December 8, 2009

NRC 2009-0120
10 CFR 50.90

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 261, Supplement 3
Extended Power Uprate

References:
(1) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)
(2) NRC letter to NextEra Energy Point Beach, LLC, dated November 5, 2009, Point Beach Nuclear Plant, Units 1 and 2 - Request for Additional Information from Technical Specifications Branch RE: Non-Conservative Setpoint Changes (TAC Nos. ME1083 and ME1084) (ML093060331)
(3) NextEra Energy Point Beach, LLC, Letter to NRC, dated June 17, 2009, License Amendment Request 261 Supplement 1, Extended Power Uprate (ML091690090)

Pursuant to 10 CFR 50.90, NextEra Energy Point Beach, LLC (NextEra) hereby submits Supplement 3 to License Amendment Request (LAR) 261 (Reference 1) for Point Beach Nuclear Plant (PBNP) Units 1 and 2. This supplement provides responses to NRC questions transmitted on November 5, 2009 (Reference 2). This supplement also provides revised proposed Technical Specification (TS) changes for the Reactor Protection System (RPS) Instrumentation TS Table 3.3.1-1, and Engineered Safety Features Actuation System (ESFAS) Instrumentation TS Table 3.3.2-1 that were previously submitted in Reference (1) and Reference (3). The proposed TS changes are required based on the NextEra response to NRC questions in Reference (2).

Enclosure 1 provides the NextEra responses to the NRC staff's questions transmitted in Reference (2). Enclosure 2 contains the proposed TS changes, including a determination that the proposed TS changes involve no significant hazards as defined in 10 CFR 50.92. An evaluation concludes that this change satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment.

Enclosure 3 contains a markup of proposed TS changes. These changes replace the TS Tables 3.3.1-1 and 3.3.2-1 previously submitted in Reference (1), Attachment 2 and Reference (3), Enclosure 3.

Enclosure 4 contains a markup of proposed TS Bases changes. These changes replace the TS Bases for B 3.3.1 and B 3.3.2 previously submitted in Reference (1), Attachment 3 and Reference (3) Enclosure 4. The Bases changes are provided for information. NRC approval is not being requested.

Enclosure 5 contains Licensing Report (LR) Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems, which replaces Section 2.4.1 previously submitted in Reference (1), Attachment 5.

Enclosure 6 contains Appendix E, Supplement to LR Section 2.4.1, which replaces the Appendix E previously submitted in Reference (1), Appendix E.

This letter contains no new regulatory commitments and no revisions to existing commitments.

The proposed TS changes have been reviewed by the Plant Operations Review Committee.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 8, 2009.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261 SUPPLEMENT 3
EXTENDED POWER UPRATE

The NRC staff determined that additional information was required (Reference 1) to enable the Technical Specifications Branch to continue its review of the RPS/ESFAS setpoints portion of License Amendment Request (LAR) 261, Extended Power Uprate (Reference 2). The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's questions.

Question 1

Explain how the proposed use of the Limited Safety System Setting (LSSS) term in Technical Specification (TS) Table 3.3.1-1 and Table 3.3.2-1 is in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.36(c)(ii)(A), as well as existing TS actions.

Background: The license amendment request proposes to replace the term "Allowable Value" with the term "Limited Safety System Setting" in the columns for TS Table 3.3.1-1 (Reactor Protection System (RPS)) and Table 3.3.2-1 (Engineered Safety Feature Actuation System). As a result, the proposed LSSS column will contain allowable values (i.e. values that contain margin to account for uncertainties associated with trip settings).

Defining the LSSS as an allowable value as opposed to a trip setting is inconsistent with the definition of LSSS in 10 CFR 50.36(c)(ii)(A). The regulation at 10 CFR 50.36(c)(ii)(A) states "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

In addition, the proposal appears to conflict with the current actions in Note 3.b of Table 3.3.1-1 and Note 1.b of Table 3.3.2-1. The current Notes refer to the LSSS as being the trip setting.

