DUKE ENERGY®

Scott L. Batson
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672

office: 864.873.3274
fax: 864.873.4208
Scott.Batson@duke-energy.com

ONS-2016-022 10 CFR 50.90

February 26, 2016

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Duke Energy Carolinas, LLC (Duke Energy)

Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed License Numbers DPR-38, DPR-47, and DPR-55

Subject: Response to NRC Request for Information associated with License Amendment Request to Add High Flux Trip for 3 Reactor Coolant Pump Operation License Amendment Request No. 2014-05, Supplement 2

On May 19, 2015, Duke Energy submitted a License Amendment Request (LAR) proposing to add a Reactor Protective System (RPS) Nuclear Overpower - High Setpoint trip for three (3) reactor coolant pump (RCP) operation to Technical Specification Table 3.3.1-1. Duke Energy supplemented the LAR by letter dated August 20, 2015. On January 29, 2016, the Nuclear Regulatory Commission (NRC) issued a Request for Additional information (RAI).

The Enclosure provides responses to the RAI questions. If there are any additional questions, please contact Boyd Shingleton, ONS Regulatory Affairs, at (864) 873-4716.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 26, 2016.

Sincerely,

Scott L. Batson
Vice President
Oconee Nuclear Station

Enclosure: Duke Energy Response to NRC Request for Additional Information
Attachment: Calculation of Total Loop Uncertainty in the High Flux RPS Trip Function

A001
NRR

cc w/enclosure and attachment:

Ms. Catherine Haney
Administrator Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Mr. James R. Hall
Senior Project Manager
(by electronic mail only)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Mail Stop O-8G9A
Rockville, MD 20852

Mr. Jeffrey A. Whited
Project Manager
(by electronic mail only)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Mail Stop O-8B1A
Rockville, MD 20852

Mr. Eddy Crowe
NRC Senior Resident Inspector
Oconee Nuclear Station

Ms. Susan E. Jenkins, Manager, Infectious and Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

ENCLOSURE

Duke Energy Response to NRC Request for Additional Information

Duke Energy Response to NRC Request for Additional Information

RAI-01

The proposed Nuclear Overpower-High Setpoint Trip function allowable value in TS Table 3.3.1-1 has a trip setpoint valid for four RCP operation (≤105.5 percent(%) Reactor Thermal Power (RTP)) and a trip setpoint for three RCP operation (≤80.5% RTP). After going to three RCP operation, overpower protection is initially provided by the Nuclear Overpower Flux/Flow/Imbalance trip function until such a time that the three RCP allowable value is manually entered as the trip setpoint. In its Technical Evaluation provided in Section 4 of the license amendment request (LAR), Duke states, in part, that:

> The existing overpower protection for three RCP operation is the Nuclear Overpower Flux/Flow/Imbalance trip function. However, if [Reactor Coolant System] RCS flow were to increase, as it would for an overcooling event such as a steam line break accident described in UFSAR Chapter 15.17, the flux/flow/imbalance trip setpoint would increase. This increase would result in either a delayed reactor trip or avoidance of a reactor trip altogether.

If the RCS flow situation is such that there could be an "avoidance of a reactor trip altogether," what are the current alternate trip protection functions available?

Duke Energy Response

One of four automatic reactor trip protection functions would cause a reactor trip for the three Reactor Coolant Pump (RCP) Small Steam Line Break (SSLB accident), except for a limited population of break sizes in combination with the time-in-life. They are the flux/flow/imbalance trip, high flux trip, low RCS pressure trip, or variable low pressure-temperature trip. A fifth trip, manual reactor trip, is credited 10 minutes after break initiation if an automatic trip is not actuated. The "avoidance of a reactor trip altogether" is part of the method for determining the limiting break size/time-in-life (i.e., Moderator Temperature Coefficient (MTC)) combination for the SSLB accident. Too large a break with too negative an MTC results in a rapid power excursion that is mitigated by either the flux/flow/imbalance trip function or the existing high flux trip setpoint of 105.5% Rated Thermal Power (RTP) (Technical Specification (TS) allowable value). Too large a break with a less negative MTC value is mitigated by either the low RCS pressure or variable low pressure-temperature trip functions. The limiting break size/time-in-life combination for the existing Reactor Protective System (RPS) (i.e., without the proposed three RCP high flux trip function) and the currently limiting SSLB analysis avoids all four of those RPS trip functions.

