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10 CFR 50.90

December 27, 2018
Serial: RA-18-0261

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: Response to Request for Additional Information Regarding License Amendment Request on Reactor Trip System and Engineered Safety Features Actuation System Instrumentation Trip Setpoints

Ladies and Gentlemen:

By letter dated July 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18211A546), as supplemented by letter dated September 24, 2018 (ADAMS Accession No. ML18267A102), Duke Energy Progress, LLC (Duke Energy), submitted a license amendment request (LAR) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment would modify Technical Specification Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints,” and Technical Specification Table 3.3-4, “Engineered Safety Features Actuation System Instrumentation Trip Setpoints,” to optimize safety analysis margin in the Final Safety Analysis Report Chapter 15 transient analyses.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the LAR and determined that additional information is needed to complete their review. Duke Energy received the request for additional information (RAI) from the NRC through electronic mail on November 28, 2018 (ADAMS Accession No. ML18337A127). Response to this request is required by December 28, 2018.

Attachment 1 provides Duke Energy’s response to the RAI questions. Attachment 2 contains proposed Technical Specification changes associated with supplemental information provided in Attachment 1.

In addition, the No Significant Hazards Consideration provided in the original submittal is being supplemented to account for the additional proposed changes discussed in Attachment 1 and captured in Attachment 2. No regulatory commitments are contained in this letter.

In accordance with 10 CFR 50.91(b), HNP is providing the state of North Carolina with a copy of this response.

Should you have any questions regarding this submittal, or require additional information, please contact Art Zaremba, Manager – Nuclear Fleet Licensing, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 27, 2018.

Sincerely,

Tanya M. Hamilton

Attachments:

1. Response to Request for Additional Information
2. Proposed Technical Specification Changes

cc: J. Zeiler, NRC Senior Resident Inspector, HNP
W. L. Cox, III, Section Chief N.C. DHSR
M. Barillas, NRC Project Manager, HNP
C. Haney, NRC Regional Administrator, Region II

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RA-18-0261

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-400

RENEWED LICENSE NO. NPF-63

(19 pages including cover)

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SRXB RAI #1

In Section 3.2 of the LAR dated July 30, 2018, the licensee discussed the use of a nuclear instrumentation (NI) system component uncertainty (SCU) in determining the high neutron flux trip setpoint. It indicated that a 5% NI SCU was used to determine the current high neutron flux trip setpoint in TS [Technical Specification] Table 2.2-1 as functional unit 2. This NI uncertainty encompassed the reactor vessel downcomer water density and radial power redistribution effects. For the safety analysis, the NRC-approved Duke Energy methodologies explicitly modelled effects such as downcomer attenuation, rod shadow, and power tilt. The licensee claimed that when the Duke Energy methodologies are implemented at HNP, the 5% SCU would be overly conservative. Based on its evaluation, the licensee proposes a NI SCU of 3.2% span reduced from the current 5% span SCU.

Please provide a discussion to explain how the proposed NI SCU of 3.2% span is derived and justify the acceptance of the derivation of the proposed 3.2% uncertainty used to determine the TA [Total Allowance] for the high neutron flux trip setpoint.

Duke Energy Response SRXB RAI #1

The 5% Rated Thermal Power (RTP) NI SCU assumed in determining the current TA for the high neutron flux trip setpoint contains transient NI terms. Transient NI terms such as downcomer attenuation, rod shadow effects, and radial power redistribution (power tilt) are accounted for explicitly in Duke Energy Chapter 15 transient analyses per the NRC-approved Duke Energy methodology DPC-NE-3009, "FSAR/UFSAR Chapter 15 Transient Analysis Methodology" (ADAMS Accession Package No. ML18060A404). The uncertainty associated with these effects is removed from the NI SCU and is applied as a penalty in the transient analyses because the uncertainty is a function of initial reactor power, time in life, and event specific considerations such as power distribution, rod position, and downcomer density changes. Accounting for transient effects within the transient analyses is more accurate than

use of a single penalty, and it is more conservative to treat the uncertainty separately as opposed to reducing it via the square root sum of the squares method with other uncertainties.

With the removal of transient effects from the NI SCU, the term is based on the calibration of the excore detectors to the power calorimetric. Per Technical Specification Table 4.3-1 Note 2, the excore indicated power must remain within 2% RTP of the calorimetric power while operating above 15% RTP. Since the allowance is in absolute terms (% RTP), the allowed relative miscalibration is larger at lower powers.

During startup after a refueling outage, the increase in uncertainty at low power is mitigated by procedural restrictions on the high neutron flux trip setpoint. The power range NI gain is adjusted to predicted beginning of cycle (BOC) values and the high neutron flux trip setpoint is reduced to 85% RTP. The high neutron flux trip setpoint is not increased to the Technical Specification nominal trip setpoint, 108% RTP, until a power calorimetric and flux map are performed at 75% RTP. Reducing the high flux trip setpoint to 85% RTP allows up to (113.5-85)/85 = 33.53% RTP uncertainty before the analytical limit is challenged. While relative uncertainty increases at low power startup conditions, it doesn't increase above 33.5% RTP, meaning that this reduction in the high flux trip setpoint ensures that the 113.5% RTP safety analysis limit is protected.

