Joseph M. Farley Nuclear Plant Units 1 & 2
Licensing Report for Technical Specification Changes Associated With Revised Core Limits, Revised OTΔT/OPΔT Trip Setpoints and Inclusion of RAOC Control Strategy

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I Proposed Technical Specification Changes

Listed below are the proposed Technical Specifications changes for J. M. Farley Units 1 & 2. The Technical Specifications have been grouped to show the common impact.

1.0 Reactor Core Safety Limits (Figure 2.1-1 and Bases)

Nuclear Enthalpy Hot Channel Factor (T/S 3/4.2.3 and Bases)

The revised core limits provide improved margins for DNB-related events while providing the proper limits for which reactor core protection can be assured. Reanalysis/Re-evaluation of DNB-related events have resulted in all acceptance criteria being met.

The revised core limits are based on VANTAGE 5 fuel. To ensure that any burned low-parasitic (LOPAR) assembly that may be re-inserted into the core will not be limiting with respect to DNB and will not invalidate the proposed core limits, a revised limit for LOPAR FΔH has been determined. A value of 1.30 (vs. 1.55) has been established as the appropriate limit. LOCA analyses are unaffected by this change, since the current analysis value bounds this lower value.

2.0 Reactor Trip System Instrument Trip Setpoints (Table 2.2-1 and Bases)

Note 1: Overtemperature ΔT
Note 2: Overpower ΔT
Note 3 and Note 6: Allowable Values

The setpoint equations and associated constants for OTΔT/OPΔT are defined in Notes 1 and 2 of Table 2.2-1. The revised OTΔT and OPΔT setpoints have been determined to provide improved margins as well as to provide protection for DNB-related events which use OTΔT and OPΔT as the primary reactor trip. Analysis/evaluation of these events have shown that all acceptance criteria continue to be met. The setpoints have been adjusted to reflect Farley specific uncertainties. In addition, modifications to the allowable values have been made.

The Allowable Values are associated with the drift uncertainties in the process rack electronics. Farley procedures contain provisions to ensure that the channel calibrations have target setpoints that are more conservative than Technical Specification setpoints. The target setpoints also have predetermined calibration tolerances. The plant procedures ensure the as-left setpoints following calibration and/or functional testing are within these same calibration tolerances. Should a given channel be discovered outside the Technical Specification allowable value, the appropriate action statement will be followed.

3.0 Axial Flux Difference (AFD) (TS 3/4.2.1, Figure 3.2-1 and Bases Including Figure B3/4.2-1)

Heat Flux Hot Channel Factor - Fq(Z) (T/S 3/4.2.2 and Bases)

Radial Peaking Factor Limiting Report (Administrative Control Section 6.9.1.11)

FNP will change their axial offset control strategy from the current Constant Axial Offset Control (CAOC) to Relaxed Axial Offset Control (RAOC). The effects of the expected axial shapes associated with RAOC have been evaluated with respect to Condition I and Condition II events. The RAOC limits have been verified to be acceptable to all applicable criteria. The revised reactor trip setpoints reflect the RAOC strategy. The change to RAOC required changes to the above listed Technical Specifications including F(q) surveillance and its associated W(Z) functions and $Fq^C(Z)$ penalty factors in the Radial Peaking Factor Limit Report (RPFLR). No modifications are required for the specific F(q) in Specification 3.2.2.

Since this is first time implementation of RAOC, a review of NUREG-1431 was conducted so that the most current version of these specifications could be implemented. The existing requirement as it appears in LCO 3.2.1, Axial Flux Difference (AFD), to reduce the power range neutron flux high trip setpoints would be deleted in conformance with NUREG-1431. Reducing the power level to less than or equal to 50 percent RTP maintains the plant in a benign condition since under the RAOC methodology there are no AFD limits below 50 percent of RTP. In addition, a rapid rise to greater than 50 percent RTP with AFD outside limits does not immediately create an unacceptable situation. Since the transient analyses setpoint calculations for f(ΔI) (input to the overtemperature delta-T trip function) are based on the same core power distributions that the fuel designers use for a reload cycle design, the OTΔT trip function should provide an acceptable level of protection for such an excursion. It is also noted that the event would be successfully terminated by a trip at the previous setpoint level. Therefore, the potential for a reactor trip caused by the setpoint adjustment is not justified by the potential consequences of not reducing the trip setpoint. Justification for this deletion is provided in WOG letter OG-90-54 to the NRC (Jose' Calvo), dated September 5, 1990.

Also the explicit requirement to identify and correct the cause of the out-of-limit condition has been deleted. This type of statement was omitted from NUREG-1431 on the basis that it is implicit that the out-of-limit condition would have to be corrected in order to restore compliance with the LCO. This change is considered to be an editorial simplification of the specification.

The following is the justification for deletion of the action requirement:

"The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY".