RAI-02

In its LAR, Duke proposed Limiting Condition for Operation (LCO) 3.4.4.b, which restricts thermal power to ≤75% when only three RCPs are operating and requires the allowable value of the Nuclear Overpower-High Setpoint Trip function to be reset for three RCPs operating. Duke also proposed a new Condition A to TS 3.4.4 which specifies a 10 hour COMPLETION TIME to perform this reset. The 10 hour COMPLETION TIME is consistent with the Davis-Besse Nuclear Power Station Unit 1 TS, which was referenced as a precedent in Duke's LAR.
Other than the precedent discussed above, please provide additional specific factors considered in the bases for the selection of this 10 hour COMPLETION TIME.

Duke Energy Response

One of the situations which could result in three RCP operation is one in which the plant is initially operating at 100% power with four RCPs in service. An equipment issue is then encountered with one of the RCPs which requires Operations personnel to reduce power in a timely manner to remove the problematic RCP from service. This places the plant at approximately 75% power with three RCPs in service and the Reactor Protective System (RPS) Nuclear Overpower-High Setpoint Trip still at the four RCP value. If this situation occurs outside normal dayshift work hours, qualified maintenance personnel may not be available on site. This would require qualified maintenance personnel to be contacted to come to the site to perform the setpoint change. Once qualified personnel are on site, the needed personnel must obtain needed procedures, work order, and clearances to begin work. The setpoint changes are then implemented by Operations personnel placing each RPS channel in Manual Bypass one at a time, Maintenance personnel performing the change to the Nuclear Overpower-High Trip Setpoint from the Engineered Safeguards/Reactor Protective System (ES/RPS) Service Unit interface for the digital system, and then Operations removing the affected RPS channel from Manual Bypass so that work on the next channel can begin. These needed actions account for the requested 10-hour Completion Time.

RAI-03

In its response to Information Request-4, provided by letter dated August 20, 2015, Duke provided an explanation for the proposed TS allowable value setpoint of 80.5% RTP, by stating, in part, that:

> The 5.5% RTP delta is simply added to the maximum power level allowed for three RCP operation, which is 75% RTP. Adding 5.5% RTP to 75% RTP results in the proposed high flux trip setpoint of 80.5% RTP.

Duke also provided the following formula:

$\Phi m \geq \Phi sp$ + trip setpoint uncertainty allowance

Where Φm = flux measured at excore detectors adjusted for transient effects (e.g., downcomer attenuation) and excore detector calibration tolerances,

Φsp = Technical Specification allowable value trip setpoint

Trip setpoint uncertainty = current analysis assumes 1.0% RTP for convenience since that is the old analog RPS trip bistable uncertainty and it bounds the uncertainty on the setpoint in the digital RPS. There is no uncertainty on the trip setpoint in the digital RPS.

Please provide information (such as an uncertainty calculation) to demonstrate that instrument uncertainty has been accounted for in the establishment of this setpoint. The information provided should show how associated instrument Total Loop Uncertainty (TLU) was considered for the new setpoint.

Duke Energy Response

The original safety analysis trip setpoint methodology developed by Babcock & Wilcox (B&W) in the early 1970's (when Oconee was built) calculated the TS allowable value by starting from a design overpower value of 112% RTP and subtracting off various uncertainty allowances. The historical allowances included were:

2% for heat balance uncertainty

2% for transient excore neutron detector (NI) errors

<u>2.5% for steady-state NI error and reactor trip bistable uncertainty</u>

=6.5% allowance

TS allowable value = 112% RTP – 6.5% RTP = 105.5% RTP

When Duke Energy (DE) assumed responsibility for performing the safety analyses with the submission and acceptance of DPC-NE-3005, UFSAR Chapter 15 Transient Analysis Methodolgoy, DE determined it was more appropriate to start with the TS allowable value for the high flux trip setpoint and account for the above uncertainties differently (see response to NRC Information Request 4 in letter dated August 20, 2015). Of the 6.5% original allowance, each accident analysis explicitly models the transient NI errors and the steady-state nuclear instrumentation (NI) error. The heat balance error is either accounted for in the initialization (non-statistical core design method) or in the DNBR limit itself (statistical core design method). The only remaining uncertainty that must be accounted for is the reactor trip bistable uncertainty (denoted in the digital RPS as the Processor Output Trip Device). The bistable uncertainty is what is referred to as the "trip setpoint uncertainty" in the equation given in RAI-03 above. As stated in the response to NRC Information Request 4 in DE letter dated August 20, 2015, the bistable is an analog component and the uncertainty allowance for it is 1.0% RTP. There is no uncertainty in the trip setpoint itself in the digital RPS.