Technical Specification Task Force (TSTF)-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," is currently being developed by the Owners Groups and is the result of a major joint effort between the industry and the Nuclear Regulatory Commission (NRC) in resolving instrument setpoint methodology and LSSS issues. The licensee's proposed definition of LSSS is also not in accordance with the joint industry/NRC efforts which reflect the NRC's position on 10 CFR 50.36(c)(ii)(A).

Pertinent documents relevant to the development of TSTF-493, Revision 4, are:

- *NRC Reply to Industry Plan to Resolve TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions." March 9, 2009 (ML090560592).*
- *Industry Plan to Resolve TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions." February 23, 2009 (ML090540849).*
- *NRC Issues with TSTF-493, Revision 3. November 4, 2008 (ML082800484).*
- *TSTF-493, Revision 3, "Clarify Application of Setpoint Methodology for LSSS Functions." January 18, 2008 (ML080180441).*

It is unclear how the licensee's proposed use of the LSSS term is in accordance with 10 CFR 50.36(c)(ii)(A), current TS actions, or joint industry/NRC efforts.

NextEra Response

NextEra will retain the Allowable Value terminology in Technical Specification (TS) Tables 3.3.1-1, RPS Instrumentation, and 3.3.2-1, ESFAS Instrumentation (see proposed Technical Specification changes in Enclosure 3). Changing the column heading from "Allowable Value" to "Limiting Safety System Setting" would have resulted in the Function tables not conforming to NUREG-1431 (Reference 3). Retaining the Allowable Value (AV) column is consistent with industry efforts to clarify the application of setpoint methodology for limiting safety system setting functions and Reference (3).

Question 2

Explain why the Underfrequency Bus A01 and A02 reactor trips are no longer proposed to be associated with the operability of the P-7 interlock.

Background: TS Table 3.3.1-1 (RPS) contains Function 17.b, "Reactor Trip System Interlock-Low Power Reactor Trips Block, P-7." The Bases provides a description of the reactor trips that the P-7 interlock automatically enables and disables. The Bases deletes the reference to the Underfrequency Bus A01 and A02 trips. However, no justification for this change appears to be provided in the submittal. In addition, the Bases for Function 12 "Underfrequency Bus A01 and A02," indicates that the P-7 interlock still automatically enables and disables this reactor trip function.

The regulation at 10 CFR 50.36(c)(ii)(A) states "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

It is unclear why the Underfrequency Bus A01 and A02 reactor trips are no longer proposed to be associated with the operability of the P-7 interlock.

NextEra Response

An underfrequency signal received simultaneously on buses A01 and A02 causes both reactor coolant pump (RCP) breakers to trip. RCP breaker position causes a reactor trip (Function 10). Since the P-7 Permissive is indirectly associated with the bus underfrequency trip (Function 12) through Function 10, NextEra has decided not to remove this information from the TS Bases (See Enclosure 4).

References

(1) NRC letter to NextEra Energy Point Beach, LLC, dated November 5, 2009, Point Beach Nuclear Plant, Units 1 and 2 - Request for Additional Information from Technical Specifications Branch RE: Non-Conservative Setpoint Changes (TAC Nos. ME1083 and ME1084) (ML093060331)

(2) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)

(3) NUREG-1431, Revision 3, dated June 30, 2004, Standard Technical Specifications Westinghouse Plants (ML041830612 and ML041830205)

ENCLOSURE 2

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261 SUPPLEMENT 3

EVALUATION OF CHANGES
REACTOR PROTECTION SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

34 pages follow

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, NextEra Energy Point Beach, LLC (NextEra), proposes to revise the Technical Specifications (TS) for Point Beach Nuclear Plant (PBNP) Units 1 and 2, in support of License Amendment Request (LAR) 261, Extended Power Uprate (EPU) (Reference 1).