Assuming that "in at least HOT STANDBY" means MODES 3, 4, 5, or 6, compliance with this requirement will require that the plant be placed in MODE 3 for the setpoint reduction. Considering that the OPΔT setpoint can be reduced while in any MODE, this limitation is not based on equipment capabilities. Although documentation of the basis of this requirement is not available, it is expected

that it is based on the increased potential for a spurious trip while in MODE 1, if the channel is placed in trip (or bypass, if available) for setpoint adjustment. Considering that the mode changes also represent challenges to operators and equipment, and that NUREG-1431, NUREG-0452, as well as other plant Fq(Z) Technical Specifications do not include this restriction, deletion of this requirement is considered to be an operational improvement.

4.0 Maximum Allowable Power Range Flux High Setpoint With Inoperable Main Steamline Safety Valve During 3 Loop Operation (Table 3.7-1)

An evaluation for the maximum power range flux high setpoint with inoperable main steamline safety valve(s) (MSSVs) resulted in revision to the allowable high setpoints. In addition, burnup dependent setpoint for operation with 1 MSSV out of service has been developed. For core average burnups of 14,000 MWD/MTU a setpoint of 60% has been established; for higher burnups the current setpoint of 87% is acceptable. These setpoints ensure adequate protection throughout the entire fuel cycle.

II Revised Core Limits and Implementation of RAOC

To support the OTΔT/OPΔT setpoint margin improvement, revised core limit lines were developed for FNP. The revised core limits are based on VANTAGE-5 as the most limiting fuel design. The revised core limits continue to be based on the approved Revised Thermal Design Procedure (RTDP) methodology (Reference 1) and the WRB-2 DNB correlation (Reference 2). The FΔH limit for the VANTAGE-5 fuel remains at the current Technical Specification value, and the FΔH limit for the LOPAR fuel is reduced for the margin improvement program to gain margin for the OTΔT setpoints. The analysis for the LOPAR fuel is still based on the WRB-1 DNB correlation (Reference 3).

Also, the RTDP Design Limit DNBR values continue to be 1.24 for typical cells and 1.23 for thimble cells for the VANTAGE-5 fuel, and the RTDP Design Limit DNBR values continue to be 1.25 for typical cells and 1.24 for thimble cells for the LOPAR fuel. Operation with a mixed core of VANTAGE-5 and LOPAR fuel is still addressed using the approved transition core DNB methodology (References 4, 5, 6, 7, and 8).

The OTΔT/OPΔT setpoint margin improvement program also included the first time implementation of Relaxed Axial Offset Control (RAOC) for the Farley Nuclear Plant using approved methods (Reference 9). The features of the Farley RAOC analysis include:

- A RAOC ΔI band of +9, -12 at 100% power and +24, -30 at 50% power; and
- An increased thermal overpower limit of 120% nominal (current limit is 118% nominal).

Consistent with the RAOC methodology, the Condition I axial power shapes were analyzed to demonstrate compliance with the LOCA Fq limit. The normal operation axial power shapes were also evaluated relative to the assumed limiting normal operation axial power shape in the analysis of the DNB-limited events which are not terminated by the OTΔT reactor trip, e.g., the loss of flow accident. The Condition II RAOC shapes were analyzed to demonstrate that the fuel melting design criterion was met. In addition, the Condition II axial power distributions were evaluated relative to the axial power distribution assumptions used to generate the DNB core limits. The limiting Condition II axial power distributions were used to define the f(ΔI) function in the revised OTΔT setpoint such that the DNB design criterion is met for accidents which are terminated by OTΔT reactor trips.

References:

1. WCAP-11397-P-A
2. WCAP-10444
3. WCAP-8762-P-A
4. WCAP-9500-A
5. SER for WCAP-9500-A
6. Supplemental Acceptance No. 2 for Referencing WCAP-9500
7. WCAP-11837-P-A
8. ET-NRC-91-3618
9. WCAP-10216-P-A Revision 1A

III Analysis and Evaluation

1.0 Background

Farley Units 1 and 2 have experienced hot leg temperature fluctuations, including random spikes, which impact the operating margin to both OTΔT and OPΔT reactor trip setpoints. In the past, this has caused the plant to experience overtemperature and/or overpower alarms during steady-state operation. Significant reanalysis work was performed in order to improve plant operating margin to both the OTΔT and OPΔT trip functions and thereby minimizing the alarms.

Also, FNP was sensitive to perturbations in pressure and temperature. These revised OTΔT and OPΔT reactor trip setpoints will provide improved operating margin to accommodate pressure/temperature fluctuations, thereby reducing the possibility of unnecessary reactor trips.

As part of this reanalysis program, the following were considered.

- Development of new core limits utilizing the RTDP
- Relaxation of the OTΔT and OPΔT trip setpoint constants
- Addition of lag compensator (filter) on measured Tavg
- Addition of lag compensator (filter) on measured ΔT
- Operation with RAOC

A comparison of current and revised setpoint constants and dynamic compensation terms for both protection functions are shown in Table 1. No hardware modifications are required since the appropriate hardware is already installed at FNP.

