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September 9, 2011

SOUTHERN COMPANY

Docket Nos.: 50-348
50-364

NL-11-1057

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant Units 1 and 2
License Amendment Request for Technical Specification Table 3.3.1-1

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is submitting a request for an amendment to the Technical Specifications (TS) for Farley Nuclear Plant (FNP) Units 1 and 2.

The proposed change will add Surveillance Requirement (SR) 3.3.1.14 to FNP TS Table 3.3.1-1, "Reactor Trip System Instrumentation," Function 3, "Power Range Neutron Flux High Positive Rate." SR 3.3.1.14 requires verification that the RTS RESPONSE TIME is within limits every 18 months on a STAGGERED TEST BASIS. Function 3 is the Power Range Neutron Flux High Positive Rate Trip (PFRT) function. This change is being proposed based on a reanalysis of the Rod Cluster Control Assembly Bank Withdrawal at Power event.

Enclosure 1 provides a description and justification for the proposed change. Enclosure 2 provides the marked-up TS pages and TS Bases pages for the proposed change. Enclosure 3 provides the clean typed TS pages and TS Bases pages.

SNC requests approval of the proposed license amendments by August 31, 2012. The proposed change will be implemented 90 days from the date of issuance.

This letter contains no NRC commitments.

If you have any questions, please contact Jack Stringfellow at (205) 992-7037.

Mr. M. J. Ajluni states he is Nuclear Licensing Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

Sworn to and subscribed before me this 9th *day of* September*, 2011.*

Notary Public

My commission expires: 11-30-14

MJA/DWM/lac

Enclosures: 1. Basis for Proposed Change
2. Technical Specifications and Bases Markup Pages
3. Technical Specifications and Bases Clean Typed Pages

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. J. T. Gasser, Executive Vice President
Mr. M. L. Stinson Vice President – FNP
Ms. P. M. Marino, Vice President – Engineering
RType: CVC7000

U. S. Nuclear Regulatory Commission
Mr. V. M. McCree, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – FNP
Mr. E. L. Crowe, Senior Resident Inspector – FNP

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Farley Nuclear Plant Units 1 & 2
License Amendment Request for Technical Specification Table 3.3.1-1

Enclosure 1

Basis for Proposed Change

Farley Nuclear Plant Units 1 & 2
License Amendment Request for Technical Specification Table 3.3.1-1

Enclosure 1

Basis for Proposed Change

Table of Contents

1.0 Summary Description

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is proposing a change to the Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS). This proposed change would revise TS 3.3.1, "Reactor Trip System (RTS) Instrumentation" for FNP Units 1 and 2 to add Surveillance Requirement (SR) 3.3.1.14 to Function 3, "Power Range Neutron Flux High Positive Rate" of TS Table 3.3.1-1, "Reactor Trip System Instrumentation."

2.0 Detailed Description

TS 3.3.1 Table 3.3.1-1 Proposed Change

The proposed change would revise TS 3.3.1 to add SR 3.3.1.14 to Function 3 of TS Table 3.3.1-1. SR 3.3.1.14 requires that RTS RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 3 is the power range neutron flux high positive rate trip (PFRT) function. This change is being proposed based on a reanalysis of the Rod Control Cluster Control Assembly Bank Withdrawal at Power event.

3.0 Technical Evaluation

SR 3.3.1.14 requires a verification that RTS RESPONSE TIMES are within limits every 18 months on a STAGGERED TEST BASIS, as defined in the FNP TS. As stated in the TS Bases, SR 3.3.1.14 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the Final Safety Analysis Report (FSAR), Chapter 16 (Reference 1). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

FNP FSAR Table 7.2-5 lists the associated response time as N/A for item 3 (Power Range Neutron Flux High Positive Rate). As a result, the response time for the PFRT is not being verified.