The proposed changes replace the markups of TS Table 3.3.1-1 (RPS Instrumentation) and TS Table 3.3.2-1 (ESFAS Instrumentation) previously submitted in Reference (1), Attachment 2, with the markup included in Enclosure 3. The proposed changes also replace the change proposed to TS Table 3.3.2-1 Item 6.e submitted in Reference (2). Additionally, EPU LAR 261, Attachment 5, Licensing Report (LR) Section 2.4.1, Reactor Protection, Safety Feature Actuation, and Control Systems, and Appendix E, RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1, are replaced and included in Enclosures 5 and 6, respectively.

2.0 DETAILED DESCRIPTION

Markups of TS Table 3.3.1-1 and TS Table 3.3.2-1 were submitted in Reference (1), Attachment 2. The changes proposed in Reference (1) included changing the column heading from "Allowable Value" to "Limiting Safety System Setting" and updating the column inputs and notes as necessary to agree with revised calculations. The renaming of the column heading would have resulted in the tables not conforming to NUREG-1431 (Reference 3). After discussion with the NRC, NextEra decided to retain the "Allowable Value" heading title to be consistent with Reference (3) and consistent with guidance related to the joint NRC and industry efforts to clarify the application of setpoint methodology for Limiting Safety System Settings (LSSS) functions. A new column entitled "Nominal Trip Setpoint" was also added for consistency with Reference (3), and joint NRC and industry efforts.

A detailed description of the associated proposed TS changes is provided below for each change. Proposed markups for TS Table 3.3.1-1 and TS Table 3.3.2-1 are provided in Enclosure 3 which replace the markups of TS Table 3.3.1-1 and TS Table 3.3.2-1 previously submitted in Reference (1), Attachment 2 and the markup of Table 3.3.2-1, Item 6.e submitted in Reference (2).

Proposed markups for the Bases for Section B 3.3.1 and Section B 3.3.2 are provided in Enclosure 4 for information and replace the markups of the Bases for Section B 3.3.1 and Section B 3.3.2 previously submitted in Reference (1), Attachment 3, and markups of Section B 3.3.2, Item 6.e previously submitted in Reference (2), Enclosure 4. NRC approval is not being requested for the Bases.

2.1 <u>Proposed TS 3.3.1 Changes</u>

The following replaces the subsection entitled, "6. Technical Specification 3.3.1, RPS Instrumentation," included in Reference (1), Attachment 1, Section 3.1, Pages 1.0-4 through 1.0-14.

6. Technical Specification 3.3.1, RPS Instrumentation.

   This section contains changes to Allowable Values (AV) for Reactor Protecton System (RPS) setpoints and adds Nominal Trip Setpoints.

Technical Specification Table 3.3.1-1 (RPS Instrumentation) is being revised as follows:

1) A new column entitled NOMINAL TRIP SETPOINT is added to convert the table to a "multiple column" format allowed by Reference (3). Values in the Nominal Trip Setpoint (NTSP) column are predetermined nominal field trip setpoints contained in channel calibration procedures and evaluated in setpoint calculations.
2) Values in the column entitled ALLOWABLE VALUE are revised for individual functions identified below, either to reflect changes resulting from EPU or to provide a setpoint that accounts for instrument uncertainty that replaces current values that are not consistent with the AV calculations presented in this submittal.
3) Notes 3 and 4 are added to the table to specify channel operability criteria related to comparing the as-found setting value to as-left and as-found tolerances during surveilllance testing.

Specific changes to TS Table 3.3.1-1 are as follows:

a. TS Table 3.3.1-1 adds a new column identified as "NOMINAL TRIP SETPOINT" and populates the column with nominal field trip setpoint values.

   Licensing Report Sections: Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems

   Basis for the change: To convert the table to a "multiple column" format as allowed by Reference (3). This change also incorporates guidance based on joint NRC and industry efforts to clarify the application of setpoint methodology for LSSS functions.

b. TS Table 3.3.1-1 revises the values in the ALLOWABLE VALUE column.