Table 1

Comparison of Current versus Revised OTΔT and OPΔT
Setpoint Constants and Dynamic Compensation Terms

	Current	Revised
$K_{1,\ SAL}$	1.27	1.33
$K_{1,\ Nominal}$	1.14	1.17
K_2	0.0250 /°F	0.0170 /°F
K_3	0.001275 /psi	0.000825 /psi
FΔI Deadband	-39 to +13	-23 to +15
FΔI Positive Wing Slope	2.17%	2.05%
FΔI Negative Wing Slope	1.92%	2.48%
$K_{4,\ SAL}$	1.144	1.166
$K_{4,\ Nominal}$	1.07	1.10
K_5 (for increasing T_{avg})	0.02 /°F	unchanged
K_5 (for decreasing T_{avg})	0.0 /°F	unchanged
K_6	0.00165 /°F	0.00109 /°F
Tavg Lag Compensation (filter)	0 sec	6 sec
ΔT Lag Compensation (filter)	0 sec	6 sec

2.0 Non-LOCA Transients Affected by Proposed Changes

The effect of the revised OTΔT and OPΔT trip setpoint constants and the addition of a 6 second lag filter on measured Tavg and ΔT were evaluated for the Farley units. The evaluation was performed with the Revised Thermal Design Procedure (RTDP) and considered the Relaxed Axial Offset Control (RAOC) strategy. The parameters listed in Table 1, with the exception of the OTΔT f(ΔI) penalty function, are explicitly modeled in the plant licensing-basis safety analyses. The f(ΔI) function has been developed to assure that all anticipated axial shapes have been assessed and that the OTΔT reactor trip setpoint equation provides protection for these anticipated conditions. All events which rely on the OTΔT and OPΔT reactor trip functions for primary protection were reevaluated. The Farley Nuclear Plant FSAR Chapter 15 safety analyses which rely on the OTΔT reactor trip function for primary protection are listed below.

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (Section 15.2.2)
- Uncontrolled Boron Dilution (Section 15.2.4) [indirectly credits OTΔT trip function]
- Loss of External Electrical Load and/or Turbine Trip (Section 15.2.7)
- Accidental Depressurization of the Reactor Coolant System (Section 15.2.12)

Chapter 15 of the Farley FSAR also explicitly addresses the core consequences of a steamline break occurring at hot zero power (see Sections 15.2.13 and 15.4.2.1). The scope of steamline break analyses explicitly performed for Farley are in part based on the overall response of the OPΔT trip function. That is, the plant licensing basis has not included, nor required, an explicit analysis of a steamline break event initiated at power conditions and the OPΔT has not historically been credited as a primary reactor trip function. However, for the purposes of this evaluation, the proposed relaxed OPΔT function response will be used. Therefore, a steamline break analysis initiated from full power conditions has been performed, which explicitly models the OPΔT function, in support of the relaxed protection system response.

The above noted events were each analyzed or evaluated as summarized below.

2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The Rod Withdrawal at Power (RWAP) event is analyzed with a variety of reactivity insertion rates at 10%, 60%, and 100% of rated thermal power. Depending on the case analyzed, the reactor is tripped on either a High Neutron Flux or OTΔT signal. This event was conservatively reanalyzed incorporating the revised values shown in Table 1. Based on the results of the analysis, all applicable safety analysis criteria continue to be satisfied and the FSAR conclusions remain valid.

2.2 Uncontrolled Boron Dilution

The Boron Dilution event is analyzed to identify the minimum amount of time available to terminate an inadvertent boron dilution prior to complete loss of shutdown margin. Although this transient is analyzed for various operational modes, only the Mode 1 analysis is impacted by changes in the OTΔT trip function.

The Mode 1 event is analyzed with the reactor in either manual or automatic mode. If the control rods are in automatic, the operator would be alerted to a boron dilution occurrence by rod insertion limit alarms. Since the time of occurrence of these alarms does not change due to the proposed modifications, the case currently presented in the Farley FSAR remains valid. If the reactor is in manual control, the first indication may be a reactor trip on either an OTΔT or High Neutron Flux signal. The time of reactor trip, for operator notification, is based on an equivalent reactivity insertion case (1 pcm/sec) of the Uncontrolled RCCA Withdrawal at Power (RWAP) analysis. For the Farley RWAP analysis, the 1 pcm/sec case initiated from full power generates a reactor trip via the High Neutron Flux function. Therefore, the revised OTΔT setpoint and response modelling do not impact the time at which the operator is assumed to receive initial notification of an uncontrolled boron dilution. Therefore, the results and conclusions of the FSAR analysis remain valid.

2.3 Loss of External Electrical Load and/or Turbine Trip

The Loss of Load/Turbine Trip event is analyzed from 100% rated thermal power conditions and beginning of life reactivity conditions as this results in the limiting conditions in the primary and secondary systems (i.e., minimum DNBR, system pressures, etc.). The Farley plant analysis of this event was reanalyzed to conservatively incorporate the revised OTΔT function characteristics shown in Table 1. The analysis demonstrates that all applicable safety analysis criteria continue to be met, and therefore, the conclusions of the FSAR remain valid.