In the event of a mid-cycle power reduction or shutdown, site procedures limit the extent to which uncertainty may increase. If reactor power decreases below 70% RTP, the gain on the power range NI cannot be reduced without a corresponding reduction in high neutron flux trip setpoint. An increase in NI gain is allowed as it increases indicated power, producing an earlier reactor trip if a severe reactivity transient were to occur. The reduction in the trip setpoint is calculated as follows:

$$\frac{Calculated\ calorimetric\ power\ (\%)}{Avg\ NI\ Power\ Range\ Indicated\ Power\ (\%)} * 108\% = Overpower\ High\ Range\ trip\ Setpoint\ (\%)$$

A 1% reduction in NI power range gain must be therefore paired with a 1% reduction in the trip setpoint when reactor power is below 70% RTP. Thus, the maximum NI miscalibration that can occur without a corresponding decrease in the nominal trip setpoint is at 70% RTP. At 70% RTP, NI may indicate as low as 70 - 2 = 68% RTP, resulting in a maximum miscalibration of (68-70)/(70) = -2.86%. NI gain would need to be increased by 2.86% for NI indicated reactor power to match measured calorimetric power. At 108% RTP, this -2.86% miscalibration reduces indicated power as follows:

[108% RTP – (108% RTP)/(1 – 0.0286)] = -3.18% RTP

The nominal trip setpoint, 108% RTP, is selected for this calculation as the effects of other uncertainties on indicated power are included in the determination of the TA and Safety Analysis Limit (SAL), and transient effects are included directly within the transient analyses. -3.18% RTP

is rounded to -3.2% RTP in the determination of the TA and is treated as a random uncertainty term.

In summary, the 5% RTP NI SCU term is adjusted to 3.2% RTP to account for the ±2% RTP allowance provided in Technical Specification Table 4.3-1 Note 2. Transient terms are calculated on a transient specific basis and applied in the transient analyses. The remaining uncertainty terms used to determine the TA and SAL for the high neutron flux trip are unchanged and remain bounding and conservative.

SRXB RAI #2

In Section 3.5 of the LAR dated July 30, 2018, the licensee indicates that the high power range negative neutron flux rate trip is currently credited in the dropped rod analysis of record (AOR) in Final Safety Analysis Report (FSAR) Section 15.4.3.1. After Cycle 22, when the NRC-approved Duke Energy methodologies are implemented at HNP, this negative flux rate trip would no longer be credited in the analysis for any Chapter 15 events, and the trip currently designed as functional unit 4 in TS Table 2.2-1, would be deleted from the TS table.

Please provide a discussion of the analysis of the dropped rod event performed with the Duke Energy methodologies and demonstrate that for cases without crediting the high power range negative neutron flux rate trip, the results of the dropped rod analysis meet the applicable Chapter 15 accident analysis acceptance criteria.

Duke Energy Response SRXB #2

In Section 5.4.3 of DPC-NE-3009, the dropped rod transient is defined to begin when one or more rod cluster control assemblies (RCCAs) from the same group drop into the core, causing a decrease in reactor power and an increase in power peaking. Section 5.4.3 of DPC-NE-3009 describes five steps to the evaluation of the dropped rod event, which are summarized as follows. The first step involves using an NRC-approved physics code to calculate dropped rod worth, control rod worth available for withdrawal, peaking information, and excore detector response. In the second step, bounding inputs for each of these parameters are selected and input to a RETRAN-3D model for the systems analysis. The third step involves a VIPRE-01 calculation to determine the minimum departure from nuclear boiling ratio (DNBR) using the core thermal-hydraulic boundary conditions from the RETRAN-3D analysis. If a limiting case can be clearly identified, then a fourth step may be used if desired that involves using the limiting RETRAN-3D boundary conditions with the VIPRE-01 model to calculate maximum allowable radial peaking (MARP) limits. In the fifth step, post-drop core power distributions would be compared to the MARP limits to ensure that the DNBR limits are not exceeded. Additionally, the resulting return to power is used to demonstrate that the acceptance criteria for centerline fuel melt (CFM) are satisfied.

The results of representative RETRAN-3D and VIPRE-01 calculations are provided below. These representative calculations do not credit the high power range negative neutron flux rate trip. Each VIPRE-01 case assumes a radial pin peak of 1.66 and a constant axial shape. As

discussed above, evaluation of the MARP limits addresses the post-drop core power distributions.

With the Rod Control System in automatic mode, the limiting single failure involves power signal inputs to the automatic rod control (ARC) and reactor protection system (RPS) (refer to Section 5.4.3 of DPC-NE-3009). Reducing the power signal input to the ARC and RPS conservatively delays the response of these systems and increases the power escalation.

A range of dropped rod worths are evaluated at each burnup to ensure the limiting thermal-hydraulic state-point is represented adequately. Reactor trip only occurs for the higher worth end of cycle (EOC) cases and minimum DNBR occurs well before reactor trip (Table SRXB RAI 2-1). The limiting thermal-hydraulic conditions occur at EOC with a dropped rod worth of 400 pcm. At EOC, dropped rod worths of 410 pcm and 450 pcm yield less restrictive minimum DNBR results. In a dropped RCCA bank event, the power overshoot is less severe than the limiting EOC dropped rod worth due to the greater worth of the RCCA bank. In addition, a dropped RCCA bank yields less of an increase in power peaking due to the symmetric nature of the event.

The following discussion is based on the limiting case at EOC with a dropped rod worth of 400 pcm. As shown in Figure SRXB RAI 2-2 and Figure SRXB RAI 2-3, reactor power decreases rapidly in response to the dropped rod and subsequently recovers as the ARC reacts to this decrease in power by withdrawing Control Bank D. Reactor power overshoots its initial power level and reaches a maximum level of approximately 117% RTP before the ARC begins to insert Control Bank D (Table SRXB RAI 2-1). Minimum DNBR occurs at approximately 33 seconds, which is well before reactor trip and turbine trip (Table SRXB RAI 2-1). Reactor trip occurs on Over-Power Differential Temperature at approximately 42 seconds (Table SRXB RAI 2-1). The reactor trip signal is assumed to stop the motion of Control Bank D. This is a modeling conservatism that reduces the worth available for scram.

The power input signal to the ARC System is also shown in Figure SRXB RAI 2-2. The impact of the quadrant tilt modeling is evident as seen in the peak value of approximately 90% RTP in the signal to ARC compared to the peak core power of approximately 117% RTP. The differences in quadrant tilt modeling between the ARC and RPS signals is also reflected in Figure SRXB RAI 2-2.

