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10 CFR 50.90

April 19, 2016

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

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Response to Draft Request for Additional Information Regarding Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

References:

1. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request – Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03," dated January 15, 2016 (ADAMS Accession No. ML16015A316).

2. Electronic mail message from Richard Ennis, U.S. Nuclear Regulatory Commission, to Glenn Stewart, Exelon Generation Company, LLC, "Draft RAIs - Limerick LAR to Reduce Steam Dome Pressure (CACs MF7263 & 64)," dated March 10, 2016 (ADAMS Accession No. ML16085A025).

By letter dated January 15, 2016 (Reference 1), Exelon Generation Company, LLC (Exelon) submitted a license amendment request (LAR) for Limerick Generating Station (LGS), Units 1 and 2. The proposed amendment would reduce the reactor vessel steam dome pressure associated with the Technical Specifications (TS) Safety Limits (SLs) specified in TS 2.1.1 and TS 2.1.2. The amendment would also revise the setpoint and allowable value for the main steam line low pressure isolation function in TS Table 3.3.2-2. The proposed changes address a 10 CFR Part 21 issue concerning the potential to violate the SLs during a pressure regulator failure maximum demand (open) (PRFO) transient.

The NRC staff reviewed the information provided that supports the proposed amendment and identified the need for additional information in order to complete their evaluation of the amendment request. The draft request for additional information (RAI) was sent from the NRC to Exelon by electronic mail message on March 10, 2016 (Reference 2). A conference call was conducted on March 24, 2016, to provide clarification of the questions. Subsequent to the teleconference, it was agreed that Exelon would respond to the RAI by April 25, 2016, which was acceptable to the NRC.

Attachment 1 to this letter provides a restatement of the RAI questions followed by our responses. Attachment 2 provides revised markups for TS page 2-1. The proposed changes to Table 3.3.2-2 on TS page 3/4 3-18 remain unchanged by this response but are

included in Attachment 2 for completeness. Attachment 3 provides revised TS Bases markups (for information only). Attachment 4 provides a copy of Loop Uncertainty Calculation LI-00032, "LU Calculation for PT-001-2N076C."

Exelon has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of the Reference 1 letter. Exelon has concluded that the information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, Exelon has concluded that the information in this response 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 in this response.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the Commonwealth of Pennsylvania of this RAI response by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 19th day of April 2016.

Respectfully,

D. P. Helker

David P. Helker
Manager, Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1. Response to Draft Request for Additional Information Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03
2. Revised Markup of Proposed Technical Specifications Pages
3. Revised Markup of Proposed Technical Specifications Bases Pages (Information Only)
4. Loop Uncertainty Calculation LI-00032, "LU Calculation for PT-001-2N076C"

cc:	Regional Administrator - NRC Region I	w/ attachment
	NRC Senior Resident Inspector - Limerick Generating Station	"
	NRC Project Manager, NRR - Limerick Generating Station	"
	Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Protection	"

ATTACHMENT 1

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Response to Draft Request for Additional Information
Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

By letter dated January 15, 2016 (Reference 1), Exelon Generation Company, LLC (Exelon) submitted a license amendment request (LAR) for Limerick Generating Station (LGS), Units 1 and 2. The proposed amendment would reduce the reactor vessel steam dome pressure associated with the Technical Specifications (TS) Safety Limits (SLs) specified in TS 2.1.1 and TS 2.1.2. The amendment would also revise the setpoint and allowable value for the main steam line low pressure isolation function in TS Table 3.3.2-2. The proposed changes address a 10 CFR Part 21 issue concerning the potential to violate the SLs during a pressure regulator failure maximum demand (open) (PRFO) transient.

The NRC staff reviewed the information provided that supports the proposed amendment and identified the need for additional information in order to complete their evaluation of the amendment request. Below is a restatement of the questions followed by our responses.

SRXB-RAI-1

The current LGS TS 2.1.2 requires that the minimum critical power ratio (MCPR) be ≥ 1.09 for two recirculation loop operation and ≥ 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

An LAR dated November 19, 2015 (ADAMS Accession No. ML15323A257), for LGS Unit 1, was submitted to the NRC regarding TS 2.1, “Safety Limits,” to revise Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the cycle specific analysis performed by Global Nuclear Fuel for the upcoming Cycle 17. The proposed changes to the SLMCPR values are from ≥ 1.09 to ≥ 1.10 for two loop operation and from ≥ 1.12 to ≥ 1.14 for single loop operation. The NRC staff requests that the licensee clarify whether the proposed steam dome pressure change considered the SLMCPR change for TS 2.1.2 in the referenced LGS Unit 1 LAR.

Response

LGS Unit 1 transitioned to a full core of GNF2 fuel during the 1R16 refueling outage which was completed on April 17, 2016. The lower bound limit of 700 psia for the GEXL17 critical power correlation is justified for GNF2 fuel as indicated in the LAR for the proposed reactor vessel steam dome pressure change (Reference 1). The same correlation is used for the LGS Unit 1 TS SLMCPR change consistent with its range of applicability, which includes the lower bound limit of 700 psia. The LAR for the proposed reactor vessel steam dome pressure change, to extend the low pressure applicability, does not affect the LAR for the LGS Unit 1 TS SLMCPR proposed change. The noted LARs remain independent when the GEXL17 correlation is used within its application range.