RWAP RCS Overpressure Analyses for FNP

Nuclear Safety Advisory Letter (NSAL) 09-1 dated February 4, 2009 (Reference 2), issued by Westinghouse discussed the potential for Reactor Coolant System (RCS) overpressurization as a result of a control rod bank withdrawal during power operation (RWAP) and the overall results of analyses crediting PFRT for this event. Westinghouse determined that the methodology used for the generic and plant-specific RWAP RCS overpressure analyses incorrectly assumed that a minimum initial power level creates the most limiting condition. Previous analyses have assumed an initial power level of 10 percent of rated thermal power (RTP), minus calorimetric uncertainty. Further investigation has identified cases from higher initial power levels that are more limiting. With the generic analysis key parameters and methodology, some cases exceeded the RCS overpressure limit for initial power levels in the range of 60 percent to 80 percent RTP.

Since these are the results of very conservative methodology, Westinghouse has concluded that a substantial safety hazard does not exist for Westinghouse PWRs or within the AP1000 design certification. However, Westinghouse completed specific analyses for FNP Units 1 and 2 which addressed the potential for RCS overpressure following a RWAP. The results for FNP, using a conservative methodology, demonstrate that the RCS overpressure limit listed in FNP TS 2.1.2 (i.e., 2735 psig) is not violated, assuming that credit is taken for a PFRT at or below 9 percent of RTP with a lag time constant of 2 seconds and a trip delay time of 0.65 second.

These analyses were based on the key parameters for the Farley units listed on Table A-1. The Farley analyses for a PFRT at 9% of RTP with 0.65 second delay involved 680 individual LOFTRAN computer runs at various initial conditions and reactivity insertion rates. The results are plotted on the attached Figure A-1. Two plots are provided: peak RCS pressure versus reactivity insertion rate; and peak RCS pressure versus initial power level. Both plots are based on the 340 transient calculations starting from a pressurizer pressure of 2200 psia and the 340 transient calculations starting from a pressurizer pressure of 2300 psia. These analyses calculated an acceptable result (i.e., RCS peak pressure not in excess of 2750 psia) with a PFRT at 9% of RTP, a lag time constant of 2.0 seconds, and a trip delay time of 0.65 second.

Table A-1: Key Reactor Trip Parameters Used for the FNP Units 1 and 2 RWAP RCS Pressure Analysis

118	High Flux Trip SAL (%)
0.5	High Flux Trip Delay (sec)
2440	High Przr Pressure Trip Setpoint (psia)
1	High Przr Pressure Trip Delay (sec)
9	Positive Flux Rate Trip SAL (%)
2	Positive Flux Rate Trip Time Constant (sec)
0.65	Positive Flux Rate Trip Delay (sec)

The initial conditions and sequence for FNP of events of the worst case analyzed (peak RCS pressure) are:

Initial Conditions:

Power	0.75 fraction of RTP
Pressurizer pressure	2200 psia
Pressurizer water volume	740.15 ft^3
RCS Tavg	575.65 ^oF
Reactivity insertion rate	27 pcm/sec

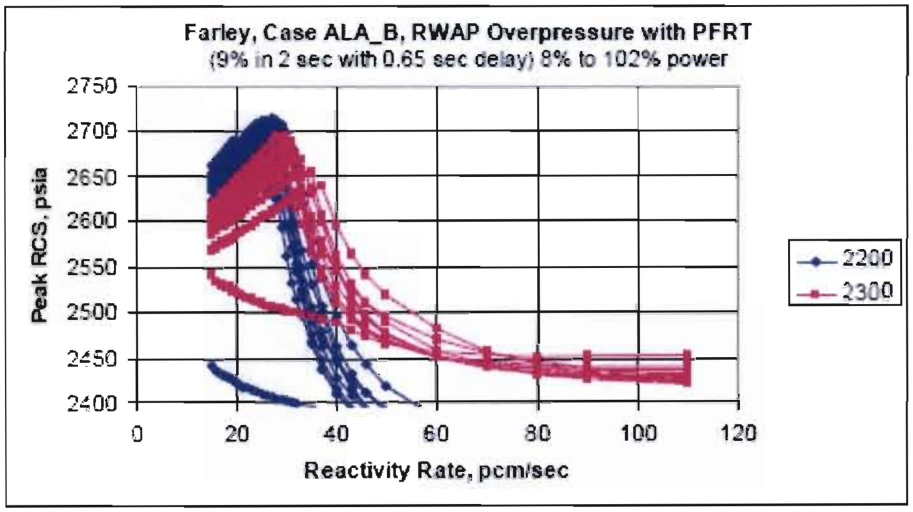
Time Sequence of Events:

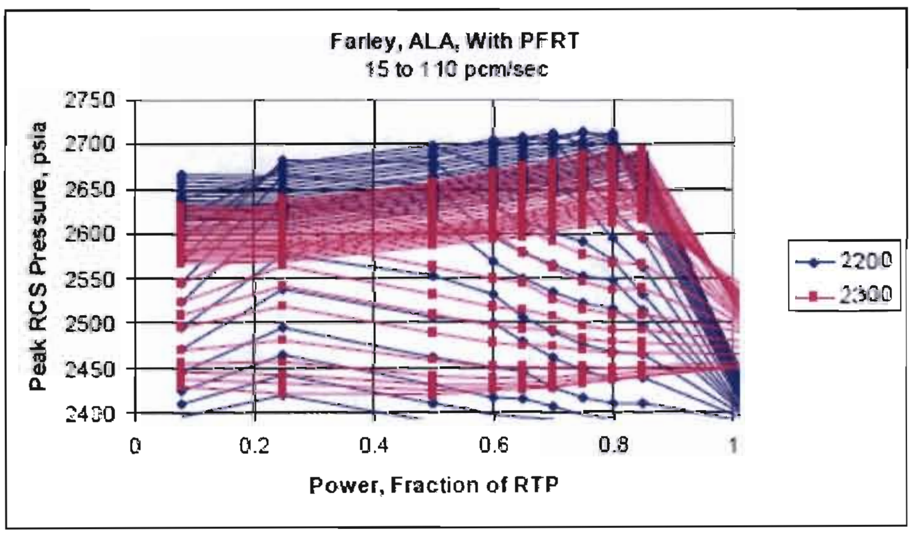
Event	Value	Time (sec)
High PZR pressure trip point reached *	2440 psia	10.17
PFRT trip point reached	9% of RTP	11.10
RCCAs released and began falling	N/A	11.17
Maximum core heat flux reached	1.0994 fraction of RTP	12.20
Maximum RCS pressure reached	2715.50 psia	13.50
High neutron flux trip point reached	118% of RTP	999.99 **

* The RCCAs were released via the high PZR pressure signal; the Farley high PZR pressure trip delay is 1 sec. The RCCAs were released via the PFRT signal for other cases analyzed. The maximum calculated RCS pressure would not have met the analysis acceptance criterion in some of those other cases if the PFRT had not been credited.

** The High Neutron Flux (HNF) trip point was not reached during the event; the "999.99 sec" default indicates that this trip was not reached.

Figure A-1


Farley, Case ALA_B, RWAP Overpressure with PFRT
(9% in 2 sec with 0.65 sec delay) 8% to 102% power
Peak RCS, psia
2750
2700
2650
2600
2550
2500
2450
2400
0
20
40
60
80
100
120
Reactivity Rate, pcm/sec
2200
2300


Farley, ALA, With PFRT
15 to 110 pcm/sec
Peak RCS Pressure, psia
2750
2700
2650
2600
2550
2500
2450
2400
0
0.2
0.4
0.6
0.8
1
Power, Fraction of RTP
2200
2300

Power Range Neutron Flux High Positive Rate Response Time Verification

The PFRT response time of 0.65 second, is explicitly credited in the RWAP analysis and must be met. The definition of Reactor Trip System (RTS) Response Time in the Farley Units 1 and 2 Technical Specifications states: "The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage."

The PFRT response time is verified as per WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" (Reference 3). This approach is based on having performed an initial baseline RTS response time test. Following the initial RTS response time test on this reactor trip function, periodic response time tests are not required consistent with WCAP-14036-P-A, Revision 1, since other required tests ensure that the response time continues to be met.