This representative dropped rod calculation shows that the high power range negative neutron flux rate trip is not needed. The results of the representative dropped rod analysis meet the applicable Chapter 15 accident analysis acceptance criteria (see Figure SRXB RAI 2-1).

Table SRXB RAI 2-1 Sequence of Events for an EOC case

Event	Time (s)
Start Transient Simulation	0
Rod Drop	1.00E-06
Control Bank D Withdrawal	2.66
Pressurizer Spray Initiates	28.20
Reach Minimum DNB Ratio	33.00
Control Bank D Insertion	37.40
Reactor Trip (OPDT 2/3)	41.87
Turbine Trip	42.87
Pressurizer Spray Terminates	45.08

Figure SRXB RAI 2-1 Demonstration Analysis Results: Minimum DNBR

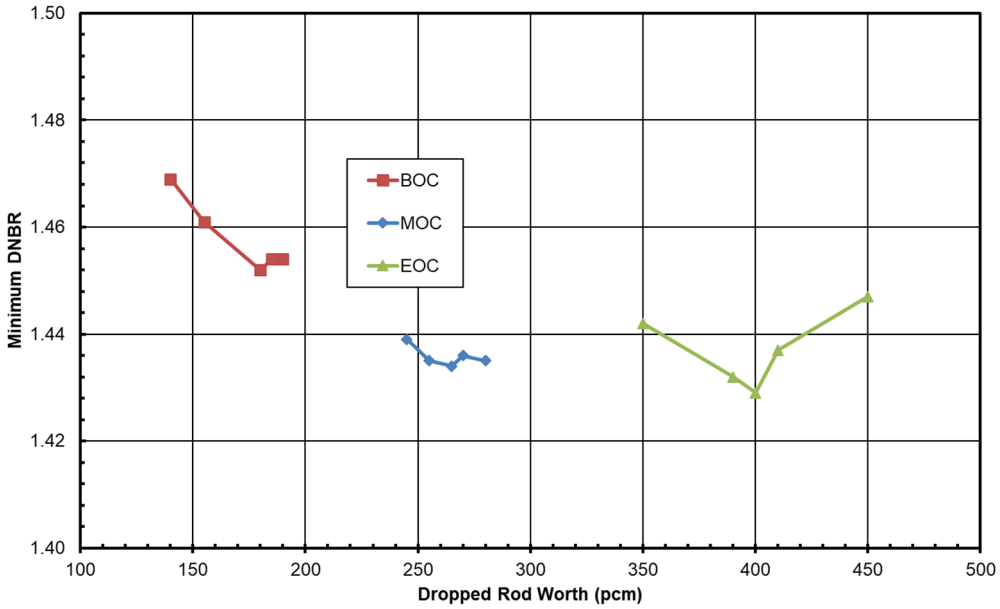


Figure SRXB RAI 2-2 Normalized Actual and Indicated Core Power

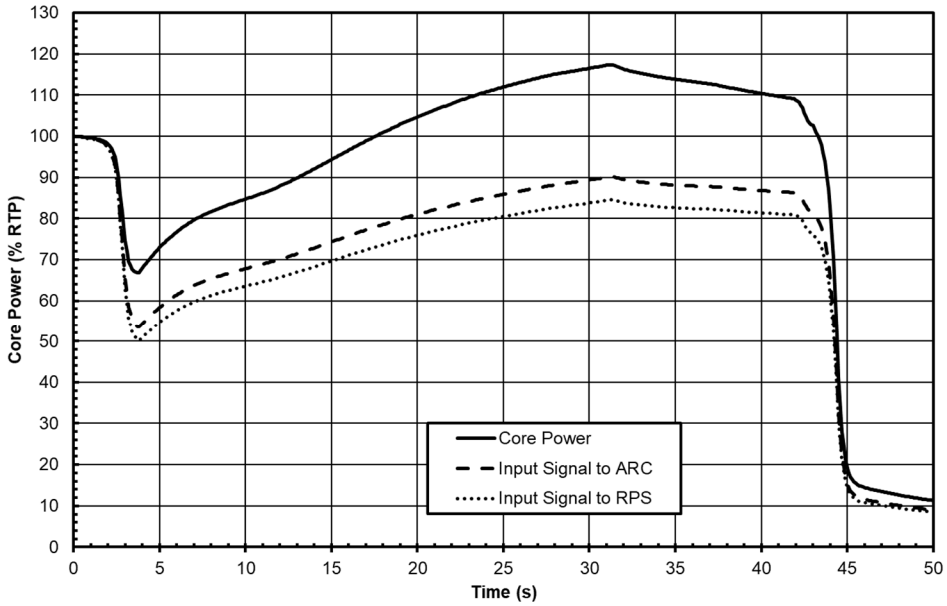
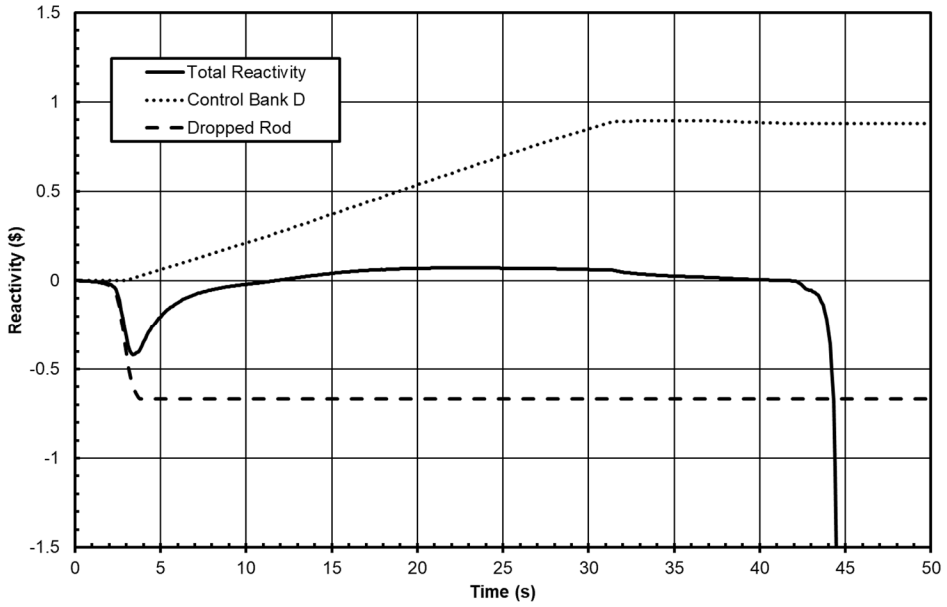