The LGS Unit 1 SLMCPR Amendment No. 221 was issued by letter dated March 15, 2016 (Reference 2) and has been implemented by LGS. Therefore, the revised markup for TS page 2-1 for LGS Unit 1 included in Attachment 2 is based on the current LGS Unit 1 TS which incorporates the changes to the SLMCPR that were approved in Amendment No. 221.

SRXB-RAI-2

The LAR states that main steam isolation valve (MSIV) low pressure isolation setpoint (LPIS) setting, calculated at 840 pounds per square inch gauge (psig), is based on the new analytical limit of 805 psig. The NRC staff requests that the licensee (1) provide a description of how the new analytical limit of 805 psig was arrived at, and (2) how the proposed MSIV LPIS setting of 840 psig is based on this new analytical limit.

Response

1. The current low LPIS setting (720 psig analytical limit) is not sufficient to preclude steam dome pressure from falling below 685 psig (700 psia) when above 25% power for current operation during a PRFO Anticipated Operational Occurrence (AOO) event. The approach discussed in Section 5 of the Boiling Water Reactor Owners Group (BWROG) report, NEDC-33743, Rev. 0 (Reference 3), was followed for application of the BWROG method to LGS. The results from Section 4 of the BWROG report, which are most applicable to the LGS configuration, were used. A change of analytical limit by scaling up the results in the BWROG report Table 5 for an increased LPIS analytical limit from 720 psig to 805 psig is required to meet the acceptance criterion. Accordingly, the approach considers the most limiting plant configuration and operating conditions for evaluating the effect of the SC05-03 issue.

2. The increased LPIS analytical limit of 805 psig was used as input to revise the loop uncertainty calculation for LGS. Based on this new analytical limit, the associated changes to allowable value and actual trip setpoint were established as part of the loop uncertainty calculation update (see Attachment 4).

SRXB-RAI-3

The NRC staff requests that the licensee discuss the impact of this Main Steam Line Pressure – Low allowable value change, primarily focusing on the PRFO transient.

Response

PRFO Anticipated Transient Without Scram (ATWS) - An increased LPIS analytical limit requires an increase in the allowable value from 736 psig to 821 psig. The allowable value of 736 psig is used in the current LGS analysis of ATWS (Reference 4). The LGS ATWS analysis considers failure of the pressure regulator to maximum demand (PRFO) as a limiting event. The event causes a drop in reactor vessel pressure and water level which continues until MSIV isolation is initiated on steam line low pressure isolation setpoint (LPIS). As the increased LPIS analytical limit will result in an increased allowable value that results in earlier steam line isolation and recirculation pump trip action, the LGS ATWS analysis remains applicable with respect to this change.

PRFO AOO - The analysis uses the LPIS analytical limit to initiate steam line isolation. The revised analytical limit of 805 psig, which is higher than the current value of 720 psig, will result in earlier steam line isolation to terminate depressurization. The change in LPIS analytical limit

will not affect the significance of the PRFO event as a non-limiting event with respect to fuel thermal limit.

SRXB-RAI-4

The licensee proposes to reduce the reactor steam dome pressure specified in TS 2.1.1 and TS 2.1.2 from 785 psig to 685 psig based on the lower-bound pressure for the critical power correlation for the fuel currently used in the LGS, Units 1 and 2 cores. The licensee's application references Global Nuclear Fuel (GNF) reports NEDC-33270P, NEDC-33292P and NEDC-32851P-A as the basis supporting the proposed change. The LGS Unit 1 core currently consists of GE14 and GNF2 fuel types and LGS Unit 2 uses GNF2 fuel.

Section 3.8.3 of GNF report NEDC-33270P discusses the critical power correlation for GNF2 fuel (i.e., GEXL17 correlation). This section includes the pressure range over which the GEXL17 correlation is valid for GNF2 fuel consistent with the information provided in Table 5-4 of GNF2 report NEDC-33292P. As discussed in Section 3.0 of Attachment 1 of the licensee's application, the lower bound pressure limit for the GEXL17 correlation is 700 pounds per square inch atmospheric (psia).

GNF report NEDC-32851P-A discusses the critical power correlation for GE14 fuel (i.e., GEXL14 correlation). Similar to the GEXL17 correlation, Section 5.2 of the report states that the lower bound pressure limit for the GEXL14 correlation is 700 psia.

Converting 700 psia to psig, the lower bound pressure for the GEXL17 and GEXL14 correlations is approximately 685.3 psig. As such, the 685 psig value specified in the proposed TS change is slightly outside the pressure range in which the GEXL17 and GEXL14 correlations are valid for GNF2 and GE14 fuel. Please provide further justification for the proposed 685 psig value or propose a revised pressure value for this TS change that is supported by the GEXL17 and GEXL14 correlations (e.g., 700 psia).