As demonstrated above, the TLU is not considered in the calculation of the existing TS allowable value. However, the allowable value should still be greater than the allowed power level plus the TLU. The calculation of the TLU is applicable to all power levels since the calculation is performed in %span and converted to %RTP, neither of which are changing and is as follows:

$$\text{TLU} = \pm\sqrt{\text{SS NI Flux Uncertainty}^2 + \text{Heat Balance Uncertainty}^2} = \pm\sqrt{1.176^2 + 2.0^2} = \pm 2.32\%\ \text{RTP}$$

Where: Heat Balance Uncertainty = the 10 CFR 50 Appendix K value of 2.0%

SS NI Flux Uncertainty = the SRSS of the neutron string measurement uncertainty and the long term reactivity changes during normal power operation

Adding 2.32% RTP to the NI calibration tolerance of ± 2% RTP results in a total uncertainty of 4.32% which is bounded by the 5.5% RTP margin in the current TS allowable value for when four RCPs are operating and the proposed TS allowable value for when three RCPs are operating.

The details of the TLU calculation are included in the Attachment to this RAI submittal.

RAI-04

In Section 2 of the LAR, Duke states, in part, that:

> The three (3) RCP trip will provide protection for power excursion events initiated from three (3) RCP operation, most notably the small steam line break accident.

Please provide a list of accidents discussed in the UFSAR that are initiated from three RCP operation (not including the small steam line break (SSLB) since it's thoroughly discussed in the application). Please also briefly describe the impact on these accidents once the new three RCP Nuclear Overpower-High setpoint trip is implemented.

Duke Energy Response

Since three RCP operation is allowed, most of the accidents in Chapter 15 must consider three RCP operation as an initial condition. In addition to SSLB in Section 15.17, all of the accident analyses below consider event initiation from three RCP operation to ensure the limiting initial condition is analyzed.

15.2 – Startup Accident

15.3 – Rod Withdrawal at Power

15.5 – Cold Water Accident

15.6 – Loss of Coolant Flow and Locked Rotor Accidents

15.7 – Control Rod Misalignment Accidents – Dropped Rod

15.8 – Turbine Trip

15.12 – Rod Ejection Accident

Of these analyses, the three RCP initial condition case is either limiting or presented for completeness (denoted) for the following accidents:

15.2 – Startup Accident (three RCP limiting)
15.5 – Cold Water Accident (three RCP limiting)
15.6 – Loss of Coolant Flow and Locked Rotor (presented for completeness)
15.12 – Rod Ejection Accident (presented for completeness)

Regardless of whether the three RCP initial operation case is presented in the UFSAR or not, implementation of the proposed three RCP high flux trip setpoint will either have no impact on the analysis (positive or negative) or will be beneficial if the accident were reanalyzed to credit it, which Duke Energy may opt to credit in the future following NRC approval of the new setpoint. Nevertheless, the above four accidents for which explicit cases are presented in the UFSAR, and the impact of the proposed new trip setpoint on those accidents, are described below.

15.2 – Startup Accident

The accident is initiated from zero power conditions with three RCPs operating and is analyzed for the peak RCS pressure acceptance criterion. The accident postulates the uncontrolled withdrawal of a bank of control rods from this initial condition and, as such, the accident is an over power accident. The UFSAR analysis does not credit the Nuclear Overpower – Low Setpoint (≤ 5%RTP) in effect when reactor power is ≤ 2% RTP. The UFSAR analysis does trip on high power when the excore detectors reach the Nuclear Overpower – High Setpoint (105.5% RTP). If the proposed three RCP setpoint of 80.5% RTP were credited, a lower peak RCS pressure would result. Therefore, implementation of the three RCP setpoint would be an analysis benefit.

15.5 – Cold Water Accident

The accident postulates the start of the fourth RCP from three RCP operation, 80% RTP initial conditions and is analyzed for the departure from nucleate boiling ratio (DNBR) acceptance criterion. A conservatively fast increase in pump flow is assumed upon start of the fourth RCP which causes core power to rapidly increase above 100%RTP. However, reactor trip does not occur on either the Nuclear Overpower – High Setpoint (105.5% RTP) or any other trip function. If the proposed three RCP setpoint of 80.5% RTP were modeled, reactor trip would occur almost immediately following the conservatively fast ramp up of the fourth RCP unless the initial power level were reduced such that the proposed setpoint was just avoided, in which case the DNBR results improve significantly relative to the current UFSAR analysis, which is already acceptable. Therefore, implementation of the three RCP setpoint would be an analysis benefit.