Figure SRXB RAI 2-3 Reactivity (Total, Dropped Rod, Control Bank D)



SRXB RAI #3

In the supplemental information dated September 24, 2018, the licensee identifies the following safety analysis limits (SALs) used to determine the total allowance (TA) for the following functional units in TS Table 2.2-1 or TS Table 3.3-4:

Functional Unit No.12	- Reactor Coolant Flow – Low	(SAL = 88.0% RCS Flow)
Functional Unit No. 2.a	- Power Range, Neutron Flux – High	(SAL = 113.5% RTP)
Functional Unit No. 9	- Pressurizer Pressure – Low	(SAL = 1923 psig)
Functional Unit No. 10	- Pressurizer Pressure – High	(SAL = 2422 psig)
Functional Unit No. 7	- Overtemperature ΔT	(K_3 = 0.1% RTP/psig)
Functional Unit No. 8	- Overpower ΔT	(SAL = 115% RTP)
Functional Unit No.1.d	- Safety Injection, Pressurizer Pressure – Low	(SAL = 1742 psig)

The staff notes the values of the above SALs are different from those assumed in the FSAR Chapter 15 analysis. Please provide a discussion to address the effects of each of the above SALs on the analysis of the FSAR Chapter 15 events and demonstrate that the safety analysis limits are acceptable in meeting the Chapter 15 acceptance criteria.

Duke Energy Response SRXB #3

The revised SALs were proposed with the intent of increasing margin to the FSAR Chapter 15 acceptance criteria. With the exception of Overtemperature ΔT K_3, the revised SALs have been adjusted such that Chapter 15 analyses performed using the Duke Energy transient analysis methodology may credit a reactor trip or safety injection signal on a smaller deviation in the monitored parameter(s). This will result in a reactor trip or safety injection actuation signal earlier in the analyzed transients at less severe conditions relative to an analysis that assumes the currently existing SALs. A comparison of the current SALs and the proposed SALs is provided below for convenience, with markups to FSAR Table 15.0.6-2 associated with these changes and a typo correction to K_2 provided on the following pages.

Functional Unit / Trip Setpoint	Current SAL	Revised SAL
Functional Unit No. 12 Reactor Coolant Flow – Low	85.0% RCS Flow	88.0% RCS Flow
Functional Unit No. 2.a Power Range, Neutron Flux – High	115% RTP	113.5% RTP
Functional Unit No. 9 Pressurizer Pressure – Low	1920 psig	1923 psig
Functional Unit No. 10 Pressurizer Pressure – High	2445 psig	2422 psig
Functional Unit No. 7 Overtemperature ΔT K_3	0.12% RTP/psig	0.1% RTP/psig
Functional Unit No. 8 Overpower ΔT – K_4	118% RTP	115% RTP
Functional Unit No. 1.d Safety Injection, Pressurizer Pressure – Low	1700 psig	1742 psig

TABLE 15.0.6-2 TRIP SETPOINTS AND TIME DELAYS TO TRIP ASSUMED IN AREVA ACCIDENT ANALYSES

Tech. Spec. / COLR Trip Setpoint Item	Tech. Spec. Trip Setpoint	0.0458 Tech. Spec. Total Allowance	Analysis Value	Response Time[a]
Power Range, Neutron Flux			113.5%	
High Setting	≤ 108%	+ (0.0583) (120% of RTP)	115% of RTP	≤ 0.5 sec
Low Setting	≤ 25% of RTP	+ (0.0783) (120% of RTP)	34.4% of RTP	
High Negative Flux Rate	≤ 5% of RTP with a time constant ≥ 2 sec	- (0.0233) (120% of RTP)	-7.8% of RTP with a time constant ≥ 2 sec	≤ 0.5 sec
High Pressurizer Water Level[b]	≤ 87% of instrument span	+ (0.08)(100%)	95% of instrument span	≤ 2.0 sec
High Pressurizer Pressure	≤ 2385 psig	+ (0.075) (800 psi)	2445 psig	≤ 2.0 sec
Low Pressurizer Pressure	≥ 1960 psig	(0.05) (800 psi)	1920 psig	≤ 2.0 sec
Lead Time Constant, τ_4	2.0 sec		1.8 sec[c]	
Lag Time Constant, τ_5	1.0 sec		1.1 sec[(c)]	
Low Primary Coolant Flow	≥ 90.5% of full flow	- (0.0458) (120%)	85% of full flow	≤ 1.0 sec
Low-Low Steam Generator Level	≥ 25.0% of Narrow Range Span	- (0.089) (100%) - (0.25%) (100%)	16.1% span[d] 0.0 span[e]	≤ 3.5 sec
Undervoltage – Reactor Coolant Pumps[f]	≥ 5148 volts			≤ 1.5 sec
Underfrequency – Reactor Coolant Pumps	≥ 57.5 Hz	- (0.05) (10Hz)	57.0 Hz	≤ 0.6 sec
Turbine Trip and Main Feedwater Isolation on SG Water Level – High-High	≤ 78.0% of Narrow Range Span	+ (0.22) (100%)	100% span	≤ 2.5 sec (Turbine Trip) ≤ 10 sec (Feedwater Isolation)
Over Temperature ΔT				
ΔT_0			$\Delta T/\Delta T_0 = 1.0$	4.75 sec RTD lag time and 5.65 sec delay
K_1	1.185	+ (0.09) (150%)	1.32	
K_2	0.224/°F		0.224/°F	
K_3	0.0012/psig		0.0012/psig	
τ_1	0.0 sec		0.0 sec	
τ_2	0.0 sec		0.0 sec	
τ_3	4.0 sec		4.4 sec	
τ_4	22.0 sec	-10%	19.8 sec	
τ_5	4.0 sec	+10%	4.4 sec	

