PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
Nuclear LLC

JUN 02 2015

10 CFR 50.90

LR-N15-0124
LAR S15-02

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

Salem Nuclear Generating Station Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and 75
NRC Docket Nos. 50-272 and 50-311

Subject: Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems (TAC Nos. MF6065 and MF6066)

References

1. PSEG letter to NRC, "License Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems," dated April 3, 2015 (ADAMS Accession No. ML15093A291)
2. NRC letter to PSEG, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 - Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems (TAC Nos. MF6065 and MF6066)," dated May 14, 2015 (ADAMS Accession No. ML15127A287)

In the Reference 1 letter, PSEG Nuclear LLC (PSEG) submitted a license amendment request for Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The proposed amendment would revise technical specification (TS) 3/4.3.1, "Reactor Trip System Instrumentation," to support the planned replacement of the existing source range (SR) and intermediate range (IR) nuclear instrumentation.

In the Reference 2 letter, the U.S. Nuclear Regulatory Commission staff requested that PSEG supplement the application with information necessary to enable the NRC staff to begin its detailed technical review. The requested information is provided in Attachment 1.

PSEG has determined that the information provided in this submittal does not alter the conclusions reached in the 10 CFR 50.92 no significant hazards determination previously submitted. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Mr. Brian Thomas at 856-339-2022.

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

Executed on JUN 02 2015
(date)

Respectfully,

John F. Perry
Site Vice President
Salem Nuclear Generating Station

Attachment 1 - Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems (TAC Nos. MF6065 and MF6066)

Enclosure 1 - Analysis of SC-NIS001-01 and SC-NIS002-01

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. C. Parker, NRC Project Manager, Salem
NRC Senior Resident Inspector, Salem
Mr. P. Mulligan, Manager IV, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. Cachaza, Salem Commitment Tracking Coordinator

Attachment 1

Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems (TAC Nos. MF6065 and MF6066)

1. Provide the equipment qualification information for the detectors, amplifiers, and the control room electronics for harsh and mild environments.

   a) For equipment in harsh environments, provide the qualification information that meet the guidance of Title 10 of the Code of Federal Regulations (10 CFR) Part 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants."

   PSEG Response:

   For the harsh environment, the Nuclear Instrumentation System (NIS) source and intermediate range detectors and associated cables inside containment were evaluated to the 10CFR50.49 criteria above and were exempted from the Salem Equipment Qualification (EQ) Program requirements. The exempt classification status for the NIS detectors was submitted to the NRC as part of the design basis review for Salem EQ equipment. Based on this analysis, the source and intermediate range detectors in a harsh environment were exempted from the requirements of 10 CFR 50.49 and not included in the Salem EQ program.

   Although this equipment was exempted from the requirements of 10 CFR 50.49, PSEG Specification S-C-DE-NIS-0210 states that the qualification of this NIS equipment shall be demonstrated in accordance with S-C-ZZ-SDC-1419, “Salem Generating Station Environmental Design Criteria,” which requires testing per IEEE-323-1974.

   b) For equipment in non-harsh environments, provide the qualification information that protects the detectors, amplifiers, and the control room electronics, for example, by meeting the guidance of Regulatory Guide (RG) 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants." (ADAMS Accession No. ML070190294)

   PSEG Response:

   There are no computer based control components within the replacement NIS that are critical to performance of its intended functions (Refer to the response to Question 3 for more information). For components installed in a mild environment, the new replacement NIS equipment is being supplied to meet the mild environment criteria identified in PSEG Specification S-C-DE-NIS-0210 as follows:

Normal Operating and Abnormal Conditions

Parameter	Normal Min	Normal Max	DBA Max
Temperature	55°F	85°F	85°F
Humidity	20%	90%	90%
Pressure	0.125”WC	0.125”WC	0.125”WC
Radiation, Gamma Rate (mr/hr)	N/A	0.25	Not Calculated
Radiation, Gamma TID (r)	N/A	87.7	884

If generic qualification documentation is used, explain how the generic qualification meets the site-specific qualification requirements for the applicable location of the equipment.

PSEG Response:

As documented in PSEG Specification S-C-DE-NIS-0210, if previous generic qualification documentation is provided by the vendor, the vendor is to provide a summary report or certificate of compliance for all testing performed. This documentation will be reviewed as part of the design change process to ensure it envelopes the PSEG criteria.

2. Please document how the equipment meets the guidance of RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," (ADAMS Accession No. ML003740218) for electromagnetic and radio-frequency interference, or provide justification for meeting an alternative.

PSEG Response:

The replacement NIS components provided by Thermo Fisher Scientific are equipped with design features to limit their susceptibility to EMI/RFI. The proposed replacement NIS will use only analog and logic components. Linear power supplies will be used due to the low power requirements of the simple electronics. These linear power supplies have no oscillating components and a very high immunity to EMI/RFI. Bistable circuits are filtered to limit spurious effects. Bistable outputs are also isolated from the other cabinets. The logic outputs are electrically isolated from other circuits.

Testing, qualification, and analysis of the replacement NIS is in accordance with EPRI TR-102323, Guidelines for Electromagnetic Compatibility Testing of Power Plant Equipment, Rev 2. As documented in RG 1.180, Revision 1, the NRC staff accepted the Electric Power Research Institute (EPRI) topical report TR-102323, in a Safety Evaluation Report (SER) by letter dated April 17, 1996, as one method of addressing issues of electromagnetic compatibility (EMC).

3. Please state whether or not equipment uses software or embedded electronic components with software. If such components are used, then document how the equipment meets the software common cause failure guidance in Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," (ADAMS Accession No. ML070380094) or provide justification for meeting an alternative.

PSEG Response:

The only microprocessor based device associated with the replacement NIS is the Shutdown Margin Monitor (SDMM). The remaining portion of the NIS system is analog. The shutdown margin monitor continuously measures and displays the count rate from the neutron flux monitor. The indication and alarm function provided by the SDMM is located in the Control Equipment Room. It is separate and independent from the audible count rate indication provided in the Main Control Room. The SDMM does not provide a function that is required for system operability in accordance with the Technical Specifications.

The power to the SDMM assembly is via a fuse in the SDMM that provides the isolation between the SDMM and the power source associated with the Source Range (SR) Monitor. The circuit isolation between SDMM and SR is provided by two relays. The SDMM provides ±12v to LEDs for local indication of the status of the instrument power and does not interface with any other components. The SR signal processor provides a pulse signal to the SDMM through an isolation device in the SR channels. Therefore, there are no failure modes associated with SDMM that could affect the ability of the SR channels to perform their intended function.

4. Please provide the setpoint calculation or the following additional information pertaining to the setpoint calculation:

   a) Limiting trip setpoint for Source Range (SR) and Intermediate Range (IR).