Response

Exelon has decided to reference the lower bound limit for the critical power correlation in absolute pressure (i.e., 700 psia) for the GNF2 fuel currently used in the LGS, Unit 1 and Unit 2 cores, as referenced by GNF reports, NEDC-33270P and NEDC-33292P. Exelon proposes to revise the lower bound reactor steam dome pressure for the reactor core safety limits specified in TS 2.1.1 and TS 2.1.2 to reference the absolute pressure value of 700 psia. Note: The Unit 2 core already uses all GNF2 fuel. In addition, Unit 1 transitioned to all GNF2 fuel during the Unit 1 refueling outage which was completed on April 17, 2016.

Attachment 2 provides a copy of the revised TS mark-up pages that reflect the proposed change. Attachment 3 provides the corresponding revised TS Bases mark-up pages (for information only).

EICB-RAI-1

The proposed amendment request entails changes to TS Table 3.3.2-2 and revises the trip setpoint and the allowable value for the main steam line low pressure isolation function. In order for the NRC staff to verify compliance to the regulations and the guidance pertaining to setpoint changes, the staff requests the licensee to submit the calculation for staff review. The calculation will be used to assess the methodology, the changes in assumptions, calculation of total loop uncertainty, and other pertinent information in the calculation.

Response

Attachment 4 provides a copy of Loop Uncertainty Calculation LI-00032, "LU Calculation for PT-001-2N076C" for NRC review. As discussed in a recent LGS amendment (ADAMS Accession No. ML14324A808), the LGS setpoint methodology, which is currently contained in Exelon Procedure CC-MA-103-2001, is based on the NRC-approved GE Topical Report NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996. The NRC staff previously found the LGS setpoint methodology acceptable as discussed in an NRC letter dated February 16, 1995, "Revised Maximum Authorized Thermal Power Limit, Limerick Generating Station, Unit No. 2 (TAC No. M88393)" (ADAMS Accession No. ML011560773).

References:

1. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03," dated January 15, 2016 (ADAMS Accession No. ML16015A316).

2. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to Bryan C. Hanson, Exelon Nuclear, "Limerick Generating Station, Unit 1 - Issuance of Amendment, RE: Safety Limit Minimum Critical Power Ratio Change (CAC No. MF7101), dated March 15, 2016 (ADAMS Accession No. ML16041A021).

3. NEDC-33743P, Revision 0, "BWR Owners' Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," dated April 2012.

4. 0000-0097-1195-R0, Exelon Nuclear Limerick Units 1 and 2 Thermal Power Optimization, Task 902: Anticipated Transients Without Scram, December 2009.

ATTACHMENT 2

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Response to Draft Request for Additional Information
Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

Revised Markup of Proposed Technical Specifications Pages

Unit 1 TS Pages
2-1
3/4 3-18

Unit 2 TS Pages
2-1
3/4 3-18

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than ~~785 psig~~ 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than ~~785 psig~~ 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.10 for two recirculation loop operation and shall not be less than 1.14 for single recirculation loop operation with the reactor vessel steam dome pressure greater than ~~785 psig~~ 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.10 for two recirculation loop operation or less than 1.14 for single recirculation loop operation and the reactor vessel steam dome pressure greater than ~~785 psig~~ 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ ~~756~~ 840 psig	≥ ~~736~~ 821 psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than ~~785 psig~~ 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than ~~785 psig~~ 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 for two recirculation loop operation and shall not be less than 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than ~~785 psig~~ 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 for two recirculation loop operation or less than 1.12 for single recirculation loop operation and the reactor vessel steam dome pressure greater than ~~785 psig~~ 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATION CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
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2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ ~~756~~ 840 psig	≥ ~~736~~ 821 psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

ATTACHMENT 3

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Response to Draft Request for Additional Information
Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

Revised Markup of Proposed Technical Specifications Bases Pages (Information Only)

Unit 1 TS Bases Page
B 2-1

Unit 2 TS Bases Page
B 2-1

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that more than 99.9% of the fuel rods avoid transition boiling. Meeting the Safety Limit can be demonstrated by analysis that confirms less than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the values specified in Specification 2.1.2 for two recirculation loop operation and for single recirculation loop operation. Less than 0.1% of fuel rods in transition boiling and MCPR greater than the values specified for two recirculation loop operation and for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

700 psia

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below ~~785 psig~~ or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/h, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below ~~785 psig~~ is conservative.

700 psia

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principle barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that more than 99.9% of the fuel rods avoid transition boiling. Meeting the Safety Limit can be demonstrated by analysis that confirms less than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the values specified in Specification 2.1.2 for two recirculation loop operation and for single recirculation loop operation. Less than 0.1% of fuel rods in transition boiling and MCPR greater than the values specified for two recirculation loop operation and for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

700 psia

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below ~~785 psig~~ or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below ~~785 psig~~ is conservative.