Positive
+ (0.0233)
+7.8%
(0.04625)
2422
1923
91.7%
(0.04625)
(0.0308)
88%
0.0224/°F
0.001/psig
0.0224/°F
0.001/psig

TABLE 15.0.6-2 TRIP SETPOINTS AND TIME DELAYS TO TRIP ASSUMED IN ~~AREVA~~ ACCIDENT ANALYSES

Tech. Spec. / COLR Trip Setpoint

Item	~~Tech. Spec. Trip Setpoint~~	Tech. Spec. Total Allowance	Analysis Value
τ_6	0.0 sec		0.0 sec
$f_1(\Delta I)$	~~0 when +12% ≥ ΔI ≥ -21.6% 1.75% per neg. % ΔI 1.50% per pos. % ΔI~~	See Cycle Specific COLR Value	~~0 when +12% ≥ ΔI ≥ -21.6% 1.75% per neg. % ΔI 1.50% per pos. % ΔI~~
Over Power ΔT			
ΔT_0	1.10	(0.0333)	$\Delta T/\Delta T_0$ = 1.0 1.15
K_4	~~1.12~~	+ ~~(0.040)~~ (150%)	~~1.18~~
K_5	0.02/°F for increasing average temperature 0.0 for decreasing average temperature		0.02/°F for increasing average temperature 0.0 for decreasing average temperature
K_6	0.002/°F for T > T" 0.0 for T ≤ T"		0.002/°F for T > T" 0.0 for T ≤ T"
τ_7	13.0 sec	See Cycle Specific COLR Value	11.7 sec
$f_2(\Delta I)$	~~0.0 for all ΔI~~		~~0.0 for all ΔI~~

[a] The Reactor Trip System Response Time is defined as the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage
[b] The high pressurizer water level trip is only used in the Section 15.2.3 turbine trip overpressure analyses performed by Duke Energy
[c] A specific undervoltage setpoint was not assumed in the analysis
[d] For loss of normal feedwater
[e] For feedwater or main steamline break
[f] A specific undervoltage setpoint was not assumed in the analysis

The proposed change to the Overtemperature ΔT K_3 SAL differs from the other revised SALs in that the change may result in an earlier or later reactor trip depending on analysis specific conditions. From HNP Technical Specification Table 2.2-1, Note 1, the Overtemperature ΔT function is as follows:

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)}\left[T\left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$$

When pressurizer pressure (P) increases above reference pressure (P′), the $K_3(P-P')$ term is positive, and when pressurizer pressure decreases below reference pressure, $K_3(P-P')$ is negative. Due to the reduction in the proposed K_3 SAL, transients in which pressurizer pressure is above reference pressure at the time of reactor trip will see a reduction in the OTΔT trip setpoint and therefore an earlier reactor trip. Conversely, transients in which pressurizer pressure is below reference pressure at the time of reactor trip will see an increase in the OTΔT trip setpoint compared to the current K_3 SAL and a later reactor trip. A review of the Duke Energy transient analysis methodology, DPC-NE-3009, identified transients for which the OTΔT reactor trip is listed as a potential mitigating reactor trip function. A summary of these events is below. Note that transients that were determined to be not applicable to HNP or bounded by another transient per RAI 14 of DPC-NE-3009 (ADAMS Accession No. ML17303B205) are not considered for this evaluation.

Event	**Pressure at Rx Trip**
Section 5.1.1 / FSAR Chapter 15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	P < P′
Section 5.2.1 / FSAR Chapter 15.2.2 Loss of External Electrical Load	P > P′
Section 5.2.2 / FSAR Chapter 15.2.3 Turbine Trip	P > P′
Section 5.2.4 / FSAR Chapter 15.2.7 Loss of Normal Feedwater Flow	P > P′
Section 5.2.5 / FSAR Chapter 15.2.8 Feedwater System Pipe Break	Varies
Section 5.4.2 / FSAR Chapter 15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	P > P′
Section 5.4.3 / FSAR Chapter 15.4.3.1 Dropped Full Length RCCA or RCCA Bank	Varies
Section 5.4.4 / FSAR Chapter 15.4.3.2 Withdrawal of a Single Full Length RCCA	P > P′
Section 5.4.8.4 / FSAR Chapter 15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents	P < P′
Section 5.6.1 / FSAR Chapter 15.6.1 Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve	P < P′
Section 5.6.2, RAI 56 & 57 / FSAR Chapter 15.6.3 Steam Generator Tube Rupture	P < P′

Events in which P > P′ will produce an earlier reactor trip and would increase margin to the

acceptance criteria if the proposed K_3 SAL is implemented. Other events are evaluated individually as follows:

FSAR Chapter 15.1.2 – Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

The limiting case in the current analysis of record occurs at hot zero power (HZP) conditions, with reactor trip occurring on high neutron flux per FSAR Table 15.1.2-4. HZP conditions are expected to remain limiting following the transition to Duke Energy methods, and the mitigating reactor trip function is expected to remain the high neutron flux reactor trip. Overtemperature ΔT is not expected to be the mitigating reactor trip function in the limiting case, so no impact to reactor trip timing is anticipated due to the reduction in the K_3 SAL.