      PSEG Response:

      The Setpoint Analysis (Enclosure 1) describes calculation of the Source Range and Intermediate Range trip setpoints and allowable values. The SR and IR setpoints are maintained at the existing Technical Specification setpoint values. The SR and IR trip functions do not have an analytical limit since they are not credited in the Salem accident analysis as discussed in response to Question 7.

   b) Drift over the calibration period and the length of the calibration interval.

      PSEG Response:

      The equipment vendor (Thermo Fisher) provided a 24 month rack drift of ±1.0 % of span. The data provided by Thermo Fisher is included in the Setpoint Analysis (Attachment 1 of Enclosure 1). The current calibration interval for the Source Range and Intermediate Range systems is 18-months in accordance with the Surveillance Frequency Control Program.

   c) Please provide the basis for changing the allowable values for intermediate range neutron flux from ≤ 30% to ≤ 38.5% of rated thermal power and for the source range neutron flux allowable value from ≤ 1.3×10^5 to ≤ 1.44×10^5 counts per second.

      PSEG Response:

      The source range indication scale changes from 10^0 - 10^6 cps to 10^{-1} - 10^6 cps increasing the span from six to seven decades. The intermediate range indication scale changes from 10^{-11} – 10^{-3} amps (equivalent to 10^{-6} to 120% Rated Thermal Power) to 10^{-8} – 200% Rated Thermal Power.

      The total rack uncertainty increases as span increases, requiring a change in the allowable value. As described in the Setpoint Analysis (Enclosure 1), the allowable values for the Source Range and Intermediate Range neutron flux were changed to account for an increased rack uncertainty. The increased rack uncertainty is the product of the increased instrument span.

5. Item B on top of page 5 of attachment 1 to the license amendment request states, "Due to the changes in the IR detector output and units, an assessment was completed to verify adequate coordination between the P-6 setpoint and the SR neutron flux reactor trip setpoint for the Thermo Scientific instrumentation." Please provide a summary of this assessment for NRC staff review.

   PSEG Response:

   Section 3.7.1.3 of Enclosure 1 provides a discussion of the P-6 Interlock. Figure 1 of Enclosure 1 provides a graphical representation of the margin between the P-6 setpoint and the SR neutron flux reactor trip setpoint.

6. The license amendment request does not include all the Applicable Regulatory Requirements/Criteria. Please address the requirements and criteria listed below. Salem Updated Final Safety Analysis (UFSAR) Section 3.1.1, "Conformance with AEC [Atomic Energy Commission] General Design Criteria (July 1971)," states that the Salem Plant design conforms with the intent of the General Design Criteria (GDC) for nuclear power plants with the exceptions listed.

   a) Please address how Salem meets the intent of GDC 1, GDC 4 (with exceptions listed in UFSAR Section 3.1.1), GDC 13, and GDC 19, as it applies to this change request.

      PSEG Response:
      Salem's conformance with the AEC General Design Criteria (July 1971) is described in UFSAR Section 3.1.3.

      GDC 1 requires structures, systems, and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The replacement SR and IR instrumentation is safety related. The design, procurement, installation, and operation of the replacement instrumentation is subject to the requirements of the PSEG quality assurance program.

      GDC 4 requires structures, systems, and components important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. The replacement instrumentation is designed to function in the environmental conditions associated with normal operation, maintenance, testing, and the postulated accidents for which it is credited.

      GDC 13 requires instrumentation to 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. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges. Following implementation of the proposed changes, Salem will continue to meet the intent of GDC 13.

      GDC 19 requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a

safe condition under accident conditions, including loss-of-coolant accidents. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. Following implementation of the proposed changes, Salem will continue to meet GDC 19.

b) Please address 10 CFR 50.36, "Technical Specifications," and 10 CFR 50.49.

PSEG Response:

The reactor trip system instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

As discussed in response to Question 1, the SR and IR were exempted from 10 CFR 50.49.

c) In addition, RG 1.105, RG 1.180, and BTP 7-19, should be addressed, as applicable.

PSEG Response:

RG 1.105, Instrument Setpoints For Safety-Related Systems – Revision 2 of RG 1.105 endorsed ISA S67.04-1982 for use in establishing and maintaining setpoints in safety-related systems. The methodology used to calculate the Salem trip setpoints is consistent with ISA-S67.04-1982.

RG 1.180, Revision 1. “Guidelines For Evaluating Electromagnetic And Radio-Frequency Interference In Safety-Related Instrumentation And Control Systems” – As discussed in response to Question 2, testing, qualification, and analysis of the replacement NIS is in accordance with EPRI TR-102323, “Guidelines for Electromagnetic Compatibility Testing of Power Plant Equipment.”

BTP 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," – As discussed in the response to Question 3, the only microprocessor based device associated with the replacement NIS is the Shutdown Margin Monitor (SDMM). There are no failure modes associated with SDMM that could affect the ability of the SR channels to perform their intended function.

7. Page 3 of Attachment 1 indicated that "No credit is taken for the reactor trips associated with either the SR or IR channels in the accident analyses described in Chapter 15 of the Salem UFSAR." However, SR readings and/or its associated reactor trip signals are credited for some accident analysis at some plants. Please provide information addressing why UFSAR Chapter 15 analysis for each of the events (including a Boron dilution event) will not be affected by the proposed TS changes. A list of reactor trips credited for each event confirming that the SR and IR instrumentation have no role in the transient analyses would help address this.

PSEG Response:

As discussed in the Salem TS Section 2.0 bases:

> The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

No SR or IR trip functions are listed in UFSAR Table 15.1-3 which provides the limiting trip setpoints assumed in the Chapter 15 accident analyses. The table below lists the reactor trips credited for each event.

UFSAR Section	Accident	Reactor Trip Initiator
15.2.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	Power Range High Neutron Flux (low setting)
15.2.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Power Range High Neutron Flux (118% of Nominal Full Power) Overtemperature ΔT Overpower ΔT
15.2.3	Rod Cluster Control Assembly Misalignment	Reactor trip is not credited
15.2.4	Uncontrolled Boron Dilution	Operator Action based upon Audible Count Rate*
15.2.5	Partial Loss of Forced Reactor Coolant	Low Flow in Any Loop (>36% Power) Low Flow in Any Two Loops (Between 11% and 36% Power)
15.2.6	Deleted from the UFSAR	N/A
15.2.7	Loss of External Electrical Load and/or Turbine Trip	Turbine Trip High Pressurizer Pressure High Pressurizer Water Level Low-Low Steam Generator Water Level Overpower ΔT Overtemperature ΔT No Direct or Immediate Trip Credited For a Loss of External Electrical Load without Turbine Trip.
15.2.8	Loss of Normal Feedwater	Low-Low Steam Generator Water Level
15.2.9	Loss of Offsite Power to the Station Auxiliaries	Low-Low Steam Generator Water Level