15.6 – Loss of Coolant Flow and Locked Rotor

The accident postulates the loss of one or more RCPs in the event of a loss of coolant flow or the instantaneous seizure of a pump shaft in one RCP in the event of a locked rotor. All of the cases result in a rapid reduction in flow but no increase in power. Since power level does not increase, implementation of the three RCP setpoint would be inconsequential to the analysis results.

15.12 – Rod Ejection Accident

The accident postulates the ejection of the highest worth control rod assembly from the core at three RCP 77% RTP and three RCP hot zero power (HZP) conditions (in addition to four RCP 102% RTP). The event is analyzed for peak fuel enthalpy, DNBR (fuel failures), and peak RCS pressure. The rod ejection from HZP initial conditions does not credit the Nuclear Overpower – Low Setpoint (≤ 5%RTP) in effect when reactor power is ≤ 2% RTP. Due to the rapid nature of the power excursion, crediting the proposed three RCP setpoint would provide minimal benefit to the worst case analysis results.

RAI-05

In its response to NRC Information Request-3, provided by letter dated August 20, 2015, Duke states, in part, that:

> The analysis of the three RCP SSLB with the proposed high flux trip setpoint for when three RCPs are operating demonstrates that true core power is significantly reduced before reactor trip occurs.

With the new trip implemented, please estimate the time when the trip is expected to occur and the maximum actual core power for the current limiting case in the UFSAR. Would changing any of the initial conditions cause a significant delay or avoid exceeding the new trip setpoint completely? If so, would any of these cases be the limiting DNB case for 3 RCP operation?

Duke Energy Response

Q: With the new trip implemented, please estimate the time when the trip is expected to occur and the maximum actual core power for the current limiting case in the UFSAR?

A: The current UFSAR Chapter 15.17 analysis documents the worst case SSLB. It is a SSLB initiated from three RCP, 75% RTP. The peak power is approximately 118%RTP and no RPS trip setpoint is exceeded in the first 10 minutes of the simulation, at which time manual reactor trip is credited. For approximately the same break size/MTC combination as the current UFSAR analysis, the reanalysis with the proposed three RCP high flux setpoint set at 80.5 %RTP results in a maximum actual power level of approximately 102% RTP with a reactor trip approximately 81 seconds into the transient. If the break size/MTC sensitivity is redone with the three RCP high flux trip setpoint to determine the combination that avoids all reactor trips, the reanalysis results in a new steady-state actual power level of approximately 105% RTP. If the break size/MTC sensitivity is redone to determine the combination that yields the worst DNBR, reactor trip occurs on the proposed setpoint of 80.5% RTP when the actual core power reaches approximately 109 %RTP approximately 2 minutes after break initiation.

Q: Would changing any of the initial conditions cause a significant delay or avoid exceeding the new trip setpoint completely? If so, would any of these cases be the limiting DNB case for 3 RCP operation?

A: Yes, it is possible to delay or avoid exceeding the new trip setpoint if the initial conditions, specifically, the initial power level changed. However, as explained below, the delay and or avoidance of the new trip setpoint is not expected to result in a worse minimum DNBR.

The main initial conditions are RCS flow, hot leg pressure, RCS average temperature, core power, and time in core life. Secondary initial conditions are core average fuel temperature, pressurizer level and steam generator mass. Sensitivity cases are performed to ensure the most conservative initial conditions are modeled for a given initial power level, especially the time in core life assumption as modeled by the MTC initial condition. RCS flow, hot leg pressure, and RCS average temperature are independent of initial core power. Therefore, the most significant initial condition is the initial core power, particularly as it affects the initial operating margin to the proposed high flux trip, and the MTC assumption which is subject to sensitivity cases with break size to determine the worst assumption.

The analysis starts at the maximum allowed power level for three RCP operation but could just as easily be performed at lower initial powers without affecting the results significantly. The reason is that sensitivities are performed on break size and MTC such that 1) reactor trip is avoided or, 2) if reactor trip occurs, the core power excursion is maximized at the time of reactor trip. In the case of avoiding reactor trip, changing the initial power level will not significantly change the new sustained steady-state power level of approximately 105% RTP since the break size/MTC combination is chosen to just avoid all reactor trips, particularly the low RCS pressure trip. To obtain the same terminal, steady-state power level, a new break size/MTC sensitivity would need to be performed. A larger break would result in a greater depressurization and most likely result in a low RCS pressure trip actuation. Therefore, a more negative MTC would be required to cause a larger power excursion and subsequent steady-state power level. The overcooling of the primary system would not be affected if the break size is not changed significantly and, consequently, downcomer attenuation effects will not shield the excore detectors more than the analysis initiated from higher power. Therefore, the MTC chosen would still be constrained by the high flux trip setpoint and the resultant steady-state power would be approximately the same. Thus, starting the analysis from a lower initial power level will not result in a significantly different steady-state power level 10 minutes after break initiation.