700 psia

ATTACHMENT 4

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Response to Draft Request for Additional Information
Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

Loop Uncertainty Calculation LI-00032, "LU Calculation for PT-001-2N076C"

ATTACHMENT 1
Design Analysis Cover Sheet

Page 1 Followed by 1A

Design Analysis		**Last Page No.**[6] Page 21	
Analysis No.:[1]	LI-00032	**Revision:**[2] 0A Major ☐	Minor ☒
Title:[3]	LU Calculation for PT-001-2N076C		
EC/ECR No.:[4]	LG 15-00017	**Revision:**[5] 0	

		Component(s):[14]	
Station(s):[7]	Limerick		
Unit No.:[8]	1 & 2	PT-001-1N076A	PT-001-2N076A
Discipline:[9]	Electrical	PT-001-1N076B	PT-001-2N076B
Descrip. Code/Keyword:[10]	N/A	PT-001-1N076C	PT-001-2N076C
Safety/QA Class:[11]	Safety Related	PT-001-1N076D	PT-001-2N076D
System Code:[12]	001		
Structure:[13]			

CONTROLLED DOCUMENT REFERENCES[15]

Document No.:	From/To	Document No.:	From/To
UFSAR Section 7.3.1.1.2.4.5	To		
Tech Specs Table 3.3.2-2 Item 1.c	To		
DBD L-S-16	To		

Is this Design Analysis Safeguards Information?[16] Yes ☐ No ☒ If yes, see SY-AA-101-106

Does this Design Analysis contain Unverified Assumptions?[17] Yes ☐ No ☒ If yes, ATI/AR#:

This Design Analysis SUPERCEDES:[18] **N/A** **In its entirety.**

Description of Revision (list changed pages when all pages of original analysis were not changed):[19]
This revision changes the Analytical Limit from 720 to 805 psig based on new references 4.17 and 4.18. The new Analytical Limit results in changes to the Allowable Value, Nominal Trip Setpoint, Actual Trip Setpoint, and Acceptable Limits as determined in section 7.7, 7.5, 7.6, and 7.8 respectively. Changed pages include:

2 - Updated section 1.3 to reflect new allowable value, setpoint, and TODI as basis.
4 - Updated section 2.2.1 to reflect new analytical limit value and TODI as basis.
5 - Updated section 2.2.5 and 2.2.6 to reflect new pressure margin values, and updated section 3.1.1 to delete the assumption that is no longer valid.
6 - Updated document revision numbers in sections 4.1 thru 4.5, and deleted references 4.6 and 4.7.
7 - Updated section 4.9 with new revision number, deleted historical references from sections 4.10 and 4.11, and added new references 4.17 and 4.18.
16 - Updated Limit and NTSP values in section 7.5.
17 - Updated margin and pressure values in sections 7.6, 7.7, and 7.8.
19 - Marked Attachment 2 (calculation results) for replacement by new IISCP calculation results.
21 - Updated Setpt, Allw, and Analytical/Proc Lmt values in Attachment 4.

The use of a minor revision was approved by the Ray George (LEDE branch manager) on behalf of the SMDE on 9/28/15.

Preparer: [20]	Don Hoolahan Print Name	Sign Name	9/28/2015 Date
Method of Review: [21]	**Detailed Review** ☒	**Alternate Calculations (attached)** ☐	**Testing** ☐
Reviewer: [22]	Mike Lazar Print Name	B.F. Wells for MIKE LAZAR Sign Name	9/30/15 Date
Review Notes: [23]	Independent review ☒	Peer review ☐	

Calculation has been independently reviewed per CC-AA-309 and CC-AA-309-1001. All comments have been satisfactorily incorporated.

(For External Analyses Only)

External Approver: [24]	John Pelliccone Print Name	Sign Name	9/30/15 Date
Exelon Reviewer: [25]	ANTHONY ADDEO Print Name	Sign Name	10/1/15 Date
Independent 3rd Party Review Reqd? [26]	**Yes** ☐	**No** ☒	
Exelon Approver: [27]	Raymond T. George Print Name	Sign Name	10/1/15 Date

ATTACHMENT 2
Owner's Acceptance Review Checklist for External Design Analyses
Page 1B Followed by 1C

