in References 7 and 8.

7.0 REFERENCES

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APPENDIX A

DEFINITION OF AN OPERABLE CHANNEL AND HOT LEG RTD FAILURE COMPENSATION PROCEDURE

RTD BYPASS ELIMINATION

FOR

J. M. FARLEY NUCLEAR PLANT
UNITS 1 AND 2

DEFINITION OF AN OPERABLE CHANNEL AND HOT LEG RTD FAILURE COMPENSATION PROCEDURE

Westinghouse Electric Corporation
Pittsburgh, PA

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. J. M. Farley Nuclear Plant (FNP) will have dual element RTDs installed in each hot leg thermowell location. The second element may be used when the first element fails and the three RTD average maintained. In the event of the second element failing in the same RTD, then this procedure could be invoked.

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} Typically this data is recorded at initial 100% power and, thereafter, during the normal protection channel surveillance interval.

Once [.]^{a,c} any hot leg can then tolerate failure of both elements of a single dual element RTD and still remain operable. If the situation arises where such a failure occurs a bias value must be applied to the average of the remaining two valid RTDs.
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The plant may operate with a failed hot leg RTD at any power level during that same fuel cycle. It is permissible to shutdown and startup during the cycle without requiring that the failed RTD be replaced.

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The Median Signal Selector will eliminate any control system concerns, the Tavg and ΔT signal associated with the loop containing the failed hot leg RTD will most likely not be the Median Signal chosen as the input to the control systems. If another hot leg RTD fails in a different loop the FNP should operate using manual control. Manual Rod Control is recommended so that the operator can control the plant based on the best measurement available. If automatic operation is continued the control system may choose the biased channel due to the positive (or zero) bias application. This means the control system will perceive a higher Tavg than actually exists at reduced power and the plant will operate at reduced temperatures. While this is not necessarily undesirable it does reduce the total plant megawatt output. The use of automatic rod control should be considered based on utility power requirements.

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

Double RTD Failure: Inoperable Channel

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

PROCEDURE FOR OPERATION WITH A HOT LEG DUAL ELEMENT 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.
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In the event that one of the three dual element RTDs fails, the failed RTD will be disconnected and the hot leg temperature measurement will be obtained by averaging the remaining two RTD measurements.
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The bias adjustment corrects for
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.]^{a,c} To assure that the measured hot leg temperature is maintained at or above the true hot leg temperature, and thereby avoid a reduction in safety margin at reduced power,
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An RTD failure will most likely result in an offscale high or low indication and will be detected through the normal means in use today (i.e., T_{avg} and ΔT deviation alarms and indicators). Although unlikely, the RTD (or its electronics channel) can fail gradually, causing a gradual change in the loop temperature measurements.

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

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APPENDIX

CALCULATION OF HOT LEG TEMPERATURE BIAS

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APPENDIX B

ACRONYMS FOR UNCERTAINTY CALCULATIONS

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