In the case of maximizing the core power excursion, the reanalysis shows that the low RCS pressure trip would be exceeded shortly after reaching the new setpoint of 80.5% RTP. This suggests the break size cannot be any larger than the reanalysis assumed or the power excursion would be terminated prematurely by the low pressure trip. The MTC could become more negative resulting in more of a power excursion, however, since the analysis would be starting at a lower initial power level, a greater power excursion is required to trigger the 80.5% high flux trip setpoint. Since break size cannot increase without causing a low RCS pressure trip, a more negative MTC would be required (or the same MTC assumed resulting in a shallower power excursion slope) to reach the new high flux trip setpoint. As with the case in avoiding reactor trip altogether, since break size cannot increase, the overcooling will not change significantly and downcomer attenuation will not alter the excore NI response. Consequently, actual core power will not be significantly different at the time of reactor trip as it would be starting from 75% RTP. So while the timing of reactor trip is expected to change, the DNB parameters at the time of reactor trip are not expected to be significantly different if the analysis were started from a lower initial power level. Consequently, the minimum DNBR will not be significantly different for different initial conditions. The reanalysis demonstrates that the DNBR results for the worst three RCP case improve enough that the existing DNBR results for the four RCP analysis become limiting.

ATTACHMENT

Calculation of Total Loop Uncertainty in the High Flux RPS Trip Function

Calculation of Total Loop Uncertainty in the High Flux RPS Trip Function

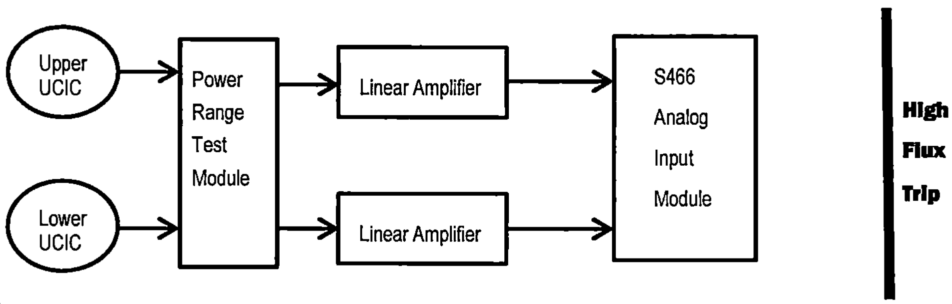
This attachment provides the details of the TLU calculation provided in the response to RAI-03 of the enclosure to this submittal. The calculation is organized as follows: Section 1 contains the instrument block diagram of the RPS high flux trip function; Section 2 contains the RPS high flux reactor trip function uncertainty analysis; and, Section 3 contains the setpoint analysis.

1.0 Instrument Block Diagram

Figure 1-1 shows a schematic block diagram of the high flux RPS reactor trip function instrumentation.

FIGURE 1-1

INSTRUMENT BLOCK DIAGRAM



2.0 RPS High Flux Reactor Trip Function Uncertainty Analysis

The signals (corresponding to the measured neutron flux) from the excore neutron detectors (Westinghouse UCICs) are directed through a power range test module (see Note below) and linear amplifiers to a S466 Analog Input Module. Output from the module is used by the Teleperm software which sums/averages the signals from the top and bottom detectors to obtain the total measured neutron flux signal.

NOTE: The power range test module is used to switch in a simulated neutron flux signal during calibration. The actual measured neutron flux signal (from the excore detector) merely passes through the power range test module to the linear amplifier, and it is assumed that negligible error is introduced to the measured neutron flux signal.

2.1 Full Power Steady State NI Flux Error

The steady state NI flux error term accounts for the neutron measurement string instrument uncertainties and for long term reactivity changes in the core which could occur in-between calibrations. These effects include burnup and poison depletion which can affect the flux shape in the core and thus the measurement of the flux by the excore neutron detectors.

During the NI power-to-best estimate thermal power calibration the total uncertainty for the NI power string is obtained by combining the following uncertainty components:

- Individual string component (linear amplifier and S466 Analog Input Module) uncertainties
- Calibration effect (associated with the NI power-to-best estimate thermal power calibration)
- Uncertainty in the simulated neutron flux test signal from the power range test circuit

A. Individual Module Uncertainties

Linear Amplifier (Lin_Amp_Unc)

The linear amplifier uncertainties include reference accuracy, drift, relative humidity, range adjustment, zero adjustment, temperature and power supply effects.