2.4 Accidental RCS Depressurization

The Accidental RCS Depressurization event is analyzed from 100% rated thermal power assuming beginning of life reactivity conditions. The Farley plant analysis of this event was reanalyzed to conservatively incorporate the revised OTΔT function characteristics shown in Table 1. The analysis demonstrates that all applicable safety analysis criteria continue to be met; therefore, the conclusions of the FSAR remain valid.

2.5 Steamline Break Core Response at Power

As previously noted, the Farley steamline break analyses explicitly address the effect on the reactor core of a steamline break occurring at hot zero power. The plant licensing basis has not historically included, nor required, an explicit analysis of a steamline break event initiated at full power conditions. Therefore, in support of the margin improvement program, an explicit full power steamline break analysis has been performed.

Following a large steamline break from full power, a reactor trip signal would be generated from a low steam pressure SI signal. For smaller break sizes at full power, a reactor trip is assumed to occur on an OPΔT signal. Transient conditions are analyzed for several break sizes in order to identify the limiting RCS conditions (i.e., peak core heat flux). An explicit calculation of the minimum DNBR is performed for this limiting setpoint.

A conservative analysis was performed explicitly modeling the low steamline pressure and OPΔT protection functions for reactor protection. The analysis results demonstrate that, all applicable safety analysis criteria are met (i.e., minimum DNBR is greater than limit value and fuel centerline temperature does not exceed that which would cause fuel melt). Thus, the conclusions of the FSAR remain valid. In that the OPΔT trip is now credited as a primary trip for this event, periodic response time testing requirements will be incorporated in the FNP FSAR.

2.6 Operation at Reduced Power Levels with Inoperable MSSVs

Technical Specification Table 3.7-1 defines the maximum power levels at which the Farley units can operate with 1, 2, or 3 inoperable main steam safety valves (MSSVs) on any loop. Reduced High Neutron Flux (HNF) reactor trip setpoints in Table 3.7-1 are based on Farley specific analysis which model a direct reactor trip on OTΔT. In support of the margin improvement program and relaxed OTΔT trip setpoint equation, an analysis also allows for a relaxed HNF setpoint with one inoperable MSSV on any loop for near end-of-cycle operation. The corresponding revised Techical Specification HNF setpoints are indicated below.

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	60*
2	43
3	24

*For plant operation approaching end of cycle (i.e., core average burnup ≥ 14,000 MWD/MTU), with one inoperable safety valve on any steam generator, the maximum allowable Power Range Neutron Flux setpoint may be increased from 60% to 87% RTP.

3.0 Other Non-LOCA Events (including mass and energy release)

No other non-LOCA safety analyses, including steamline break mass/energy release calculations, require reanalysis due to these proposed changes.

4.0 Condition I Transients

All Condition I transients have been evaluated with respect to the revised OTΔT and OPΔT setpoints, and all Condition I criteria have been met. The most limiting transient (50% load rejection from 100% power) has been reanalyzed, and sufficient steady state operating margin is maintained for this transient. This analysis demonstrates that no reactor trip will occur.

5.0 LOCA/LOCA Related Events and Steam Generator Tube Rupture

OTΔT and OPΔT provide no primary protection for these events. No other proposed changes impact these events; therefore, the current licensing basis is unaffected.

6.0 Conclusions

The impact of revised OTΔT and OPΔT setpoint equations, as indicated in Table 1, and the protection system response has been evaluated for the licensing basis safety analysis. The evaluation was performed with the RTDP and supports implementation of RAOC. The analysis results demonstrate that all acceptance criteria continue to be met and that the conclusions in the FSAR remain valid.

IV OTΔT and OPΔT Trip Setpoints Uncertainties

1.0 Introduction

In March of 1977, the NRC requested several utilities with Westinghouse Nuclear Steam Supply Systems to reply to a series of questions concerning the methodology for determining instrument setpoints. A revised methodology was developed in response to those questions with a corresponding defense of the technique used in determining the overall allowance for each setpoint.

The basic underlying assumption used is that several of the uncertainty components and their parameter assumptions act independently, e.g., rack versus sensors and pressure/temperature assumptions. This allows the use of a statistical summation of the various components instead of a strictly arithmetic summation. A direct benefit of the use of this technique is increased margin in the total allowance. For those parameter assumptions known to be interactive, the technique uses the standard, conservative, arithmetic summation approach to form independent quantities.

Farley Nuclear Plant was licensed with Technical Specifications which utilized the two-column format, wherein an Allowable Value based on a certain amount of rack drift was included along with the nominal trip setpoint. As part of the Farley programmatic efforts to enhance the documented basis for Reactor Trip System (RTS)/Engineered Safety Features Actuation System (ESFAS) setpoints, Westinghouse performed the Farley-specific setpoint study utilizing the methodology described in WCAP-13751, "Westinghouse Setpoint Methodology for Protection Systems, Farley Nuclear Plants 1 & 2."