Design Analysis No.: L1-00032 **Rev:** 0A

No	Question	Instructions and Guidance	Yes / No / N/A
1.	Do assumptions have sufficient documented rationale?	All Assumptions should be stated in clear terms with enough justification to confirm that the assumption is conservative. For example, 1) the exact value of a particular parameter may not be known or that parameter may be known to vary over the range of conditions covered by the Calculation. It is appropriate to represent or bound the parameter with an assumed value. 2) The predicted performance of a specific piece of equipment in lieu of actual test data. It is appropriate to use the documented opinion/position of a recognized expert on that equipment to represent predicted equipment performance. Consideration should also be given as to any qualification testing that may be needed to validate the Assumptions. Ask yourself, would you provide more justification if you were performing this analysis? **If** yes, the rationale is likely incomplete.	☐ ☐ ☒
2	Are assumptions compatible with the way the plant is operated and with the licensing basis?	Ensure the documentation for source and rationale for the assumption supports the way the plant is currently or will be operated post change and they are not in conflict with any design parameters. If the Analysis purpose is to establish a new licensing basis, this question can be answered yes, if the assumption supports that new basis.	☐ ☐ ☒
3	Do all unverified assumptions have a tracking and closure mechanism in place?	**If** there are unverified assumptions without a tracking mechanism indicated, **then** create the tracking item either through an ATI or a work order attached to the implementing WO. Due dates for these actions need to support verification prior to the analysis becoming operational or the resultant plant change being op authorized.	☐ ☐ ☒
4	Do the design inputs have sufficient rationale?	The origin of the input, or the source should be identified and be readily retrievable within Exelon's documentation system. If not, then the source should be attached to the analysis. Ask yourself, would you provide more justification if you were performing this analysis? **If** yes, the rationale is likely incomplete.	☒ ☐ ☐
5	Are design inputs correct and reasonable with critical parameters identified, if appropriate?	The expectation is that an Exelon Engineer should be able to clearly understand which input parameters are critical to the outcome of the analysis. That is, what is the impact of a change in the parameter to the results of the analysis? If the impact is large, then that parameter is critical.	☒ ☐ ☐
6	Are design inputs compatible with the way the plant is operated and with the licensing basis?	Ensure the documentation for source and rationale for the inputs supports the way the plant is currently or will be operated post change and they are not in conflict with any design parameters.	☒ ☐ ☐

ATTACHMENT 2
Owner's Acceptance Review Checklist for External Design Analyses

Page 1C Followed by 1D

Design Analysis No.: L1-00032 **Rev:** 0A

No	Question	Instructions and Guidance	Yes / No / N/A
7	Are Engineering Judgments clearly documented and justified?	See Section 2.13 in CC-AA-309 for the attributes that are sufficient to justify Engineering Judgment. Ask yourself, would you provide more justification if you were performing this analysis? **If** yes, the rationale is likely incomplete.	☐ ☐ ☒
8	Are Engineering Judgments compatible with the way the plant is operated and with the licensing basis?	Ensure the justification for the engineering judgment supports the way the plant is currently or will be operated post change and is not in conflict with any design parameters. **If** the Analysis purpose is to establish a new licensing basis, **then** this question can be answered yes, if the judgment supports that new basis.	☐ ☐ ☒
9	Do the results and conclusions satisfy the purpose and objective of the Design Analysis?	Why was the analysis being performed? Does the stated purpose match the expectation from Exelon on the proposed application of the results? **If** yes, **then** the analysis meets the needs of the contract.	☒ ☐ ☐
10	Are the results and conclusions compatible with the way the plant is operated and with the licensing basis?	Make sure that the results support the UFSAR defined system design and operating conditions, or they support a proposed change to those conditions. If the analysis supports a change, are all of the other changing documents included on the cover sheet as impacted documents?	☒ ☐ ☐
11	Have any limitations on the use of the results been identified and transmitted to the appropriate organizations?	Does the analysis support a temporary condition or procedure change? Make sure that any other documents needing to be updated are included and clearly delineated in the design analysis. Make sure that the cover sheet includes the other documents where the results of this analysis provide the input.	☒ ☐ ☐
12	Have margin impacts been identified and documented appropriately for any negative impacts (Reference ER-AA-2007)?	Make sure that the impacts to margin are clearly shown within the body of the analysis. If the analysis results in reduced margins ensure that this has been appropriately dispositioned in the EC being used to issue the analysis.	☒ ☐ ☐
13	Does the Design Analysis include the applicable design basis documentation?	Are there sufficient documents included to support the sources of input, and other reference material that is not readily retrievable in Exelon controlled Documents?	☒ ☐ ☐
14	Have all affected design analyses been documented on the Affected Documents List (ADL) for the associated Configuration Change?	Determine if sufficient searches have been performed to identify any related analyses that need to be revised along with the base analysis. It may be necessary to perform some basic searches to validate this.	☒ ☐ ☐
15	Do the sources of inputs **and** analysis methodology used meet committed technical and regulatory requirements?	Compare any referenced codes and standards to the current design basis and ensure that any differences are reconciled. If the input sources **or** analysis methodology are based on an out-of-date methodology or code, additional reconciliation may be required if the site has since committed to a more recent code	☒ ☐ ☐

ATTACHMENT 2
Owner's Acceptance Review Checklist for External Design Analyses

Page 1D Followed by 2

Design Analysis No.: L1-00032 **Rev:** 0A

No	Question	Instructions and Guidance	Yes / No / N/A
16	Have vendor supporting technical documents and references (including GE DRFs) been reviewed when necessary?	Based on the risk assessment performed during the pre-job brief for the analysis (per HU-AA-1212), ensure that sufficient reviews of any supporting documents not provided with the final analysis are performed.	☒ ☐ ☐
17	Do operational limits support assumptions and inputs?	Ensure the Tech Specs, Operating Procedures, etc. contain operational limits that support the analysis assumptions and inputs.	☒ ☐ ☐

Create an SFMS entry as required by CC-AA-4008. SFMS Number: #51493

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Update header with Orig. Rev. Apr. and dates.

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1.0 PURPOSE

This section includes the Objective, Limitations, Conclusions, and the Applicability Statement of this calculation.