UFSAR Section	Accident	Reactor Trip Initiator
15.2.10	Excessive Heat Removal Due to Feedwater System Malfunctions	Power Range High Neutron Flux Overtemperature ΔT Overpower ΔT
15.2.11	Excessive Load Increase Incident	Power Range High Neutron Flux Overtemperature ΔT Overpower ΔT
15.2.12	Accidental Depressurization of the Reactor Coolant System	Pressurizer Low Pressure Overtemperature ΔT.
15.2.13	Accidental Depressurization of the Main Steam System	Overpower Reactor Trips (Neutron Flux and ΔT)
15.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates the ECCS	Pressurizer Low Pressure
15.3.2	Minor Secondary Pipe Breaks	No Reactor Trip
15.3.3	Inadvertent loading of a Fuel Assembly into an Improper Position	No Reactor Trip
15.3.4	Complete Loss of Forced Reactor Coolant Flow	Reactor Coolant Pump Bus Undervoltage Low Primary Coolant Loop Flow (>36% Power) Low Flow in Any Two Loops (Between 11% and 36% Power
15.3.5	Single Rod Cluster Control Assembly Withdrawal at Full Power	Overtemperature ΔT
15.3.6	Accidental Release of Waste Gases	No Reactor Trip
15.3.7	Accidental Release of Radioactive Liquids	No Reactor Trip
15.4.1	Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)	Pressurizer Low Pressure
15.4.2	Major Secondary System Pipe Rupture	The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

UFSAR Section	Accident	Reactor Trip Initiator
15.4.3	Major Rupture of a Main Feedwater Line	a. High pressurizer pressure, b. Overtemperature ΔT, c. Low-low steam generator water level in any steam generator, d. Safety injection signals from any of the following: 1. High steam flow coincident with low steam line pressure 2. High containment pressure 3. High steam line differential pressure 4. Low pressurizer pressure 5. High steam flow coincident with low-low T-avg
15.4.4	Steam Generator Tube Rupture	Pressurizer Low Pressure Overtemperature ΔT
15.4.5	Single Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break	Low Reactor Coolant Flow
15.4.6	Fuel Handling Accident	No Reactor Trip
15.4.7	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	Power Range High Neutron Flux Power Range Neutron Flux High Positive Rate

*For the uncontrolled boron dilution event, UFSAR section 15.2.4 notes that the operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. The Source Range and Intermediate Range upgrades do not affect the audible count rate instrumentation. The UFSAR also states that high count rate is alarmed in the reactor containment and the Control Room. The UFSAR does not provide for a High Count Rate alarm setpoint.

PSEG previously submitted a license amendment request (Reference 1) requiring isolation of unborated water sources and to allow use of Gamma-Metrics Post-Accident Neutron Monitors during Mode 6 (Refueling). The proposed changes in the Reference 1 LAR would permit the replacement of the SR channels in Mode 6; however, approval of this NIS SR and IR replacement LAR is not contingent on approval of the Reference 1 LAR. Absent approval of the Reference 1 LAR, the SR instrumentation replacement could only be performed In Mode 6 with core alterations suspended or with the Reactor defueled. The replacement of the Westinghouse source range detectors with the Thermo-Scientific source range detectors will continue to satisfy the current TS Mode 6 instrumentation requirements.

<u>References</u>

1. PSEG letter to NRC, " License Amendment Request to Isolate Unborated Water Sources and Use Gamma-Metrics Post-Accident Neutron Monitors during Mode 6 (Refueling)," dated March 9, 2015 (ADAMS Accession No. ML15068A359)

Enclosure 1

Analysis of SC-NIS001-01 and SC-NIS002-01

Analysis of SC-NIS001-01 and SC-NIS002-01
Analysis in Support of LAR S15-002 – SR/IR Replacement

Prepared by: _[signature]_ Date: 03/19/2015
Luis Gonzalez, S&L

Reviewed by: _[signature]_ Date: 3/19/2015
John Wilkens, S&L

PSEG Review by: _[signature]_ Date: 3/25/15
(print name) BHUPENDRA MEHTA

PSEG Approval by: _[signature]_ Date: 03/25/15
(print name) Michael J Rickers

1.0 Introduction

The new Thermo Fisher Scientific signal processors, herein designated as Rack, are designed to replace the existing source and intermediate range drawers of the Nuclear Instrumentation System (NIS). The processor will combine the functions of the existing Source Range Neutron Flux (SR) and Intermediate Range Neutron Flux (IR) drawers into a single drawer. Therefore, where required, calculations are performed to reflect the new instrumentation specifications performance including: (1) Total Loop Uncertainty (TLU), (2) As Left Tolerance (ALT) and As Found Tolerance (AFT)/Allowable Value (AV), and its effect in (3) Nominal Trip Set Points (NTSP).

The calculations use a statistical combination method (square root of the sum of the squares) in references 4.2 and 4.5 for independent random terms, and algebraic combination for dependent terms and bias effects, to determine the uncertainties. The calculations in percent of span is percent of Equivalent Linear Full Scale (% ELFS, % span).

2.0 Source Range

The source range indication scale will change from 10^0-10^6 cps to 10^{-1}-10^6 cps, that is, the source range scale changes from six to seven decades.

2.1 Design Inputs

Technical Specifications Surveillance Requirements

	Trip Setpoint	Allowable Value
Source Range Neutron Flux	≤ 1.0 x 10^5 counts	≤ 1.3 x 10^5 counts

Rack Design Inputs
(Ref. 4.3, Attachment 1):

Rack Accuracy ($RCA_{\_Source}$)	= ± 2.00% span
Rack Measurement & Test Equipment ($RMTE_{\_Source}$)	= ± 0.05 % span
Rack Comparator Accuracy ($RCSA_{\_Source}$)	= ± 0.25% span
Rack Temperature Effects ($RTE_{\_Source}$) (RCA includes temperature effects)	= ± 0.00% span
Rack Drift ($RD_{\_Source}$)	= ± 1.00% span

Span Scale Inputs

Span (Number of Decades)	= 0.1 - 10^6 cps (7 decades)

Process Design Inputs

Process Measuring Uncertainty ($PMA_{\_Source}$) (Note)	= ± 10.00% span

Note: Per reference 4.2, Section 7.1, Process Measurement Uncertainty consists of water density variations in the downcomer due to Reactor Coolant temperature

variations, radial power redistributions, detector Drift, Sensor Calibration Tolerances and Sensor Temperature Effects. These are combined such that the uncertainty is random and independent and is equal to ± 10% span.