Section 3.0 of WCAP-13751 provides a description, or definition, of each of the various uncertainty components to allow a clear understanding of the methodology. Also provided is a detailed example of each setpoint margin calculation demonstrating the methodology and noting how each parameter value is derived. The Farley-specific calculation results for each RTS and ESFAS function are shown in Tables 3-1 through 3-25, with a summary in Table 3-26. In all cases, margin exists between the summation and the total allowance. Development of new core limits and OTΔT and OPΔT reactor trip setpoints requires revisions to the setpoint uncertainty calculations, as documented herein.

2.0 Combination of Uncertainty Components

2.1 Methodology

The methodology used to combine the uncertainty components for a channel is an appropriate combination of those components or groups of components which are statistically independent, i.e., not interactive. Those uncertainties which are not independent are arithmetically summed to produce groups which are independent of each other, and which can then be statistically combined.

The methodology used is the "square root of the sum of the squares" which has been utilized in other Westinghouse reports. This technique, or others of a similar nature, have been used in WCAP-10395[1] and WCAP-8567[2]. WCAP-8567 is approved by the NRC noting acceptability of statistical techniques for the application requested. Also, various ANSI, American Nuclear Society, and Instrument Society of America standards approve the use of probabilistic and statistical techniques in determining safety-related setpoints[3,4].

The relationship between the uncertainty components and the calculated uncertainty for a channel can be shown as follows:

$$CSA = \{(PMA)^2 + (PEA)^2 + (SMTE + SD)^2 + (SPE)^2 + (STE)^2 + (SRA)^2$$

$$(RCA + RMTE)^2 + (RCSA + RMTE)^2 + (RMTE + RD)^2 + (RTE)^2\}^{1/2}$$

$$+ \{(SCA + SMTE)\}^{1/2} + EA + BIAS$$

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[1] WCAP-10395 (Proprietary), WCAP-10396 (Non-Proprietary).

[2] WCAP-8567 (Proprietary), WCAP-8568 (Non-Proprietary).

[3] ANSI/ANS Standard 58.4 - 1979.

[4] ISA Standard S67.04, 1987.

where:

CSA	=	Channel Statistical Allowance,
PMA	=	Process Measurement Accuracy,
PEA	=	Primary Element Accuracy,
SRA	=	Sensor Reference Accuracy,
SCA	=	Sensor Calibration Accuracy,
SMTE	=	Sensor Measurement and Test Equipment Accuracy,
SPE	=	Sensor Pressure Effects,
STE	=	Sensor Temperature Effects,
SD	=	Sensor Drift,
RCA	=	Rack Calibration Accuracy,
RMTE	=	Rack Measurement and Test Equipment Accuracy,
RCSA	=	Rack Comparator Setting Accuracy,
RTE	=	Rack Temperature Effects,
RD	=	Rack Drift,
EA	=	Environmental Allowance, and
BIAS	=	One directional, known magnitude.

The environmental allowance is not necessarily considered interactive with all other parameters, but as an additional degree of conservatism it is added to the statistical sum. It should be noted that, if the effect on accuracy for a channel due to cable insulation resistance degradation in an accident environment is less than 0.1% of span, the magnitude of impact is considered negligible and is not factored into the calculations. For those channels for which this effect is identified to be in excess of 0.1% span, the uncertainty is directly added as a bias.

The Westinghouse setpoint methodology for Farley Nuclear Plant (FNP) results in a value with a 95% probability. With the exception of Process Measurement Accuracy and Rack Drift, all uncertainties assumed are the extremes of the ranges of the various parameters (i.e., are better than two sigma values), or are vendor specified values. Rack Drift is based on a survey of reported plant LERs and has been shown to be conservative based on a qualitative assessment of FNP rack drift data. Process Measurement Accuracy is also considered a conservative value.

2.2 Tables and Conclusion

Tables 3-5, 3-6, 3-27, and 3-28 are included to reflect the latest calculations for current FNP hardware and calibration practices. Tables 3-5 and 3-6 determine the channel statistical allowances (CSA) for the overtemperature and overpower ΔT reactor trip setpoints, respectively. Tables 3-27 and 3-28 show that there is adequate margin between the safety analysis limits (K_1[max] and K_4[max]) and the nominal trip setpoints (K_1[nominal] and K_4[nominal]) for the overtemperature and overpower ΔT reactor trip setpoints, respectively, to account for the channel statistical allowances.