FSAR Chapter 15.2.8 – Feedwater System Pipe Break

The pressurizer pressure response in the feedwater system pipe break analysis varies depending on the acceptance criteria analyzed. However, analysis performed using the Duke Energy methodology has shown that the mitigating safety function for this event is safety injection on Steam Line Pressure – Low (Technical Specification Table 3.3-4, Functional Unit 1.e), which subsequently produces a reactor trip on receipt of the safety injection signal (Technical Specification Table 2.2-1, Functional Unit 18). Overtemperature ΔT is not the mitigating reactor trip function, so no impact to reactor trip timing is expected due to the reduction in the K_3 SAL.

FSAR Chapter 15.4.3.1 – Dropped Full Length RCCA or RCCA Bank

In the dropped rod analysis, pressurizer pressure decreases initially due to a decrease in reactor power from the dropped rod, then increases following automatic rod withdrawal. Analysis performed using the Duke Energy methodology has shown that many cases do not result in a reactor trip. The mitigating reactor trip function for cases which do result in reactor trip is typically Overpower ΔT (see the response to SRXB RAI 2); the change in the K_3 SAL is therefore determined to have a negligible impact on minimum DNBR.

FSAR Chapter 15.4.8 – Spectrum of Rod Cluster Control Assembly Ejection Accidents

In the event a rod ejection transient does not produce a reactor trip on high neutron flux due to a low ejected rod worth, the OTΔT reactor trip is expected to mitigate the event. Analysis using the Duke Energy methodology shows that when the ejected rod worth does not produce a reactor trip on high neutron flux, pressurizer pressure decreases prior to reactor trip due to the hole in the reactor vessel head. Examination of the results shows that the trip timing is primarily a function of the increase in reactor power (ΔT) and resultant increase in RCS temperature, with the decreased pressurizer pressure acting as a secondary factor. The proposed reduction in the K_3 SAL is therefore judged to slightly delay reactor trip on overtemperature ΔT, and does not

significantly reduce margin to the acceptance criteria.

FSAR Chapter 15.6.1 – Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve

The mitigating reactor trip for the limiting case in the current analysis of record is the low pressurizer pressure reactor trip per FSAR Table 15.6.1-4. The analysis performed using the Duke Energy methodology is expected to produce a reactor trip on either low pressurizer pressure or OTΔT. As the SAL for the low pressurizer pressure reactor trip is increased from 1920 psig to 1923 psig and the reduction in OTΔT K_3 is small, the analysis using Duke Energy methods is expected to continue to maintain margin to the DNB acceptance criteria.

FSAR Chapter 15.6.3 – Steam Generator Tube Rupture

For steam generator tube rupture, DPC-NE-3009 Section 5.6.2 and RAIs 56 and 57 (ADAMS Accession No. ML17282A023) provide a method to analyze the core cooling (DNB) and overfill acceptance criteria, and to determine thermal-hydraulic inputs to the offsite dose analysis. For the core cooling analysis, the proposed reduction in the K_3 SAL will slightly delay reactor trip on overtemperature ΔT. However, given that the change in the setpoint is small and steam generator tube rupture is expected to retain substantial margin to the DNB acceptance criteria, the impact of this change is deemed small.

In the steam generator (SG) overfill analysis, the steam generator overfill is largely a function of auxiliary feedwater (AFW) actuation following reactor trip. Operator action to identify the ruptured SG and control AFW is credited to occur 8.8 minutes from transient initiation or when narrow range level reaches 30% on the ruptured generator, whichever is longer. The proposed reduction in the K_3 SAL delays reactor trip, reducing the time the faulted generator is fed by AFW before operators control AFW, therefore increasing margin to the steam generator overfill acceptance criteria.

In the thermal hydraulic inputs to the dose analysis, the offsite dose is largely a function of the steam released to the atmosphere through the SG power-operated relief valve (PORV) on the ruptured SG. Prior to reactor trip, steam from the ruptured SG is removed through the turbine and therefore is not directly released to the atmosphere. A loss of offsite power is assumed coincident with reactor trip, so steam relief following reactor trip occurs through the SG PORVs. Operators isolate the ruptured SG within 12 minutes of transient initiation, then the SG PORV on the ruptured SG is assumed to fail open for an additional 20 minutes until operators close the associated block valve. The proposed reduction in the K_3 SAL delays reactor trip which decreases the time between reactor trip and closure of the block valves for the SG PORV on the ruptured SG. This reduces steam relief from the ruptured SG to the atmosphere and therefore increases margin to the offsite dose acceptance criteria.

FSAR Chapter 15.6.5 – Loss of Coolant Accidents

The FSAR Chapter 15.6.5 Loss of Coolant Accidents (LOCA) are not covered by Duke Energy methods, and will continue to be performed under vendor methods. The revised SALs for the functional units in Technical Specification Table 2.2-1 and Technical Specification Table 3.3-4 are evaluated to either have no impact or a slight benefit to the LOCA analyses. The large break LOCA analysis of record credits safety injection on low pressurizer pressure (Technical Specification Table 3.3-4, Functional Unit 1.d). The increase in the SAL from 1700 to 1742 psig would produce a slightly earlier safety injection signal which would not adversely affect the existing analysis results. However, due to the rapid primary system depressurization during the early portion of the event, safety injection occurs at 0.4 seconds into the transient per FSAR Table 15.6.5-2. The revision to the low pressurizer pressure safety injection SAL is not expected to significantly affect safety injection timing and will be conservatively neglected for Cycle 23.