1.1 Objective

The objective of this calculation is to determine the Nominal Trip Setpoint (NTSP), Actual Trip Setpoint (ATSP) and the Allowable Value (AV) for the Main Steam Line Low Pressure Isolation Actuation Instrumentation as described in the Limerick Unit 2 Technical Specifications Table 3.3.2-2, Item 1.c (Ref. 4.2). This calculation analyzes the PT-001-2N076C instrumentation loop. This calculation was performed utilizing normal environmental conditions (see Section 2.2.3).

The normal NTSP, ATSP and AV results of this calculation are documented in Section 7.

Results of this "base calculation" are also applicable to the loops listed in Section 1.4.

1.2 Limitations

The Max and Min Acceptable Limits calculated in Section 7.8 are not authorized for use in the PECo maintenance program by this revision of the calculation.

This calculation is run for a normal environment and does not account for any uncertainties associated with accident scenarios (see Section 2.2.3).

The appropriate use of this calculation to support design or Station activities, other than those specified in Section 1.1 of this calculation, is the responsibility of the user.

1.3 Conclusions

The Allowable Value of ~~736.00~~ 821.00 PSIG was calculated utilizing the IISCP Software and includes operational flexibility per Design Input 2.2 of this calculation. ~~The Allowable Value is the result displayed in Section 7.7 of this calculation.~~

The ~~Allowable Value and~~ Analytical Limit~~s~~ utilized in this calculation ~~are in accordance with the General Electric Design Specification Data Sheets (Ref. 4.11).~~ is based on TODI ES1400026 (Ref. 4.17) and General Electric Safety Concern SC05-03 (Ref. 4.18).

The results of this calculation support the ~~current~~ new station setpoint of ~~756.00~~ 840.00 PSIG.

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installations which result in a head correction of +5.6 PSIG for both units. This was documented by the issuance of IISCP Anomaly No. 114, "Head Corrections for PT-001-1(2)N076A/B/C/D" (Ref. 4.14) which describes the issuance of Action Requests A0851879 (Ref. 4.15) and A0852289 (Ref. 4.16) (Type CM-NCR) for correcting these discrepancies. These discrepancies have no affect on this calculation as the head correction pertains only to the scaling of the transmitter. The scaling of PT-001-2N076C was done using +5.6 PSIG in accordance with the field installations.

2.0 DESIGN BASIS

This section includes the Technical Background and Design Input information relevant to the calculation.

2.1 Technical Background

Low steam pressure at the turbine inlet while the reactor is operating could indicate a malfunction of the steam pressure controller in which the turbine control valves or turbine bypass valves become fully open and cause rapid depressurization of the reactor vessel. Instrumentation is installed to monitor the steam line pressure in order to mitigate the consequences of this type of occurrence. The signals generated by this monitoring instrumentation input into the NSSSS isolation logic which automatically closes the Main Steam Isolation Valves (MSIVs) whenever the Mode Switch is in RUN. The MSIV isolation signal generated by the PT-001-2N076C loop is analyzed by this calculation.

2.2 Design Input

2.2.1 An Analytical/Process Limit of ~~720~~ 805.00 PSIG has been utilized for this calculation based on ~~the Low Turbine Throttle Pressure Setpoint (MSIV Fast Closure) parameter specified in the Final OPL-3 (Ref. 4.10).~~ TODI ES1400026 (Ref. 4.17) and General Electric Safetv Concern SC05-03 (Ref. 4.18).

~~The 720 PSIG limit is also in accordance with the General Electric Design Specification Data Sheets (Ref. 4.11).~~

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2.2.2 This calculation includes any applicable System Rerate Design/Operating Conditions and Impacts as a result of the Power Rerate analyses per the guidelines contained in Specification NE-177 (Ref. 4.12).

2.2.3 This calculation was performed under normal

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environmental conditions based on the design information contained in Section 15.1.3 of the Limerick Generating Station Updated Final Safety Analysis Report (UFSAR) (Ref. 4.1). UFSAR Section 15.1.3 indicates that the design bases event for the isolation of the main steam line as a result of low steam line pressure is a failure of the main turbine pressure regulator. This failure will result in no release of steam to the Turbine Enclosure environment. Therefore PT-001-2N076C will not be subjected to any harsh environment effects when accomplishing its intended safety function.

2.2.4 Process consideration has been included to provide support for additional operational flexibility. This process consideration appears within the calculation as consideration S1. This consideration is based on engineering judgement and reflects an amount approximately twice the accuracy of the transmitter plus an additional amount which results in a conservatively rounded Allowable Value.

2.2.5 The delta between the Allowable Value (AV) and the Actual Trip Set Point (ATSP) within this calculation is ~~20~~ 19 PSIG which satisfies the IISCP Leave Alone Zone Requirement to provide at least one LAZ between AV and ATSP.

2.2.6 Additional margin of ~~18.345~~ 17.345 PSIG was added to this calculation to support the current station setpoint. Of this ~~18.345~~ 17.345 PSIG, 8.406 PSIG is "assigned margin" used to support the IISCP LAZ requirements as discussed in Section 2.2.5. The remaining ~~9.939~~ 8.939 PSIG is "unassigned margin" which is considered additional conservatism that may be utilized in future analyses.