TABLE 3-5

OVERTEMPERATURE ΔT (CORE BURNDOWN EFFECTS)

(Barton Model 763 transmitter for Pressurizer Pressure)

Assumes re-normalization of ΔT_o and T' once per fuel cycle

Parameter	**Allowance***
Process Measurement Accuracy	[ ]+a,c
Primary Element Accuracy	
Sensor Calibration Accuracy	
Sensor Reference Accuracy	
Measurement & Test Equipment Accuracy	
Sensor Pressure Effects	
Sensor Temperature Effects	
Sensor Drift	
Bias	
Environmental Allowance	
Rack Calibration Accuracy	

TABLE 3-5(continued)
OVERTEMPERATURE ΔT (CORE BURNDOWN EFFECTS)
(Barton Model 763 transmitter for Pressurizer Pressure)
Assumes re-normalization of ΔT_o and T' once per fuel cycle

Parameter	**Allowance***
Measurement & Test Equipment Accuracy	[ ]+a,c
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
Comparator (Included in string calibration)	
Rack Temperature Effect	
Rack Drift	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	

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* In percent ΔT span (ΔT - 108.3°F; Tavg - 100°F; Pressure - 800 psi; Power - 150% RTP; ΔI - ±60% ΔI; 108.3°F span = 150% power)
** See Table 3-27 for gain and conversion calculations
$ Number of hot leg RTDs used
$$ Number of cold leg RTDs used

Channel Statistical Allowance =

[ ]+a,c

TABLE 3-6

OVERPOWER ΔT (CORE BURNDOWN EFFECTS)

Assumes re-normalization of ΔT_o and T" once per fuel cycle

Parameter	**Allowance***
Process Measurement Accuracy	[ ]+a,c
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
Primary Element Accuracy	
Sensor Calibration Accuracy	
[ ]+a,c	
Sensor Reference Accuracy	
[ ]+a,c	
Measurement & Test Equipment Accuracy	
Sensor Pressure Effects	
Sensor Temperature Effects	
Sensor Drift	
[ ]+a,c	
Environmental Allowance	
[ ]+a,c	
Rack Calibration Accuracy	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
Measurement & Test Equipment Accuracy	
[ ]+a,c	
[ ]+a,c	
[ ]+a,c	
Comparator (Included in string calibration)	
Rack Temperature Effect	
Rack Drift	
[ ]+a,c	
[ ]+a,c	

TABLE 3-6(continued)
OVERPOWER ΔT (CORE BURNDOWN EFFECTS)
Assumes re-normalization of ΔT_o and T" once per fuel cycle

In percent ΔT span (ΔT - 108.3°F; Tavg - 100°F; Power - 150% RTP; 108.3°F span = 150% power)
** See Table 3-28 for gain and conversion calculations
$ Number of hot leg RTDs used
$$ Number of cold leg RTDs used

Channel Statistical Allowance =

[ ]+a,c

TABLE 3-27
OVERTEMPERATURE ΔT CALCULATIONS
(Barton 763 Transmitter for Pressurizer Pressure)
Assumes re-normalization of ΔT_0 and T′ once per fuel cycle

- The equation for Overtemperature ΔT:

$$\Delta T \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} (T (\frac{1}{1 + \tau_6 S}) - T') + K_3 (P - P') - f_1 (\Delta I) \}$$

K_1 (nominal)	=	1.17 Technical Specification value
K_1 (max)	=	[ ]+a,c
K_2	=	0.017/°F
K_3	=	0.000825/psi
Vessel ΔT	=	72.2 °F
ΔI gain	=	2.05% RTP/% ΔI

- Full power ΔT calculation:

ΔT span = [ ]+a,c

- Process Measurement Accuracy Calculations:

[ ]+a,c

[ ]+a,c

[ ]+a,c

[ ]+a,c

* Presumes normalization of ΔT_0 to as-found full power indicated value within the tolerance. T′ = Tref (the automatic rod control system reference temperature).

TABLE 3-27 (continued)
OVERTEMPERATURE ΔT CALCULATIONS
(Barton 763 Transmitter for Pressurizer Pressure)
Assumes re-normalization of ΔT_o and T' once per fuel cycle

ΔI - Incore / Excore Mismatch

[ ]+a,c

ΔI - Incore Map Delta-I

[ ]+a,c

- Pressure Channel Uncertainties

Gain = [ ]+a,c

SCA =
SRA =
SMTE =
STE =
SD =
BIAS = [ ]+a,c

RCA =
RMTE =
RD = [ ]+a,c

TABLE 3-27(continued)
OVERTEMPERATURE ΔT CALCULATIONS
(Barton 763 Transmitter for Pressurizer Pressure)
Assumes re-normalization of ΔT_o and T' once per fuel cycle

- Tavg Channel Uncertainties

Gain = [ ]+a,c

RCA = [ ]+a,c

RMTE =

RD =

- ΔI Channel Uncertainties

Gain = [ ]+a,c

RCA = [ ]+a,c

RMTE =

RD =

- Total Allowance

[ ]+a,c

TABLE 3-28

OVERPOWER ΔT CALCULATIONS

Assumes re-normalization of ΔT_o and T" once per fuel cycle

- The equation for Overpower ΔT:

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

K_4 (nominal)	=	1.10 Technical Specification value
K_4 (max)	=	[ ]+a,c
K_5	=	0 for decreasing average temperature
K_5	=	0.02/°F for increasing average temperature
K_6	=	0.00109/°F
Vessel ΔT	=	72.2 °F

- Full power ΔT calculation:

ΔT span = [ ]+a,c

- Process Measurement Accuracy Calculations:

[ ]+a,c

[ ]+a,c

[ ]+a,c

[ ]+a,c

* Presumes normalization of ΔT_0 to as-found full power indicated value within the tolerance. T" = Tref (the automatic rod control system reference temperature).