   Licensing Report Sections: Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems

   Basis for the change: As defined in 10 CFR 50.36, AV are limiting settings for automatic protective devices related to those variables having significant safety functions. 10 CFR 50.36 requires that these limiting settings be included in the TS.

   The AV for reactor trip functions in TS Table 3.3.1-1 are calculated based on limits from the safety analyses, process limits for the instrumentation, and instrument loop uncertainties calculated with 95% probability and 95% confidence to industry standard methodology. The AV for reactor trip system interlocks are calculated based on nominal setpoints used in the analyses and as-found acceptance criteria. The methods used to determine AV and summaries of setpoint calculations are provided in Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

   The AV proposed for TS Table 3.3.1-1 are limiting as-found settings for the Channel Operational Test and Channel Calibration of adjustable setpoints and are calculated such that when as-found settings are within the AV, there is 95% probability and 95% confidence that the trip will occur prior to the process variable exceeding the established limit. For interlocks, AV insure the interlock, permissive or block function will occur in accordance with the assumptions of the analyses. Therefore, the assumptions of the safety analyses and results are protected by the proposed AV.

Table 1.0-1 below identifies functions for which the AV were either affected by EPU or are revised to be consistent with setpoint calculations not impacted by EPU. These AV have been evaluated using the methods described in Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6). Four RPS functions that are shown in the table (Reactor Coolant Flow Low (single loop and two loops), Undervoltage Bus A01 & A02, Underfrequency Bus A01 & A02, and Steam Flow/Feedwater Flow Mismatch), have been evaluated using the methods described in Appendix E, Supplement to LR Section 2.4.1 and have shown that no change to the AV are required for these functions.

Table 1.0-1
Reactor Protection System Allowable Value TS Changes and Notes 3 and 4 Application

Item	Function	Notes 3 & 4 Apply	EPU Related	Allowable Value Change
2.a.	Power Range Neutron Flux – High	X	X	X
2.b.	Power Range Neutron Flux – Low	X		X
3.	Intermediate Range Neutron Flux	X		X
4.	Source Range Neutron Flux	X		X
5.	Overtemperature ΔT	X	X	
6.	Overpower ΔT	X	X	
7.a.	Pressurizer Pressure – Low	X	X	X
7.b.	Pressurizer Pressure – High	X	X	X
8.	Pressurizer Water Level – High	X		X
9.a.	Reactor Coolant Flow Low - Single Loop	X		
9.b.	Reactor Coolant Flow Low - Two Loops	X		
11.	Undervoltage Bus A01 & A02	X		
12.	Underfrequency Bus A01 & A02	X		
13.	Steam Generator Water Level – Low Low	X	X	X
14.	Steam Generator Water Level-Low-Coincident with Steam Flow/Feedwater Flow Mismatch	X		X
17.a.	Intermediate Range Neutron Flux, P-6			X
17.b.(1).	Low Power Reactor Trip Block, P-7, Power Range Neutron Flux			X
17.b.(2)	Low Power Reactor Trip Block, P-7, Turbine Impulse Pressure			X
17.c.	Power Range Neutron Flux, P-8		X	X
17.d.	Power Range Neutron Flux, P-9		X	X
17.e.	Power Range Neutron Flux, P10			X

c. Footnote (m) is added which reads, "Table 3.3.1-1 Notes 3 and 4 are applicable," and new Notes 3 and 4 are added to TS Table 3.3.1-1.

Licensing Report Sections: LR Section 2.4.1.2.3.2, NSSS Analyses and Evaluations, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: New Notes 3 and 4 are added to TS Table 3.3.1-1 to include guidance based on joint NRC and industry efforts to clarify the application of setpoint methodology for LSSS functions. The Notes are applied to the Channel Operational Test (COT) surveillance and the Channel Calibration surveillance for RPS functions with adjustable setpoints, with the exception of the Function 17 Reactor Trip System Interlocks.

Note 3 states:

> "If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service."

Note 4 states:

> "The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in FSAR Section 7.2."