The small break LOCA (SBLOCA) analysis of record credits reactor trip on low pressurizer pressure (Technical Specification Table 2.2-1, Functional Unit 9) and safety injection on low pressurizer pressure (Technical Specification Table 3.3-4, Functional Unit 1.d). The increase in the low pressurizer pressure reactor trip SAL from 1920 psig to 1923 psig would produce a slightly earlier reactor trip. Similarly, the increase in the low pressurizer pressure safety injection SAL from 1700 to 1742 psig would produce a slightly earlier safety injection signal. Both changes would not adversely affect the existing analysis results. Since the peak cladding temperature occurs much later in the transient at 2060 seconds for the limiting SBLOCA break size (FSAR Table 15.6.5-11a), the impact of the change in the SALs is expected to be small and therefore will be conservatively neglected for Cycle 23.

Based on the discussion above, the revised SALs for the functional units in Technical Specification Table 2.2-1 and Technical Specification Table 3.3-4 will either increase margin to the Chapter 15 acceptance criteria, or, in the case of OTΔT K_3, may result in a slight reduction in existing margin to DNB for steam generator tube rupture and rod ejection. Per Section 2.4 of the LAR, Duke Energy will transition from vendor methods to Duke Energy methods prior to Cycle 23 startup, coincident with implementation of the LAR. The FSAR markups associated with this change will demonstrate that Chapter 15 analyses performed using Duke Energy methods with the proposed SAL changes meet the acceptance criteria.

STSB RAI #1

One of the proposed changes to the TS is deletion of the high power range negative neutron flux rate trip (Functional Unit 4 of Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints). Sections 2.4 and 3.5 of the LAR discuss this proposed change. The LAR states that this trip function is currently credited in the dropped rod analysis of record, but will no longer be credited in any FSAR Chapter 15 accident analysis following the replacement of the current dropped rod analysis with a revised analysis performed in accordance with Duke Energy methodologies.

The LCO 3.3.1 specifies that as a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1, shall be operable. Table 3.3-1, Reactor Trip System Instrumentation, specifies the total number of channels, channels to trip, minimum channels operable, applicable mode, and a reference to the Action statement if the LCO is not met. The Power Range, Neutron Flux High Positive Rate, is included in this Table as Functional Unit 3. The LAR does not propose changes to this requirement.

Currently, Table 2.2-1 specifies the total allowance, statistical summation of analysis errors, trip setpoint and allowable value for the Reactor Trip System Instrumentation automatic trip setpoints. Equation 2.2-1 is used to provide a threshold for evaluation of operability of an instrument channel. With the proposed deletion of the high power range negative neutron flux rate trip from Table 2.2-1, the threshold value for determining proper channel performance would no longer be specified in the TS and it is not clear to the staff what controls will be established to manage the trip setpoint and allowable value in the future.

a. Please provide a description of the controls that will be applied to manage the trip setpoint and allowable value in the future.

b. Please also provide additional information to describe how the requirements of LCO 3.3.1 will be satisfied for this RTS function. Specifically, please explain what value will provide the threshold for evaluating operability, and, if applicable, explain why the threshold for operability is not being retained in the TS. Please also provide an explanation of how this value will be administratively controlled.

Duke Energy Response STSB RAI #1

As discussed in Sections 2.4 and 3.5 of the original submittal, the proposed removal of the high power range negative neutron flux rate trip will eliminate a single point vulnerability of a failed rod control fuse which would result in a rod drop and potentially trigger an automatic reactor trip. Duke Energy had proposed the removal of the high power range high negative neutron flux rate trip, Functional Unit 4, from Technical Specification Table 2.2-1. The original submittal should have also addressed the removal of Functional Unit 4 from Technical Specification Tables 3.3-1, “Reactor Trip System Instrumentation,” and 4.3-1. “Reactor Trip System Instrumentation Surveillance Requirements.”

The original design basis for the high power range negative neutron flux rate trip function was to mitigate the consequences of two or more dropped rod cluster control assemblies (RCCAs), an anticipated operational occurrence (AOO). In the event of two or more dropped RCCAs, the RTS would detect the rapidly decreasing neutron flux (i.e., high negative flux rate) due to the dropped RCCA(s) and would trip the reactor, thus terminating the transient and ensuring that the design DNBR limits were met. Since the dropped RCCAs event is an AOO, it must be shown that the DNBR design limits are met for the combination of high nuclear power, high

radial peaking factor, and other system conditions that exist following the dropped RCCAs event to satisfy General Design Criterion (GDC) 10, “Reactor Design,” requirements.

As documented in the safety evaluation (SE) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2, dated April 29, 2009 (ADAMS Accession No. ML090770181), in 1982, Westinghouse submitted WCAP-10297, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," concluding that the negative flux trip was required only if the plant exceeded a threshold value of reactivity worth, depending on plant design and fuel type. Furthermore, by letter dated May 22, 1987, the Westinghouse Owners Group submitted topical report WCAP-11394-P, "Methodology for the Analysis of the Dropped Rod Event," in which a means was provided to demonstrate that DNBR limits are met during a dropped RCCA event. The analysis using this methodology takes no credit for any direct trip due to the dropped RCCAs, and assumes that no automatic power reduction features are actuated by the dropped RCCAs. The conclusion reached in WCAP-11394-P was that sufficient margin is expected with all Westinghouse plant designs and fuel types, such that the power range neutron flux-high negative rate trip is not required, regardless of the worth of the dropped RCCA (or bank), subject to a plant cycle specific analysis. In a SE dated October 23, 1989 (ADAMS Legacy Accession No. 89110134), the NRC reviewed the Westinghouse analysis and results, concluding that the approach in WCAP-11394-P was acceptable for analyzing the dropped RCCAs event for which no credit is taken for any direct trip, such as the power range neutron flux-high negative rate trip or automatic power reduction features. The NRC's SE noted that further review by the NRC staff for each cycle is not necessary, subject to a licensee's verification that the analysis described in WCAP-11394-P has been performed and makes comparison specified in the topical report with favorable results. An approved version of the topical report (WCAP-11394-P-A) was issued in January 1990.