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2.2.7 All other design inputs to this calculation are documented on the Supporting Data Sheet Attachments.

3.0 ASSUMPTIONS

~~3.1 Assumptions Not Requiring Confirmation~~

3.1.1 None. ~~The Analytical/Process Limit for the Main Steam Line Low Pressure Isolation for Limerick 2 is assumed to be equivalent to the value in the Limerick Unit 1 OPL-3 Form for Cycle 5 (which~~

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~~includes Power Rerate information) and is documented in Section 2.2.1 of this calculation. The basis of this assumption is IISCP Project Letter to File M-P-PE001-0152 (Ref. 4.13).~~

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3.2 Assumptions Requiring Confirmation

3.2.1 None

Current revision is 16 dated September 2012.

4.0 REFERENCES

0A

4.1 Limerick Generating Station Updated Final Safety Analysis Report (UFSAR), Revision ~~03~~ (dated ~~11/93~~)

-Section 7.3.1.1.2.4.5 "PCRVICS - Main Steam Line Low Pressure"

-Section 15.1.3 Pressure Regulator Failure - Open (Design Basis reference).

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4.2 Limerick Generating Station Technical Specifications, Unit 2, Amendment ~~31~~, Table 3.3.2-2 Item 1.c (dated 02/17/94) (Operations and Surveillance requirements reference).

0007

4.3 Limerick Generating Station Units 1&2 System Design Baseline Document (DBD) L-S-16, Revision ~~0000~~, Section 3.2.9, Reactor Instrumentation System (Design Basis reference).

4.4 IISCP-PP-93-001, Revision 1 - Program Plan for the Implementation of Phase I of the PECo Improved Instrument Setpoint Control Program (IISCP) (Setpoint Methodology reference).

0017

4.5 M-171, Revision ~~0012~~ Limerick Generating Station Units 1&2 Environmental Service Conditions Specification (Location Data reference).

4.6 ~~M-0688, Sheet 0001, Revision 0027 - Limerick Generating Station Instrument Setpoint Index (Instrument Setpoint reference).~~

Deleted

4.7 ~~M-1570, Sheet 0001, Revision 0000 - Component Record List Limerick Generating Station (Instrument Data reference).~~

Deleted

4.8 Master Calibration Sheets generated in accordance with PECo procedure IC-11-50014 for PT-001-2N076C dated 06/28/88, PIS-001-2N676C dated 01/16/87. Master Loop Sheet for PT-001-2N076C dated 06/28/88 (Applicability reference).

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4.9 Calculation M-75-12, Revision ~~0006~~ 0017 - "Turbine Building Cooling Load" (Location Data reference).

0A

4.10 Philadelphia Electric Company Letter from G.C. Storey to G.R. Hull General Electric Company, subject "Final OPL-3 for Limerick ARTS/MELLLA Analysis". This document contains Limerick 1 Reload 4(Cycle 5) Resolved OPL-3 Forms that include ARTS/MELLLA at Rerate Conditions Dated 03/09/93. ~~(Analytical/Process Limit reference)~~ (Rev. 0 historical reference)

4.11 B21-4020-L-001, Revision 0022 - General Electric Nuclear Boiler System Design Specification Data Sheets (DSDS) ~~(Design Input reference)~~ (Rev. 0 historical reference)

0A

4.12 NE-177, Revision 0000 - Nuclear Safety Related Specification for Limerick Generating Station Units 1&2 Power Rerate Operating Conditions (Power Rerate Information reference).

4.13 IISCP Project Letter to File M-P-PE001-0152 - Utilization of OPL-3 (Assumptions reference).

4.14 IISCP Project Anomaly No. 114, Head Corrections for PT-001-1(2)N076A/B/C/D (Applicability reference).

4.15 Action Request (Type CM NCR) A0851879 - Head Correction for PT-001-1N076A/B/C/D (Applicability reference).

4.16 Action Request (Type CM NCR) A0852289 - Head Correction for PT-001-2N076A/B/C/D (Applicability reference).

5.0 ATTACHMENTS

5.1 See Supporting Data Sheet Attachments located within this calculation.

6.0 ANALYSIS

6.1 Loop Effects

6.1.1 Loop ID No. PT-001-2N076C Config 01

6.1.2 Loop Function MAIN STEAM LINE C LOW PRESSURE - NS4 ISOLATION

6.1.3 Configuration Description MN STN LN C PRESS INDICATION

6.1.4 Loop Instrument List

Add new references shown below.

4.17 Transmittal of Design Information (TODI) ES1400026, Rev. 0, "Low Pressure Isolation Setpoint for the Limerick Station Loop Uncertainty Calculation"

4.18 General Electric Safety Concern SC05-03, dated 3/29/2005, "Potential to exceed Low Pressure Technical Specification Safety Limit"

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7.2 DL

DL = DE + DT

where:
DE = Σ D^2
DT = Σ DTE^2
DL = 0.00006

7.3 CL

CL = V + M

where:
V = Σ (setting tolerance)^2
M = Σ MTE^2
CL = 0.00006

7.4 TLU

(Positive)TLUp = [IR + PMAp + PEAp + PCp + PMAo + PEAo + PCo + √(AL + CL + DL + PMAr + PEAr + PCr)] * Loop span
(Negative)TLUn = [- PMAn - PEAn - PCn - PMAo - PEAo - PCo + -√(AL + CL + DL + PMAr + PEAr + PCr)] * Loop span

All other variables as previously defined.