Linear Amplifier Reference Accuracy (Lin_Amp_RA)

Lin_Amp_RA = 0.1% span

Linear Amplifier Drift (Lin_Amp_D)

Lin_Amp_D = 0.06% span

Linear Amplifier Relative Humidity Effect (Lin_Amp_RH)

Lin_Amp_RH = 0.4% span

Linear Amplifier Range Adjustment (Lin_Amp_RAdj)

Lin_Amp_RAdj = 0.1% span

Linear Amplifier Zero Adjustment (Lin_Amp_ZAdj)

Lin_Amp_ZAdj = 0.001% span

Linear Amplifier Temperature Effect (Lin_Amp_TE)

Lin_Amp_TE = 0.08% span (The temperature effect is treated as a bias)

Linear Amplifier Power Supply Effect (Lin_Amp_PSE)

Lin_Amp_PSE = 0.02% span (The power supply effect is treated as a bias)

Total Linear Amplifier Uncertainty (Lin_Amp_Unc)

$$\text{Lin\_Amp\_Unc} = \text{Lin\_Amp\_TE} + \text{Lin\_Amp\_PSE} + \sqrt{\text{Lin\_Amp\_RA}^2 + \text{Lin\_Amp\_D}^2 + \text{Lin\_Amp\_RH}^2 + \text{Lin\_Amp\_RAdj}^2 + \text{Lin\_Amp\_ZAdj}^2}$$

$$\text{Lin\_Amp\_Unc} = 0.08\% + 0.02\% + \sqrt{0.1^2 + 0.06^2 + 0.4^2 + 0.1^2 + 0.001^2} = 0.528\% \text{ span}$$

S466 Analog Input Module

The following terms affect the S466 Analog Input Module

S466 Reference Accuracy (S466_RA)

Specified for the voltage input range as ± 0.2% span.

S466_RA = 0.2% span

S466 Temperature Effect (S466_TE)

Specified as ± 0.0028% span/°F about a reference temperature of 73.4 °F. The normal expected environmental conditions in the control complex are 74 - 80 °F. Accounting for an assumed +20°F temperature increase in the instrumentation cabinets means the equipment may see temperatures in the range 94 - 100°F. Therefore the temperature effect is:

S466_TE = 0.0028% (100-73.4) = 0.074% span

S466 Digital Signal Processing (S466_DSP)

The Digital Signal Processing uncertainty consists of the SRSS combination of linearity (± 0.2% span), tolerance (± 0.05% span) and polarity reversal (± 0.05% span) errors.

$$\text{S466\_DSP} = \sqrt{\text{Linearity}^2 + \text{Tolerance}^2 + \text{Polarity}^2} = \sqrt{0.02^2 + 0.05^2 + 0.05^2} = 0.073\%\ \text{span}$$

S466 Analog Input Module Total Uncertainty (S466_Total_Unc)

$$\text{S466\_Total\_Unc} = \sqrt{\text{S466\_RA}^2 + \text{S466\_TE}^2 + \text{S466\_DSP}^2} = \sqrt{0.2^2 + 0.074^2 + 0.073^2}$$
$$= 0.226\%\ \text{span}$$

For the purposes of calculating the string Allowable Value the S466 uncertainty is calculated without the temperature effect.

$$\text{S466\_AV\_Unc} = \sqrt{\text{S466\_RA}^2 + \text{S466\_DSP}^2} = \sqrt{0.2^2 + 0.073^2} = 0.213\%\ \text{span}$$

B. Calibration Effect (CE)

The overall calibration effect (CE) includes consideration for measuring and test equipment uncertainties (M&TE), calibration tolerance effects (CTE) and technician readability or resolution (RES). The calibration effect is determined using the following equation: CE = $(\text{M\&TE}^2 + \text{CTE}^2 + \text{RES}^2)^{1/2}$. Note: the resolution term (RES) is included in the measurement and test equipment term (MTE) below.

Measurement &Test Equipment Uncertainty (M&TE)

The calibration is performed by verifying the output voltage of the power range test module with an Agilent 34401A DMM and reading the output on the OAC display. Agilent 34401A DMM has an uncertainty of ± 0.00076 Vdc on the 0 - 10 volt range. The OAC has a resolution of 0.01 % RTP on a 0 to 62.5 % RTP scale.