In developing the Duke Energy FSAR Chapter 15 transient analyses for future cycles utilizing the NRC-approved Duke Energy methodology DPC-NE-3009, “FSAR/UFSAR Chapter 15 Transient Analysis Methodology” (ADAMS Accession Package No. ML18060A404) discussed in the original submittal, the analyses do not credit the high power range negative neutron flux rate trip function. As provided in the response to SRXB RAI #2 above, cases of dropped RCCAs result in the DNBR remaining greater than the safety limit value. As such, the DNBR design acceptance criterion is met and the event does not result in core damage, meeting the GDC 10 fuel integrity requirement. Additionally, as no other safety analyses credit the high power range negative neutron flux rate trip function, no other safety analyses are impacted by this proposed change.

In addition to establishing the technical basis to eliminate Functional Unit 4 of Technical Specification 3/4.3.1, it is necessary to assure that none of the four criteria of 10 CFR 50.36(c)(2)(ii)(A)-(D) were met.

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

HNP Discussion: The high power range negative neutron flux rate trip is not used for detection and indication in the control room of any degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

HNP Discussion: The high power range negative neutron flux rate trip is not an initial condition of a design basis accident or transient analysis.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

HNP Discussion: No credit is taken for the high power range negative neutron flux rate trip in the HNP accident analyses utilizing the NRC-approved Duke Energy methodology, which is to be in effect starting with Cycle 23. Consequently, the high power range negative neutron flux rate trip is not considered part of the primary success path related to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

HNP Discussion: The high power range negative neutron flux rate trip is not relied upon as a signal to initiate a reactor trip for any events modeled in the scope of the probabilistic risk analysis model. The trip function is not significant to public health and safety in that no credit was taken for this trip in any accident analyses utilizing the Duke Energy methodology approved by the NRC.

Therefore, the high power range negative neutron flux rate trip does not meet any of the four criteria of 10 CFR 50.36, and does not warrant inclusion in the HNP technical specifications as a Limiting Condition for Operation.

As a supplement to the No Significant Hazards Consideration Determination provided in Section 4.3 of the original submittal, Duke Energy has further evaluated whether a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below.

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

The removal of the high power range negative neutron flux rate trip function from the HNP Technical Specifications does not involve a significant increase in the probability or consequences of accidents resulting from dropped RCCA events analyzed utilizing the NRC-approved Duke Energy methodology for FSAR Chapter 15 transient analyses, DPC-NE-3009, "FSAR/UFSAR Chapter 15 Transient Analysis Methodology." As demonstrated in the response to SRXB RAI #2, the results of the dropped rod analysis without crediting the high power range negative neutron flux rate trip meet the applicable Chapter 15 accident analysis acceptance criteria. The safety functions of other safety-related systems and components, which are related to mitigation of these events, have not been altered by this change. All other reactor trip system protection functions are not impacted by the deletion of the trip function. The dropped RCCA accident analysis does not rely on the high power range negative neutron flux rate trip to safely shut down the plant. The safety analysis of the plant is unaffected by the proposed change. Since the safety analysis is unaffected, the calculated radiological releases associated with the analysis are not affected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not adversely alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of deleting the high power range negative neutron flux rate trip function. The proposed change does not challenge the performance or integrity of any safety-related systems or components.

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

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

The margin of safety associated with the acceptance criteria of any accident is unchanged. It has been demonstrated that the high power range negative neutron flux rate trip function can be deleted by the NRC-approved methodology described in WCAP-11394-P-A. In utilizing the NRC-approved Duke Energy methodology for FSAR Chapter 15 transient analyses, DPC-NE-3009, it has been demonstrated that the removal of the high power range negative neutron flux rate trip function does not result in exceeding the limits on DNB for dropped RCCA events. The proposed change will have no effect on the availability, operability, or performance of safety-related systems and components.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the additional proposed change to the HNP Technical Specifications does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c). As such, the finding of “no significant hazards consideration” in the original submittal remains justified.

This additional change does not impact the documented Environmental Consideration provided in Section 5.0 of the original submittal.

RA-18-0261

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-400

RENEWED LICENSE NO. NPF-63

(3 pages including cover)

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Not Used	~~4~~ N/A	~~2~~ N/A	~~3~~ N/A	~~1, 2~~ N/A	~~2~~ N/A
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature ΔT	3	2	2	1, 2	6
8. Overpower ΔT	3	2	2	1, 2	6
9. Pressurizer Pressure--Low (Above P-7)	3	2	2	1	6(1)
10. Pressurizer Pressure--High	3	2	2	1, 2	6
11. Pressurizer Water Level--High (Above P-7)	3	2	2	1	6

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	SFCP(12)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	SFCP	SFCP (2,4), SFCP (3,4), SFCP (4,6), SFCP (4,5)	SFCP	N.A.	N.A.	1, 2
b. Low Setpoint	SFCP	SFCP (4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1, 2
4. ~~Power Range, Neutron Flux, High Negative Rate~~ Not Used	N.A.	~~SFCP (4)~~ N.A.	~~SFCP~~ N.A.	N.A.	N.A.	~~1, 2~~ N.A.
5. Intermediate Range, Neutron Flux	SFCP	SFCP (4,5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	SFCP	SFCP (4,5)	S/U(1), SFCP(8)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	SFCP	SFCP (11)	SFCP	N.A.	N.A.	1, 2
8. Overpower ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
9. Pressurizer Pressure -- Low	SFCP	SFCP	SFCP	N.A.	N.A.	1 (16)
10. Pressurizer Pressure -- High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2