TLUp = 21.47 PSIG
TLUn = -21.47 PSIG

7.5 NTSP

(increasing) NTSP = limit + [- PMAn - PEAn - PCn - PMAo - PEAo - PCo + (1.645 /sigma) * -√(AL + CL + DL + PMAr + PEAr + PCr)] * Loop span
(decreasing) NTSP = limit + [IR + PMAp + PEAp + PCp + PMAo + PEAo + PCo + (1.645 /sigma) * √(AL + CL + DL + PMAr + PEAr + PCr)] * Loop span

where:
limit = loop analytical or process limit
limit = ~~720.00~~ PSIG
805.00

where:
sigma = 2
NTSP = ~~737.66~~ PSIG
822.66

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7.6 ATSP

(increasing) ATSP = NTSP + margin
(decreasing) ATSP = NTSP - margin

where:
margin = additional margin supplied by calculation originator
margin = ~~-18.345~~ -17.345
ATSP = ~~756.00~~ 840.00 PSIG

7.7 Allowable Value

(Decreasing) AV = limit + [IR + PMAp + PEAp + PCp + PMAo + PEAo + PCo + (1.645 /sigma) * √(AL + CL + PMAr + PEAr + PCr)] * Loop span

(Increasing) AV = limit + [- PMAn - PEAn - PCn - PMAo - PEAo - PCo + (1.645 /sigma) * -√(AL + CL + PMAr + PEAr + PCr)] * Loop span

All other variables as previously defined.

AV = ~~736.00~~ 821.00 PSIG

0A

7.8 Acceptable Limits

Max = ATSP + [(√DL) * Loop span]
Min = ATSP - [(√DL) * Loop span]

All other variables as previously defined

Max = ~~765.078~~ 849.078 PSIG
Min = ~~746.923~~ 830.923 PSIG

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ATTACHMENT 2: Calculation Results

Device		Accuracy	Temperature Norm	Temperature Accid	Humidity Accid	Tol.	Pwr Supp
PT-001-2N076C	T	0.00500	0.00564	0.00000	0.00000	0.00500	0.00008
PIS-001-2N676C	S	0.00250	0.00000	0.00000	0.00000	0.00250	0.00000

Device		SPE	Rad. Accid	M&TE	Drift	Over Pres	Seismic
PT-001-2N076C	T	0.00000	0.00000	0.00500	0.00504	0.00000	0.00000
PIS-001-2N676C	S	0.00000	0.00000	0.00250	0.00000	0.00000	0.00000

0A

Process Concerns:	NORMAL Positive	NORMAL Negative	NORMAL Offsetting	ACCIDENT Positive	ACCIDENT Negative	ACCIDENT Offsetting
PMA	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
PEA	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
IR				0.00000		
Other	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

Loop Results:	NORMAL		ACCIDENT	
TLU*	-21.4656	21.46566	-21.4656	21.46566
AL	0.00003		0.00003	
	Increasing	Decreasing	Increasing	Decreasing
NTSP*	N/A	737.6555	N/A	737.6555
AV*	N/A	735.9990	N/A	735.9990
Acc Limits				
Min*:	N/A	746.9227	N/A	746.9227
Max*:	N/A	765.0781	N/A	765.0781
ATSP*	N/A	756.0004	N/A	756.0004

-17.3450

Additional Margin: ~~-18.3450~~ DL: 0.00006 CL: 0.00006

~~* These values are in PSIG~~

See section 2.2.6.

Replace with new IISCP software calculation results.

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ATTACHMENT 4: Loop Calibration Data

Process Temperature Units
Min 0.00 Max 0.00 Normal 0.00 Trip 0.00

Process Radiation Units
Min 0.000e+000 Max 0.000e+000 Normal 0.000e+000 Trip 0.000e+000

Process Humidity Units
Min 0.00 Max 0.00 Normal 0.00 Trip 0.00

Process Pressure Units
Min 0.00 Max 0.00 Normal 0.00 Trip 0.00
Loop Span: Min 0.00 Max 1200.00 Units PSIG
Sigma 2
Setpt: ~~756.00~~Units: PSIGReset: 0.00Units: Allw: ~~736.00~~Units PSIG
Des/Sfty Lmt: 0.00 Units Calibration Frequency 731
Loop Setting Tolerance : 0.000 Loop Leave Alone Zone : 6.708
Loop Cal Acc: 0.000 Analytical/Proc Lmt: ~~720.00~~ UnitsPSIG
805.00

Originator HUMPHREYS GD 05/09/94 Reviewer WHITE AJ 06/01/94

840.00

821.00

0A

Replace 720.00 with
805.00 (New Analytical
Limit from TODI).