TABLE 3-28(continued)

OVERPOWER ΔT CALCULATIONS

Assumes re-normalization of ΔT_o and T" once per fuel cycle

- Tavg Channel Uncertainties

Gain = [ ]+a,c

RCA = [ ]+a,c

RMTE =

RD =

- Total Allowance

[ ]+a,c

V Conclusions

All analyses and evaluations for the proposed Technical Specification changes have been completed and all acceptance criteria continue to be met. Therefore, the proposal changes are acceptable and FNP will operate in a safe, efficient configuration.

ATTACHMENT III

Significant Hazards Evaluation

Joseph M. Farley Nuclear Plant
Technical Specifications Change Request
Revision to Core Limits and OTΔT & OPΔT Setpoints and Implementation of RAOC

Joseph M. Farley Nuclear Plant
Technical Specifications Change Request
Revision to Core Limits and OTΔT Setpoints and Implementation of RAOC

10 CFR 50.92 EVALUATION

Pursuant to 10 CFR 50.92 each application for amendment to an operating license must be reviewed to determine if the proposed change involves a significant hazards consideration. The proposed amendment modifies the Reactor Core Safety Limits; reactor trip parameters for OTΔT and OPΔT; power range neutron flux high setpoints with inoperable main steamline safety valves (MSSVs); axial offset control strategy; Administrative Control Section for Peaking Factor Limit Report; and FΔH for Low Parasitic (LOPAR) assemblies. These proposed changes to the Technical Specifications have been reviewed and deemed not to involve a significant hazards consideration. The basis for this determination follows.

BACKGROUND

Farley Nuclear Plant (FNP) has undertaken a margin improvement program that will provide improved operating margins as well as increased flexibility with respect to core designs and plant operating strategy. FNP will complete the transition from a full core of LOPAR fuel to a full core of VANTAGE 5 fuel. This transition has previously been reviewed and approved by the NRC. The margin improvement program includes revision to Reactor Core Safety Limits and the associated OTΔT and OPΔT protection setpoint equations. To ensure that all available margins have been taken into account, the proposed core limits have been developed assuming VANTAGE 5 fuel as the most limiting fuel type. Reinsertion of LOPAR assemblies is not precluded by having VANTAGE 5 as the limiting fuel type but will be reviewed as part of the normal reload safety evaluation process. The applicable Technical Specifications for LOPAR fuel will be appropriately changed to reflect this approach. The axial offset control strategy has been changed from Constant Axial Offset Control (CAOC) to Relaxed Axial Offset Control (RAOC). The new core limits, which are based on VANTAGE 5 fuel, require a modification to the FΔH for LOPAR fuel so that LOPAR will never be limiting with respect to DNB. Revised OTΔT setpoints also required revision to the maximum power range neutron flux high setpoints with inoperable MSSV(s). The revised OTΔT and OPΔT setpoints will provide improved operating margin to accommodate pressure/temperature fluctuations, thereby reducing the possibility of unnecessary reactor trips.

PROPOSED CHANGES

Core limits have been revised to reflect VANTAGE 5 fuel as the most limiting fuel design using the Revised Thermal Design Procedure (RTDP). The LOPAR FΔH will be revised to ensure that VANTAGE 5 remains the limiting fuel type with respect to DNB. Loss of coolant accidents (LOCA) are not affected by this reduced FΔH, since the current analysis bounds this value. Any reinsertion of LOPAR (Low parasitic) assemblies will have an FΔH lower than the established limit, and the proposed core limits will be verified to be acceptable prior to reinsertion of LOPAR assemblies. Any transition core penalty associated with limited reinsertion of burned LOPAR fuel will be covered by available DNB margin in VANTAGE 5 fuel core limits. Modifications to the constants in the OTΔT and OPΔT equations resulted from changes to reactor core safety limits. These include a change in the K1 term (constant in the OTΔT equation = 1.17) and modification of K2 and K3 to afford protection for the revised core limits. In addition, K4 (constant in the OPΔT equation = 1.10) and K6 were also revised. The corresponding allowable values were also changed.

Modifications to the Technical Specifications supporting the first time implementation of the RAOC strategy and use of W(Z) values for Fq surveillance were done using the guidance of NUREG-1431.

The OTΔT and OPΔT equations also reflect the inclusion of the RAOC strategy and the inclusion of a 6 second time constant (τ_5 and τ_6) for ΔT and T_{avg} to provide the plant with additional flexibility in calibrating the OTΔT and OPΔT channels and to minimize the potential impact of channel noise induced by hot leg streaming variations. Changes to the F(ΔI) function assure all anticipated operating strategies are bounded.

Modification to the maximum power range neutron flux high setpoints with MSSV(s) out of service is required. To provide maximum plant flexibility and operability, a burnup dependent value has been developed for one MSSV out of service.