Since the PFRT response time is essentially the same as the Power Range Neutron Flux – High (PRNF – High) reactor trip function, the PFRT response time can be implicitly verified during the response time test for the PRNF-High reactor trip function as discussed below.

The PFRT circuitry is part of the Nuclear Instrumentation System (NIS). The circuit consists of the difference between the Power Range Nuclear Flux (PRNF) signal and that same signal with a first order lag. That is,

$$\text{PFRT trip signal} = \text{Flux} - \text{Flux}/(1+\tau s),$$

where τ is the time constant and s is the Laplacian operator.

The nominal trip setpoint (difference) for the PFRT reactor trip function contained in FNP TS 3.3.1 is 5 percent of RTP with a time constant (τ) ≥ 2 seconds. Both the PRNF-High and PFRT reactor trip signals are sent from the NIS to the protection system via the same components and are processed from the same PRNF signals and identical bistables. There is no significant time delay for the PFRT added by the additional signal processing to form the difference between the flux signal and the lagged flux signal. Other than having different nominal trip setpoints, the PFRT trip signal sent to the protection system has a response time similar to the PRNF-High reactor trip function.

Therefore, verifying the PRNF-High reactor trip function response time provides reasonable assurance of the PFRT reactor trip function response time, until verification of the PFRT response time can be performed. Adding SR 3.3.1.14 to TS 3.3.1 Table 3.3.1-1 Function 3 would require verification of RTS response times for the PFRT.

SNC has reviewed preoperational test data and confirmed that, during preoperational testing, the response time of the NIS and Solid State Protection System (SSPS) for the PFRT function was measured and recorded. The slowest response time measured for a channel was 0.235 second. Per the Westinghouse analysis, the total function response time limit is ≤ 0.65 second. SNC has confirmed that the PRNF overall function response time is ≤ 0.65 second. This confirmation is based upon the response time measured during preoperational testing of the NIS and SSPS processing time, and the most recent surveillance data for the other components that comprise the string.

4.0 Regulatory Evaluation

4.1 Significant Hazards Consideration

Southern Nuclear Operating Company (SNC) has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on three standards set forth in 10 CFR 50.92(c) as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to Farley Nuclear Plant (FNP) Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation," Table 3.3.1-1, "Reactor Trip System Instrumentation," does not significantly increase the probability or consequences of an accident previously evaluated in the Update Final Safety Analysis Report (UFSAR). The overall protection system performance will remain within the bounds of the accident analysis since there are no hardware changes. The design of the Reactor Trip System (RTS) instrumentation, specifically the power range neutron flux high positive rate trip (PFRT) function, will be unaffected. The reactor protection system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards, that were applicable prior to the request, are maintained.

   The proposed change imposes additional surveillance requirements to assure safety related structures, systems, and components (SSCs) are verified to be consistent with the safety analysis and licensing basis. In this specific case, a response time verification requirement will be added to the PFRT function.

   The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed change will not alter any assumptions nor change any mitigation actions in the radiological consequences evaluations in the UFSAR.

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter nor prevent the ability of SSCs from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change is consistent with the safety analyses assumptions and resultant consequences.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. This change will not affect the normal method of plant operation nor change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance requirements. The additional surveillance requirements are consistent with assumptions made in the safety analyses and licensing basis.

   No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change.

   Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change does not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Limits. There will be no effect on the manner in which Safety Limits or Limiting Conditions of Operations are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

   The safety analyses limits assumed in the accident analysis are unchanged. The imposition of additional surveillance requirements increases the margin of safety by assuring that the affected safety analyses assumptions on equipment response time are verified on a periodic frequency. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standard set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.2 Applicable Regulatory Requirements / Criteria

The following lists the regulatory requirements and plant – specific design bases related to the proposed changes.

- GDC-13 requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

- GDC-20 requires that the protection system(s) shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

- GDC-22 through GDC-25 and GDC-29 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, requirements for reactivity control malfunctions, and protection against anticipated operational occurrences.

- Regulatory Guide 1.22 describes an acceptable method for ensuring that the protection system is designed to permit periodic testing of its functioning during reactor operation.