2.2 Rack Uncertainty ($Rack_{\_Source}$)

Source Range Neutron Flux Rack Uncertainty is the statistical combination of the rack's random independent accuracy terms:

$$Rack_{\_Source} = (RCA_{\_Source}^2 + RMTE_{\_Source}^2 + RCSA_{\_Source}^2 + RTE_{\_Source}^2 + RD_{\_Source}^2)^{1/2}$$

$$Rack_{\_Source} = (2.00^2 + 0.05^2 + 0.25^2 + 0.00^2 + 1.00^2)^{1/2}$$

$$Rack_{\_Source} = \pm 2.25\% \text{ span}$$

2.3 Total Loop Uncertainty ($TLU_{\_Source}$)

The Rack Total Loop Uncertainty (TLU) is determined statistically combining rack uncertainty and the process measurement effects.

$$TLU_{\_Source} = \pm (Rack_{\_Source}^2 + PMA_{\_Source}^2)^{1/2}$$

$$TLU_{\_Source} = \pm (2.25^2 + 10.00^2)^{1/2}$$

$$TLU_{\_Source} = \pm 10.25\% \text{ span}$$

2.4 As Found Tolerance

The As Found Tolerance, also termed Allowable Value, is determined on instrumentation based performance, TSTF-493, which is defined by the combination of the rack uncertainty terms comprising Accuracy, MTE, and Drift terms.

$$AFT\_Rack_{\_Source} = \pm (RCA_{\_Source}^2 + RMTE_{\_Source}^2 + RCSA_{\_Source}^2 + RD_{\_Source}^2)^{1/2}$$

$$AFT\_Rack_{\_Source} = \pm (2.00^2 + 0.05^2 + 0.25^2 + 1.00^2)^{1/2}$$

$$AFT\_Rack_{\_Source} = \pm 2.25\% \text{ span}$$

2.5 As Left Tolerance

The As Left Tolerance, is determined on instrumentation based performance, TSTF-493, and is defined by the combination of the Accuracy and MTE terms.

$$ALT\_Rack_{\_Source} = \pm (RCA_{\_Source}^2 + RMTE_{\_Source}^2 + RCSA_{\_Source}^2)^{1/2}$$

$$ALT\_Rack_{\_Source} = \pm (2.00^2 + 0.05^2 + 0.25^2)^{1/2}$$

$$ALT\_Rack_{\_Source} = \pm 2.02\% \text{ span}$$

2.6 Setpoint Analysis

In this section a review of the replacement system accuracies in cps, is performed per below equation.

Setpoint (cps) = Process Value (cps) x $10^{(CU \% / 100\% \times b)}$

where:

a = base decade operation (-1)
b= number of decades of operation (7)
CU = channel uncertainty (in % span)

The Technical Specifications setpoints are set in process values, or counts per second for the logarithmic base 10 scale; therefore, the calculated values are applied to the trip setpoint and converted into counts per second and percent of linear scale for a numerically familiar verification. The conversion is made as per below equation:

Percent Linear Scale = 100 x ($\log_{10}$(cps) - a) / 7

2.6.1 Instrumentation Setpoints

As Found Tolerance / TS Allowable Value

The determination of the As Found Tolerance, or Allowable Value, for the Technical Specifications setpoint by applying the AFT_Rack_{Source} uncertainty is:

$+ AFT = 1 \times 10^5 \times 10^{[(2.25\% / 100\%) \times 7]}$

$+ AFT = 1.44 \times 10^5$ cps

$- AFT = 1 \times 10^5 \times 10^{[(-2.25\% / 100\%) \times 7]}$

$- AFT = 6.96 \times 10^4$ cps

In percent of linear scale / volts:

$+ AFT = [\log_{10} 1.44 \times 10^5 - (-1)] / 7$

+ AFT = 87.96% or 8.796 volts (note)

$- AFT = [\log_{10} 6.96 \times 10^5 - (-1)] / 7$

- AFT = 83.46% or 8.346 volts (note)

Note: Per figure 1, the 0%-100% linear scale is equivalent to 0-10 volts.

Where the current TS Allowable Value of 1.3×10^5 cps in percent of linear scale is:

$$TS\_AV_{\_Current} = \log_{10}[1.30 \times 10^5 - (-1)] / 7$$

$$TS\_AV_{\_Current} = 87.34\%$$

The calculated value, is slightly larger than the TS value by a difference of 0.62% (87.96% - 87.34%). See section 2.7 for conclusions.

2.6.2 As Left Tolerance

The determination of the As Left Tolerance is made by applying the $ALT\_Rack_{\_Source}$ uncertainty:

$$+ ALT = 1 \times 10^5 \times 10^{[(2.02\% / 100\%) \times 7]}$$

$$+ ALT = 1.38 \times 10^5 \text{ cps}$$

$$- ALT = 1 \times 10^5 \times 10^{[(-2.02\% / 100\%) \times 7]}$$

$$- ALT = 7.22 \times 10^4 \text{ cps}$$

The positive ALT in percent linear scale:

$$+ ALT = \log_{10}[1.38 \times 10^5 - (-1)] / 7$$

$$+ ALT = 87.73\%$$

2.6.3 Analytical Limit

There is not a defined Analytical Limit (AL); therefore, the Analytical Limit is set as the 100% scale, upper range value of the system, or 1×10^6 cps, and it is verified that the setpoint(s) does not fall outside of the system range.

$$\text{Calculated Setpoint} = AL - TLU_{\_Source}$$

and Margin:

Margin = Calculated Setpoint - TS Setpoint

The setpoint calculated in Percent Scale is:

$$\text{Calculated\_Setpoint} = 100\% \text{ span} - 10.25\% \text{ span}$$

$$\text{Calculated\_Setpoint} = 89.75\% \text{ span}$$

Margin between TS Setpoint and Calculated Setpoint:

Margin = Calculated Setpoint – Setpoint

where current TS setpoint of 1×10^5 cps in percent linear scale is:

$TS\_SPT_{\_Current} = \log_{10}[(1 \times 10^5 - (-1)] / 7$

$TS\_SPT_{\_Current} = 85.71\%$

Thus the margin is:

Margin = 89.75% - 85.71%

Margin = + 4.04% span

There is a positive margin from the TS setpoint to the calculated setpoint, and to the Analytical Limit; therefore, the current TS setpoint is acceptable.

2.7 Conclusions

The current allowable value of ≤ 1.30 x 10^5 cps should be updated to the new rack performance by the calculated AFT of ≤ 1.44 x 10^5 cps.

3.0 Intermediate Range Neutron Flux Instrument Uncertainty

The Intermediate Range scale will change from 1 x 10^{-11} to 1 x 10^{-3} amperes (equivalent to 1 x 10^{-6} to 120% RTP power, Figure 2) to 1 x 10^{-8} to 1 x $10^{2.30103}$ % (200%) RTP power, the change increases the intermediate range. Per observation of the systems scales correlation, Figure 1, the Intermediate Range has 5 decades overlap with the Source Range system scale.