MTE_Agilent = 0.00076 Vdc

Agilent_Vdc_range = 10- 0 Vdc

$$\text{MTE\_Agilent\_Unc} = \frac{\text{MTE\_Agilent}}{\text{Agilent\_Vdc\_range}} = 0.0076\% \text{ span}$$

The OAC indication for % RTP shows a display resolution of 0.01% RTP. Therefore, the OAC resolution in % span is:

$$\text{MTE\_OACRes} = \left(\frac{0.01}{62.5}\right) x\ 100\% = 0.016\%$$

$$\text{MTE\_XS\_Unc} = \sqrt{\text{MTE\_Agilent\_Unc}^2 + \text{MTE\_OACRes}^2} = \sqrt{0.0076^2 + 0.016^2} = 0.018\% \text{ span}$$

Calibration Tolerance Effect (CTE)

The power calibration tolerance will be assumed to be equal to the SRSS of the XS rack components reference accuracies.

$$\text{CTE\_Unc} = \sqrt{\text{Lin\_Amp\_RA}^2 + \text{S466\_RA}^2} = \sqrt{0.1^2 + 0.2^2} = 0.224\% \text{ span}$$

Calibration effect (CE)

Combining the above uncertainties results in the calibration effect calculated below.

$$\text{CE\_Unc} = \sqrt{\text{MTE\_XS\_Unc}^2 + \text{CTE\_Unc}^2} = \sqrt{0.018^2 + 0.224^2} = 0.225\% \text{ span}$$

C. Uncertainty In The Simulated Test Signal

During calibration the power range test circuit is used to simulate the neutron flux signal. The error in the simulated signal results from the module uncertainty of the power range test circuit and the calibration effect (which includes the calibration tolerance effect of the simulated test signal and the measuring and test equipment uncertainty).

Based on the previously installed test circuit module, the worst case component uncertainty of the power range test circuit is ± 0.035% span.

PWR_RNG_Test = 0.035% span

Measurement &Test Equipment Uncertainty For Simulated Test Signal (M&TE_Test)

The simulated test signal check is performed using a Fluke model 45 digital volt meter (DVM) which has an uncertainty of ± 0.0072 vdc on the 0 - 10 volt range or An Agilent model 34401A DVM which has an uncertainty of ± 0.00076 vdc on the 0 - 10 volt range. The largest of these two uncertainties will be used as a bounding M&TE term.

Fluke_range = 10- 0 Vdc

$$\text{MTE\_Fluke45} = 0.0072 \text{ Vdc}$$

$$\text{MTE\_Unc\_Test} = \frac{\text{MTE\_Fluke45}}{\text{Fluke\_range}} = 0.072\% \text{ span}$$

Calibration Tolerance Effect (CTE)

The power calibration tolerance is ± 0.005 Vdc on a 0 - 10 volt span.

$$\text{Test\_Signal\_Span} = 10 \text{ Vdc}$$

$$\text{CTE\_Unc\_Test} = \frac{0.005}{\text{Test\_Signal\_Span}} = 0.05\% \text{ span}$$

Total Uncertainty in the Simulated Signal (Power_Test_Unc)

Combining the above uncertainties results in the total uncertainty in the power range test signal.

$$\text{Power\_Test\_Unc} = \sqrt{\text{PWR\_RNG\_Test}^2 + \text{MTE\_Unc\_Test}^2 + \text{CTE\_Unc\_Test}^2}$$
$$= \sqrt{0.035^2 + 0.072^2 + 0.05^2} = 0.094\% \text{ span}$$

D. Total Uncertainty At The S466 Analog Input Module

The total channel power uncertainty at the S466 Analog Input Module is obtained by combining the above uncertainty components (i.e. individual module uncertainties, calibration effect, and the uncertainty in the simulated test signal associated with the power range test circuit) via the SRSS methodology.

Linear Amplifier Output Uncertainty (ULAS)

The uncertainty in the linear amplifier output signal (ULAS) is obtained by combining the linear amplifier module uncertainty and the simulated test signal uncertainty.

$$\text{ULAS} = \sqrt{\text{Lin\_Amp\_Unc}^2 + \text{Power\_Test\_Unc}^2} = \sqrt{0.528^2 + 0.094^2} = 0.537\% \text{ span}$$

S466 Analog Input Module Output Summation Uncertainty (SUM_Unc)

The uncertainty in the output of the S466 Analog Input Module due to the two linear amplifier input signals is calculated below. Since the XS Processor software is summing the outputs from the S466 Analog Input Module to calculate a total power signal from the upper and lower UCIC signals, gains of 0.5 are used on each linear amplifier input signal.