- 10 CFR 50.55a paragraph (h), "Protection systems," states, in part, that "protective systems must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations."

- 10 CFR 50.36 paragraph (c)(1)(ii)(A), "Safety limits, limiting safety system settings, and limiting control settings" requires limiting safety system settings to be included in the Technical Specifications and to be "so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

- 10 CFR 50.36 paragraph (c)(3), "Surveillance requirements," states "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

4.3 Precedent

Vogtle Technical Specification Amendments 159/141, Unit 1 and Unit 2 respectively, dated February 7, 2011.

Comanche Peak Technical Specification Amendments 151/151, Unit 1 and Unit 2 respectively, dated April 26, 2010.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 Environmental Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment will not:

1. Involve a significant hazards consideration,

2. Result in a significant change in the types, or a significant increase in the amounts, of any effluents that may be released offsite, or

3. Result in a significant increase in individual or cumulative occupational radiation exposure.

SNC has evaluated the proposed changes and determined the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed change is not required.

6.0 References

1. FNP Updated Final Safety Analysis Report, Revision 23.

2. Westinghouse Nuclear Safety Advisory Letter (NSAL) 09-1, "Rod Withdrawal at Power Analysis for Reactor Control System Overpressure." February 4, 2009.

3. WCAP-14036-P-A, Revision 1 "Elimination of Periodic Protection Channel Response Time Tests."

Farley Nuclear Plant Units 1 & 2
License Amendment Request for Technical Specification Table 3.3.1-1

Enclosure 2

Technical Specifications and Bases Markup Pages

Table 3.3.1-1 (page 1 of 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1.	Manual Reactor Trip	1,2	2	B	SR 3.3.1.12	NA	NA
		3 (a), 4 (a), 5 (a)	2	C	SR 3.3.1.12	NA	NA
2.	Power Range Neutron Flux						
	a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≤ 109.4% RTP	≤ 109% RTP
	b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10 SR 3.3.1.14	≤ 25.4% RTP	≤ 25% RTP
3.	Power Range Neutron Flux High Positive Rate	1,2	4	D	SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≤ 5.4% RTP with time constant ≥ 2 sec	≤ 5% RTP with time constant ≥ 2 sec
4.	Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 40% RTP	≤ 35% RTP
		2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 40% RTP	≤ 35% RTP

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Power Range Neutron Flux – High Positive Rate (continued)

Power Range Neutron Flux — High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection event.

The LCO requires all four of the Power Range Neutron Flux — High Positive Rate channels to be OPERABLE. The channels are combined in a 2-out-of-4 trip Logic.

reactivity excursions such as an inadvertent control rod withdrawal or

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux — High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux — High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup.

(continued)

Farley Nuclear Plant Units 1 & 2
License Amendment Request for Technical Specification Table 3.3.1-1

Enclosure 3

Technical Specifications and Bases Clean Typed Pages

Table 3.3.1-1 (page 1 of 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1.	Manual Reactor Trip	1,2	2	B	SR 3.3.1.12	NA	NA
		3 (a), 4 (a), 5 (a)	2	C	SR 3.3.1.12	NA	NA
2.	Power Range Neutron Flux						
	a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≤ 109.4% RTP	≤ 109% RTP
	b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10 SR 3.3.1.14	≤ 25.4% RTP	≤ 25% RTP
3.	Power Range Neutron Flux High Positive Rate	1,2	4	D	SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≤ 5.4% RTP with time constant ≥ 2 sec	≤ 5% RTP with time constant ≥ 2 sec
4.	Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 40% RTP	≤ 35% RTP
		2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 40% RTP	≤ 35% RTP

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Power Range Neutron Flux – High Positive Rate (continued)

Power Range Neutron Flux — High and Low Setpoint trip Functions to ensure that the criteria are met for reactivity excursions such as an inadvertent control rod withdrawal or a rod ejection event.

The LCO requires all four of the Power Range Neutron Flux — High Positive Rate channels to be OPERABLE. The channels are combined in a 2-out-of-4 trip Logic.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux — High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux — High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup.

(continued)