3.1 Design Inputs

Technical Specifications Surveillance Requirements

Table 2.2-1

	Trip Setpoint	Allowable Value
Intermediate Range Neutron Flux	≤ 25% of Rated Thermal Power	≤ 30% of Rated Thermal Power

Table 3.3-1

	Condition and Setpoint	Function
P-6	With 2 of 2 Intermediate Range Range Neutron Flux Channels <6 x 10^{-11} amps (*)	P-6 prevents or defeats the manual block of source range reactor trip

(*) 6 x 10^{-11} amps in the intermediate range monitor is the P-6 reset allowable value for the bistable reset value of 7 x 10^{-11} amps

Rack Design Inputs
(Ref. 4.3, Attachment 1):

Rack Accuracy ($RCA_{\_Inter\_Range}$)	= ± 1.50% span
Rack Measurement & Test Equipment ($RMTE_{\_Inter\_Range}$)	= ± 0.05 % span
Rack Comparator Accuracy ($RCSA_{\_Inter\_Range}$)	= ± 0.25% span
Rack Temperature Effects ($RTE_{\_Inter\_Range}$)	= ± 0.00% span

(RCA includes temperature effects)
Rack Drift ($RD_{\_Inter\_Range}$) = ± 1.00% span

Span Scale Inputs

Span (Number of Decades) = 10^{-8} - $10^{2.3}$ (200) % (10.3 decades)

Process Design Inputs

Process Measuring Uncertainty ($PMA_{\_Inter\_Range}$) (Note) = ±5.558% RTP random -8.240% RTP bias

Note:

Per reference 4.5, Section 5.2.1, the four PMA terms calculated/allowed for the Intermediate Range channels are:

1) Power Calorimetric ± 2.0% RTP
2) Downcomer Temperature ± 3.3% RTP
3) Radial Power Distribution ± 4.0% RTP
4) Rod Shadowing - 8.24% RTP (bias) (based on 35% RTP)

While the Rod shadowing PMA was determined at 35% it will be applied to the 25% setpoint to determine the magnitude of the effect in the calibrated setpoint value.

Terms (1), (2) and (3) are considered independent of each other and (4) is considered as an independent parameter. The total uncertainty for these four parameters is the SRSS of their magnitudes for the three random components and adding the bias term, or:

$(2.0\%^2 + 3.3\%^2 + 4.0\%^2)^{1/2} - 8.24\% = \pm 5.558\%$ RTP (random) / - 8.240% RTP (bias)

3.2 Rack Uncertainty ($Rack_{\_Inter\_Range}$)

Intermediate Range Neutron Flux Rack Uncertainty is the statistical combination of the rack's random independent accuracy terms:

$$Rack_{\_Inter\_Range} = (RCA_{\_Inter\_Range}^2 + RMTE_{\_Inter\_Range}^2 + RCSA_{\_Inter\_Range}^2 + RTE_{\_Inter\_Range}^2 + RD_{\_Inter\_Range}^2)^{1/2}$$

$Rack_{\_Inter\_Range} = \pm 1.82\%$ span

3.3 Process Measurement Uncertainty ($PMA_{\_Inter\_Range}$)

The Technical Specifications setpoints are set in % RTP logarithmic base 10 scale; therefore, the calculated values are applied to the trip setpoint and converted into percent of linear scale below for a numerically familiar verification. The conversion is made as per below equation:

Percent Linear Scale = [$\log_{10}$(% RTP – (-8)] / 10.3

25% RTP setpoint converted to percent span linear scale:

$Setpoint_{\_25\%\_\%\_Span}$= [$\log_{10}$ 25.00 - (-8)] / 10.3

$Setpoint_{\_25\%\_\%\_Span}$= 91.24%

The PMA in linear scale is calculated by adding and subtracting the PMA in % RTP from the 25% RTP setpoint; therefore:

$PMA_{\_Inter\_Range}$ = ± 5.558% RTP random / - 8.240% RTP bias

Setpoint plus + PMA Random:

+ $Setpoint_{\_25\%\_\%\_PMA\_Random}$ = 25% RTP + 5.558% = 30.558% RTP

In percent linear scale:

+ $Setpoint_{\_25\%\_\%\_PMA\_Random\_Span}$ = [$\log_{10}$ 30.558 - (-8)] / 10.3

+ $Setpoint_{\_25\%\_\%\_PMA\_Random\_Span}$ = 92.09%

Setpoint minus - PMA Random:

- $Setpoint_{\_25\%\_\%\_Span\_Random}$ = 25% RTP - 5.558% = 19.442%

In percent linear scale:

- $Setpoint_{\_25\%\_\%\_PMA\_Random\_Span}$ = [$\log_{10}$ 19.442 - (-8)] / 10.3

- $Setpoint_{\_25\%\_\%\_PMA\_Random\_Span}$ = 90.18%

And the random PMA effect on the setpoint value in linear scale is:

PMA/R^+ = 92.09% - 91.24% = + 0.85%

PMA/R^- = 90.18% - 91.24% = -1.06%

Setpoint plus - PMA Bias:

$Setpoint_{\_25\%\_\%\_PMA\_Bias}$ = 25% RTP - 8.240% = 16.760% RTP

In percent linear scale:

$Setpoint_{\_25\%\_\%\_PMA\_Random\_Bias}$ = [$\log_{10}$ 16.760 - (-8)] / 10.3

+ $Setpoint_{\_25\%\_\%\_PMA\_Random\_Span}$ = 89.56%

And the bias PMA effect on the setpoint value in linear scale is:

PMA/B^- = 89.56% - 91.24% = -1.68%

3.4 Total Loop Uncertainty ($TLU_{\_Inter\_Range}$)

The Rack Total Loop Uncertainty (TLU) is determined combining statistically the rack random uncertainty with the process random measurement effects, and resultant random effects combination is algebraically combined with the process bias effects. The combination of the rack uncertainties with the PMAs is performed in percent of linear scale, therefore, the rack uncertainty effect in the setpoint is first converted to percent of linear scale.