Appropriate Bases changes will be made to reflect these proposed changes. The Administrative Section for Peaking Factor Limit Report (PFLR) will reflect use of W(Z) values for Fq surveillance. In addition, any required penalty factors due to Fq versus burnup dependence will be included in the PFLR.

ANALYSIS

The transient analyses that use DNB as a design basis criterion and that explicitly credit reactor trip on OTΔT and OPΔT were reanalyzed or evaluated. These FSAR events include Uncontrolled RCCA Bank Withdrawal at Power, Loss of External Electrical Load and/or Turbine Trip, Accidental Depressurization of RCS, Main Steamline Break, and Uncontrolled Boron Dilution. An evaluation of calculations supporting plant operation with main steam safety valves (MSSVs) out-of-service was also performed. In addition, the OPΔT and OTΔT setpoints have been verified to be acceptable for all Condition I transients.

The safety analyses utilized the NRC approved RTDP methodology (WCAP-11397-P-A). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNB values were determined such that there is at least a 95% probability, at a 95% confidence level, that DNB will not occur on the most limiting fuel rod during normal operation and operational transients (Condition I), and during transient conditions arising from faults of moderate frequency (Condition II events). The design limit DNB values have not changed due to these proposed changes. The uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system flow) are unchanged from the NRC approved VANTAGE 5 implementation. The revised OPΔT and OTΔT setpoints and allowable values were determined using Westinghouse setpoint methodology based on Farley-specific instrumentation and calibration practices.

The safety analyses and results support the proposed Technical Specifications changes and show that the DNB design criterion is met. The safety analyses continue to meet all acceptance criteria and support the use of the revised core limits, revised OTΔT and OPΔT setpoints and the use of the RAOC strategy. RAOC has been reviewed and approved in WCAP-10216-P-A. The Condition I and II axial power shapes resulting from the implementation of RAOC have been analyzed and all acceptance criteria continue to be met. The revised core limits and the revised OTΔT and OPΔT setpoints include the effects of this analysis. In addition, the power spike factor to account for fuel densification effects has been eliminated based on WCAP-13589-A. The proposed FΔH change for LOPAR fuel has been evaluated. The evaluation found that any LOPAR assembly with an FΔH less than the proposed value will not be limiting with respect to DNB. The modification to the OTΔT setpoint required evaluation of the maximum allowable power range High Neutron Flux (HNF) setpoints with inoperable MSSV(s). Previously, the OTΔT reactor trip provided protection for this scenario. However, since the setpoint has been relaxed, the OTΔT function no longer is used to provide protection for this postulated operating configuration. Therefore, the HNF setpoint has been revised so that protection is afforded. This evaluation has also resulted in a setpoint based on core average burnup. For a core average burnup of less than 14,000 MWD/MTU, the HNF setpoint of 60% with one inoperable MSSV is required. For core average burnups of greater than 14,000 MWD/MTU the current setpoint of 87% is appropriate. This will provide protection throughout the entire fuel cycle.

CONCLUSIONS

Based on the information presented above the following conclusions can be reached with respect to 10 CFR 50.92.

1. The proposed safety limits, reactor trip setpoints, HNF setpoints for MSSVs out of service, FΔH for LOPAR, and RAOC strategy changes do not increase the probability or consequences of an accident previously evaluated in the FSAR. The core safety limits and trip setpoints were determined using the NRC reviewed and approved DNB methodologies, namely RTDP, and approved DNB correlations. No new performance requirements are being imposed on any system or component in order to support the revised core limits. Overall plant integrity is not reduced. The DNB sensitive transients that are protected by OPΔT and OTΔT were reanalyzed or evaluated. The DNB design criterion continues to be met. None of these changes directly initiate an accident; therefore, the probability of an accident has not increased. No new performance requirements are imposed on any safety-related equipment. The acceptance criteria for the reanalyses continue to be met; therefore, the consequences of accidents previously evaluated in the FSAR are not significantly changed. All dose consequences have been evaluated for these changes and all acceptance limits continue to be met. All safety analyses that use the revised OTΔT and OPΔT setpoints continue to meet all acceptance criteria. LOCA analyses are not affected by any of these proposed changes.

2. The proposed Technical Specifications changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed Technical Specifications changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. The DNB design criterion continues to be met. The use of the revised core limits, reactor trip setpoints and RAOC have been shown to allow FNP to operate in a safe configuration. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed Technical Specifications changes do not involve a significant reduction in a margin of safety. All accident analysis acceptance criteria continue to be met. The DNB design criterion remains unchanged. The DNBR design limit values have not changed. Therefore, the DNB design limit values associated with the DNB methodology and correlations, upon which the Technical Specifications changes are based, do not result in a significant reduction in the margin of safety because the DNB design criterion continues to be met. The proposed revisions to the Technical Specifications result in an operating configuration consistent with the analytic assumptions (including LOCA analyses) used to form the bases of the Technical Specifications.

Based upon the preceding information, it has been determined that these proposed changes to the FNP Technical Specifications for core limits, Reactor Trip System setpoints, and RAOC do not involve a significant hazards considerations as defined by 10 CFR 50.92.