$$\text{SUM\_Unc} = \sqrt{(0.5 \times \text{ULAS})^2 + (0.5 \times \text{ULAS})^2} = \sqrt{0.2685^2 + 0.2685^2} = 0.38\% \text{ span}$$

Total Power Uncertainty at the S466 Analog Input Module Output to the Processor Output Trip Device

The total uncertainty at the processor output trip device of the XS Processor (Total_Flux_Unc) is obtained by combining uncertainty in the S466 Analog Input Module Output summation, S466 Analog Input Module uncertainty and the calibration effect as follows:

$$\text{Total\_Flux\_Unc} = \sqrt{\text{SUM\_Unc}^2 + \text{S466\_Total\_Unc}^2 + \text{CE\_Unc}^2} = \sqrt{0.38^2 + 0.226^2 + 0.225^2}$$
$$= 0.495\% \text{ span}$$

Total scale span of NI detectors (TotalFlux_Scale_Span) equals 125% RTP

Total_Flux_Power_Unc = Total_Flux_Unc x TotalFlux_Scale_Span = 0.495 x 125% = 0.619% RTP

The total instrument uncertainty at the processor output trip device of the XS Processor given above includes the calibration effect and the uncertainty associated with the simulated neutron flux signal. As expected this value is slightly less than the neutron measurement instrument uncertainty for the previously installed equipment (1% RTP).

Per the original B&W uncertainty calculations, it is reasonable to assume that the long term reactivity changes will not exceed 1% RTP during steady state operation.

LTerm_Reac_Unc = 1% RTP

The total flux power uncertainty and the long term reactivity effect are combined via SRSS to arrive at the total steady state NI flux error calculated below.

$$\text{SS\_NI\_Flux\_Unc} = \sqrt{\text{Total\_Flux\_Power\_Unc}^2 + \text{LTerm\_Reac\_Unc}^2} = \sqrt{0.619^2 + 1.0^2}$$
$$= 1.176\% \text{ RTP}$$

2.2 Process Measurement Errors

A. Heat Balance Error:

An allowance for calorimetric heat balance uncertainty is required by Reg. Guide 1.49, which states that the accident analysis calculations must be performed at a power level 2% greater than rated power to account for uncertainties in the determination of power level through the heat balance calculation. Thus, a 2% FP allowance is allotted to account for the uncertainty associated with the calorimetric heat balance measurement of reactor power.

Heat_Balance_Unc = 2% RTP

B. Transient NI Error:

The transient induced error term accounts for variations in the flux shape (caused by ICS induced movement of control rods or control rod drop event) and variations in the downcomer coolant temperature which affect leakage neutrons measured by the excore detectors.

These transient NI error effects are accounted for in the safety analysis on a transient specific basis for those events that result in perturbed flux shapes and/or variations in the downcomer coolant temperature and therefore, are not accounted for here.

2.3 High Flux Trip Total Loop Uncertainty

The following summarizes the applicable error terms associated with the high flux trip function:

A. Steady State NI Flux Error

The steady state NI flux error term includes the hardware (power range test circuit, linear amplifiers and S466 Analog Input Module) effect and the long term reactivity effects.

SS_NI_Flux_Unc = 1.176% RTP

B. Calorimetric Heat Balance Error

Heat_Balance_Unc = 2% RTP

C. High Flux Trip Total Loop Uncertainty (TLU_High_Flux_Trip)

The total loop uncertainty of the high flux trip function is determined by combining the above error terms, via SRSS, as follows:

$$\text{TLU\_High\_Flux\_Trip} = \sqrt{\text{SS\_NI\_Flux\_Unc}^2 + \text{Heat\_Balance\_Unc}^2} = \sqrt{1.176^2 + 2.0^2}$$
$$= 2.32\%\ \text{RTP}$$

3.0 Setpoint Analysis

The total allowance for uncertainty used in the safety analyses for the high flux trip setpoint is:

Heat Balance Allowance = 2.0% RTP
+ NI Calibration Allowance = 2.0% RTP
+ Processor Output Trip Device (formerly known as the bistable) allowance = 1.0% RTP
+ Transient specific allowance = accounted for in the transient itself
Total Allowance = 5% RTP + transient specific effects

This 5% RTP allowance is larger than the high flux trip function TLU of 2.32% RTP + NI calibration allowance of 2.0% RTP (= 4.32% RTP) and confirms that a conservative high flux trip setpoint has been selected.