<u>Setpoint plus + Rack Uncertainty Random:</u>

Setpoint in linear scale from section 3.3 is:

$Setpoint_{\_25\%\_\%\_Span} = 91.24\%$

PMAs effects in linear scale from section 3.3 are:

$PMA/R^{+} = + 0.85\%$

$PMA/R^{-} = -1.06\%$

$PMA/B^{-} = -1.68\%$

Total rack uncertainty from section 3.2 is:

$Rack_{\_Inter\_Range} = \pm 1.82\%$ span

The total loop uncertainty is:

$+ TLU_{\_Inter\_Range} = + (Rack_{\_Inter\_Range}{}^{2} + PMA/R^{+2})^{1/2}$

$+ TLU_{\_Inter\_Range} = + (1.82^{2} + 0.85^{2})^{1/2}$

$+ TLU_{\_Inter\_Range} = + 2.01\%$ span

$- TLU_{\_Inter\_Range} = - [Rack_{\_Inter\_Range}{}^{2} + PMA/R^{-2}]^{1/2} + PMA/B^{-}$

$- TLU_{\_Inter\_Range} = - [1.82^{2} + 0.85^{2}]^{1/2} - 1.68$

$- TLU_{\_Inter\_Range} = - 3.69\%$ span

3.5 As Found Tolerance

The As Found Tolerance, also termed Allowable Value, is determined on instrumentation based performance, TSTF-493, which is defined by the combination of the Rack Uncertainty Terms comprising Accuracy, MTE, and Drift terms.

$$AFT\_Rack_{\_Inter\_Range} = \pm (RCA_{\_Inter\_Range}^{2} + RMTE_{\_Inter\_Range}^{2} + RCSA_{\_Inter\_Range}^{2} + +RD_{\_Inter\_Range}^{2})^{1/2}$$

$$AFT\_Rack_{\_Inter\_Range} = \pm (1.50^2 + 0.05^2 + 0.25^2 + 1.00^2)^{1/2}$$

$$AFT\_Rack_{\_Inter\_Range} = \pm 1.82\%\ span$$

3.6 As Left Tolerance

The As Left Tolerance, is determined on instrumentation based performance, TSTF-493, and is defined by the combination of the Accuracy and MTE terms.

$$AL\_Rack_{\_Inter\_Range} = \pm (RCA_{\_Inter\_Range}^{2} + RMTE_{\_Inter\_Range}^{2} + RCSA_{\_Inter\_Range}^{2})^{1/2}$$

$$AL\_Rack_{\_Inter\_Range} = \pm (1.50^2 + 0.05^2 + 0.25^2)^{1/2}$$

$$AL\_Rack_{\_Inter\_Range} = \pm 1.52\%\ span$$

3.7 Setpoint Analysis

In this section a review of the replacement system accuracies in percent RTP logarithmic, is performed per below equation.

$$Setpoint\ (\%\ RTP) = Process\ Value\ (\%) \times 10^{(CU\ \% / 100\% \times b)}$$

where:

a = base decade operation (-8)
b= number of decades of operation (10.3)
CU = channel uncertainty (in % span)

The Technical Specifications setpoints are set in % RTP logarithmic base 10 scale; therefore, the calculated values are applied to the trip setpoint and converted into percent of linear scale below for a numerically familiar verification. The conversion is made as per below equation:

$$Percent\ Linear\ Scale = 100 \times \log_{10}[\ \%\ RTP - (-8)] / 10.3$$

3.7.1 Instrumentation Setpoints

3.7.1.1 Trip Setpoint

a. As Found Tolerance / TS Allowable Value

The determination of the As Found Tolerance, or Allowable Value, for the Technical Specifications setpoint by applying the $AFT\_Rack_{\_Inter\_Range}$ uncertainty is:

$+ AFT = 25\% \times 10^{[1.82\% / 100\%) \times 10.3]}$

+ AFT = 38.5% RTP

$- AFT = 25\% \times 10^{[-1.82\% / 100\%) \times 10.3]}$

- AFT = 16.2% RTP

Where the current TS Allowable Value is 30% percent of logarithmic scale. The calculated value is larger than the TS value by a difference of 8.5%.

The positive AFT in percent linear scale:

$+ AFT = [\log_{10}38.5 - (-8)] / 10.3$

+ AFT = 93.063%

There is margin of 106.937% (200% - 93.063%) to the upper range of the scale of 200%; therefore, the calculated value is found acceptable. However, analysis of the setpoint with channel total uncertainty included in respect to the process Analytical Limit is made in section 3.7.1.2.

b. As Left Tolerance

The determination of the As Left Tolerance is made by applying the $AL\_Rack_{\_Inter\_Range}$ uncertainty:

$+ ALT = 25\% \times 10^{[(1.52\% / 100\%) \times 10.3]}$

+ ALT = 35.9% RTP

$- ALT = 25\% \times 10^{[(-1.52\% / 100\%) \times 10.3]}$

- ALT = 17.4% RTP

The positive ALT in percent linear scale:

$+ ALT = [\log_{10}35.9 - (-8)] / 10.3$

+ ALT = 92.8%

3.7.1.2 Trip Setpoint – Analytical Limit

There is not an Analytical Limit; therefore, an Analytical Limit is set at 75% RTP and verified that the setpoint falls within an established range. Below is analyzed the trip setpoint with the new rack uncertainties to the selected Analytical Limit.

The Analytical Limit in percent linear scale:

$AL_{\_Inter\_Range} = [\log_{10}75 - (-8)] / 10.3$

$AL_{\_Inter\_Range}$ = 95.9%

And the Margin from the Trip Setpoint to the Analytical Limit is:

Margin = AL - [Trip Setpoint + (-TLU)]

Where the setpoint and TLU from sections 3.3 and 3.4 are:

$Setpoint_{\_25\%\_\%\_Span}$= 91.24%

- $TLU_{\_Inter\_Range}$ = - 3.69% span

Margin = A – (Setpoint + TLU)

Margin = 95.9% - 91.24% - 3.69%

Margin = + 0.97%

There is a positive margin from the Analytical Limit in the linear scale from the 75% RTP value; therefore, the setpoint is found acceptable.

Conclusions:

Current TS allowable value should be replaced by the calculated AFT of ≤ 38.5% RTP.

3.7.1.3 P-6 Interlock

The P-6 Permissive enables the Source Range Reactor trip setpoint to be blocked on increasing power and disables this block on decreasing reactor power. Permissive setpoint is 10^{-10} amperes (approximately 10^{-5} % RTP) and the reset is $7x10^{-11}$ amperes (approximately $7.3x10^{-6}$ % RTP) (Reference 4.5). The reset has its allowable value in the TS of <6 x 10^{-11} amperes; therefore, the setpoint of interest is the reset with current allowable value in the Technical Specifications. The specified values are nominal settings which do not have Analytical or Process Limits. Per observation of the systems scales correlation, Figure 1, the setpoint value allows extends 2 decades into the Source Range system scale. The setpoint is a permissive setpoint and is not used in any safety analyses; therefore, there is not an analysis made against any Analytical Limit.

a. <u>As Found Tolerance / TS Allowable Value</u>

The determination of the As Found Tolerance, or Allowable Value, for the P-6 Reset setpoint by applying the $AFT\_Rack_{\_Inter\_Range}$ uncertainty is:

$+ AFT = 7.3 \times 10^{-6} \% \times 10^{[(1.82\% / 100\%) \times 10.3]}$

$+ AFT = 1.1 \times 10^{-5}$ % RTP

$- AFT = 7.3 \times 10^{-6} \% \times 10^{[(-1.82\% / 100\%) \times 10.3]}$

$- AFT = 4.7 \times 10^{-6}$ % RTP

The positive AFT in percent linear scale / volts:

$$+ AFT = [\log_{10} 1.1 \times 10^{-5} - (-8)] / 10.3$$

$$+ AFT = 29.5\% \text{ or } 3.041 \text{ volts (note)}$$

The P-6 reset + AFT in percent linear scale / volts:

$$+ AFT = [\log_{10} 7.3 \times 10^{-6} - (-8)] / 10.3$$

$$+ AFT = 27.8\% \text{ or } 2.863 \text{ volts (note)}$$

Note: Per figure 1, the 0%-100% linear scale is equivalent to 0-10.3 volts.

b. As Left Tolerance

The determination of the As Left Tolerance is made by applying the $AL\_Rack_{\_Source}$ uncertainty:

$$+ ALT = 7.3 \times 10^{-6} \% \times 10^{[(1.52\% / 100\%) \times 10.3]}$$

$$+ ALT = 1.05 \times 10^{-5} \% \text{ RTP}$$

$$- ALT = 7.3 \times 10^{-6} \% \times 10^{[(-1.52\% / 100\%) \times 10.3]}$$

$$- ALT = 5.09 \times 10^{-6} \% \text{ RTP}$$

The positive ALT in percent linear scale:

$$+ ALT = = [\log_{10} 1.05 \times 10^{-5} - (-8)] / 10.3$$

$$+ ALT = = 29.3\%$$

c. Analytical Limit – Setpoints Separation

There is not a defined Analytical Limit (AL) for the P-6 Permissive reset setpoint. However, an adequate separation between the Source Range Setpoint decreasing value and the P-6 Reset increasing value exists as presented in Figure 1.

d. Conclusions

The current allowable value of < 6 x10^{-11} amps should be updated to the new rack performance by the calculated AFT of < 4.7 x 10^{-6} RTP.

Figure 1

NUCLEAR INSTRUMENT RANGES OF THE EX-CORE NEUTRON FLUX MONITORING SYSTEMS

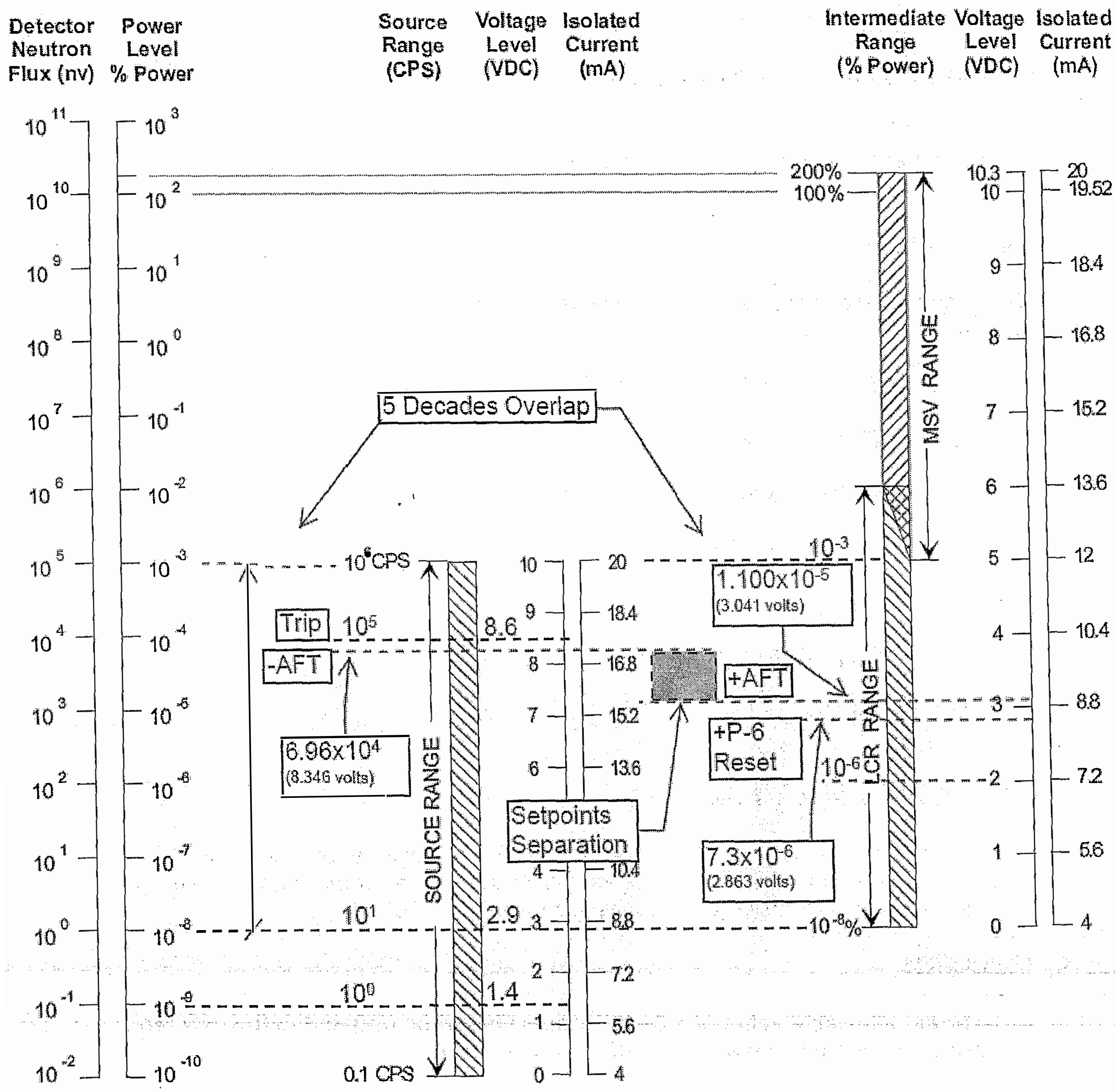
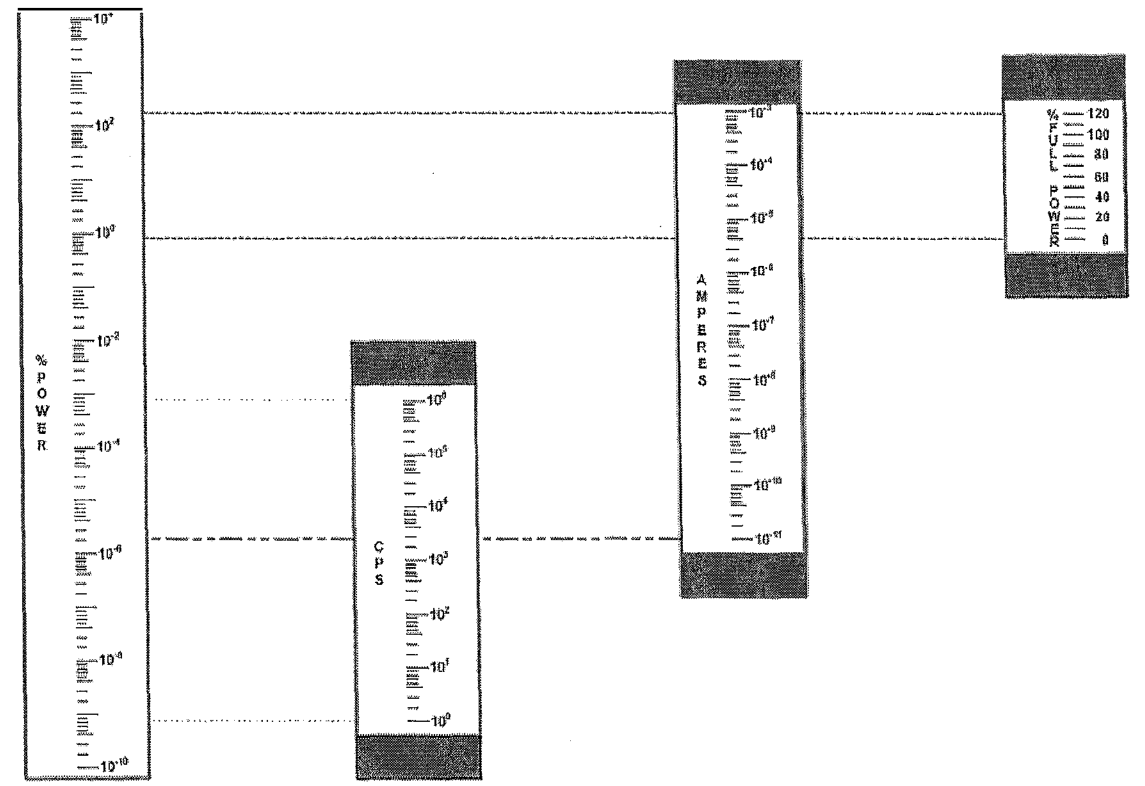


Figure 2

Current Systems Scaling Relationship



4.0 References:

4.1 Purchasing Specification S-C-DE-NIS-0210, Excore Nuclear Instrumentation System (NIS)
4.2 Calculation SC-NIS001-01, Rev. 3, Salem Unit 1/2 Reactor Flux Source Range Instrumentation
4.3 Thermo Fisher instrument accuracies, Attachment 1
4.5 SC-NIS002-01, Rev. 1, Salem Unit 1/2 Reactor Flux Intermediate Range Instrumentation, NI35B, N136B
4.6 TSTF-493, Rev. 4, Technical Specifications Task Force
4.7 Nuclear Instrument Ranges of the ex-core Neutron Flux Monitoring Systems, Thermo Fisher – Figure 1
4.8 Current Systems Scaling Relationship, PSEG's Excore Nuclear Instrumentation System, ILT-11-01, Paul Abbott July 11th and 12th 2011., Figure 2

Attachment 1

ThermoFisher
SCIENTIFIC

The world leader
in serving science

January 15, 2015

PSEG Nuclear LLC
Salem Site
Attn: Saeed Savar
Alloway Creek Neck Road
Hanocks Bridge NJ 08038 USA

Subject: Setpoint Methodology

Attachments: Table 1 - Source Range Values
Table 2 - Intermediate Range Values

Saeed,

Attached please find two tables that provide information that can be used to upgrade your Westinghouse Setpoint Methodology Calculations to reflect the more modem electronics that will be included in your new Thermo Fisher Scientific Source and Intermediate Range Neutron Flux Monitoring Systems.

Please feel free to call me if there are any questions.

Levi Parsell
Electrical Engineer
(858) 882-1293 phone
Levi.parsell@thermo.com

Process Instruments [illegible] Mesa Rim Road San Diego, CA 92121 [illegible] [illegible] fax [illegible]

TABLE 1

SOURCE RANGE, NEUTRON FLUX

PARAMETER		ALLOWANCES*
PROCESS MEASUREMENT ACCURACY	(PMA)	**
PRIMARY ELEMENT ACCURACY (Included in RCA)	(PEA)	0% OF SPAN
SENSOR CALIBRATION (Included in RCA)	(SCA)	0% OF SPAN
SENSOR MEASUREMENT & TEST	(SMTE)	0% OF SPAN
SENSOR TEMPERATURE EFFECTS	(STE)	0% OF SPAN
SENSOR PRESSURE EFFECTS (Included in RCA)	(SPE)	0% OF SPAN
SENSOR DRIFT	(SD)	0% OF SPAN
ENVIRONMENTAL ALLOWANCES	(EA)	0% OF SPAN
RACK CALIBRATION		
RACK ACCURACY	(RCA)	+/- 2.0% OF SPAN
MEASUREMENT & TEST	(RMTE)	+/- 0.05% OF SPAN
COMPARATOR	(RCSA)	+/- 0.25% OF SPAN
RACK TEMPERATURE EFFECTS (RCA includes temperature effects)	(RTE)	0% OF SPAN
RACK DRIFT (24 MONTHS)	(RD)	+/-1.0% of SPAN

* In percent of span (0.1 to 10E6 CPS)
** Salem determined value

TABLE 2

INTERMEDIATE RANGE, NEUTRON FLUX

PARAMETER		ALLOWANCES*
PROCESS MEASUREMENT ACCURACY	(PMA)	**
PRIMARY ELEMENT ACCURACY (Included in RCA)	(PEA)	0% OF SPAN
SENSOR CALIBRATION (Included in RCA)	(SCA)	0% OF SPAN
SENSOR MEASUREMENT & TEST SENSOR TEMPERATURE EFFECTS SENSOR PRESSURE EFFECTS (Included in RCA)	(SMTE) (STE) (SPE)	0% OF SPAN 0% OF SPAN 0% OF SPAN
SENSOR DRIFT	(SD)	0% OF SPAN
ENVIRONMENTAL ALLOWANCES	(EA)	0% OF SPAN
RACK CALIBRATION RACK ACCURACY (RCA includes temperature effects) MEASUREMENT & TEST	 (RCA) (RMTE)	 +/- 1.5% OF SPAN +/- 0.05% OF SPAN
COMPARATOR	(RCSA)	+/- 0.25% OF SPAN
RACK TEMPERATURE EFFECTS (RCA includes temperature effects)	(RTE)	0% OF SPAN
RACK DRIFT (24 MONTHS)	(RD)	+/-1.0% of SPAN

* All parameters are shown in percent of span (logarithmic scale 1 E-8% to 200%)
** Salem determined value.