Both Notes 3 and 4 will apply to each of those RPS functions that contain specific values in the ALLOWABLE VALUE column in the marked-up TS Table 3.3.1-1 for surveillances where a specific value is measured (i.e., COT or Channel Calibration). The notes are not applied to those functions that contain "NA" in the AV column and also are excluded from Function 17, Reactor Trip System Interlocks.

Notes 3 and 4 will apply to the functions as indicated in Table 1.0-1.

Surveillance limits are established to verify that reactor protection system instrumentation with an AV in TS Table 3.3.1-1 operates within the boundaries of applicable safety analyses, considering all instrument uncertainties. These limits are implemented in plant procedures in accordance with Notes 3 and 4 above. The determination of as-left setting tolerance and as-found criteria is described in Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

The implementation of as-left and as-found limits verifies that the instrument loops are performing in accordance with uncertainty calculation assumptions and that out-of-tolerance conditions are evaluated. If a channel cannot be set within the as-left tolerance band, the channel is declared INOPERABLE and Notes 3 and 4 apply. FSAR Section 7.2 will be revised during implementation of these changes to describe the methodology used to determine the as-found and as-left tolerances.

Footnote (m) and Notes 3 and 4 do not apply to those functions that have an "NA" as a value in the ALLOWABLE VALUE column. The notes also do not apply to the Reactor Trip System Interlocks (Function 17).

d. Function 1., Manual Reactor Trip

The Nominal Trip Setpoint is added as "NA" (two places).

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

e. Function 2.a., Power Range Neutron Flux – High

1) The AV is revised from "$\leq$ 108% RTP" to "$\leq$ 109% RTP."

Licensing Report Sections: Section 2.8.5.4.2, Uncontrolled Rod Withdrawal at Power, LR Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being implemented to support operation at EPU conditions. The uncontrolled rod withdrawal at power event for EPU assumes that the Reactor Protection System (RPS) is actuated at a conservative value of 116% of nominal full power. The AV is established by calculation to avoid exceeding the new analytical limit, taking all instrument uncertainties into account.

2) The Nominal Trip Setpoint is added as "107% RTP."

Licensing Report Sections: LR Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

f. Function 2.b., Power Range Neutron Flux – Low

1) The AV is revised from "$\leq$ 25% RTP" to "$\leq$ 28% RTP."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to replace an existing value that is not consistent with the AV calculation contained in this submittal. The subcritical rod withdrawal, rod ejection, and steam line break outside containment analyses assume trip actuation at 35% RTP. This analytical limit is applicable at both EPU conditions and current licensed power level. The AV is established by calculation to avoid exceeding the new analytical limit, taking all instrument uncertainties into account.

2) The Nominal Trip Setpoint is added as "20% RTP."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

3) Footnote (b) - "interlocks" is revised to "interlock."

Licensing Report Sections: None

Basis for the change: This is an administrative change. There is only one interlock addressed by this Note.

g. Function 3., Intermediate Range Neutron Flux

1) The AV is revised from "≤ 40% RTP" to "≤ 43% RTP."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to replace existing values that are not consistent with the AV calculations contained in this submittal. The AV is calculated based on the nominal setpoint and as-found tolerance values and is valid at current licensed power level and EPU conditions.

2) The Nominal Trip Setpoint is added as "25% RTP."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

h. Function 4., Source Range Neutron Flux

1) The AV is revised in two places from "within span of instrumentation" to "≤ 4.0 E5 cps."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to be consistent with NUREG-1431 (Reference 3). The AV is calculated based on the upper range limit and is valid at current licensed power level and EPU conditions.

2) The Nominal Trip Setpoint is added in two places as "2.0 E5 cps."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

i. Function 5., Overtemperature ΔT, Note 1

1) Note 1 states that the T', P', K1, K2, and K3 values are applicable for operation at both 2000 psia and 2250 psia. The differentiation of these values for cores containing 422V+ fuel assemblies and cores not containing 422V+ fuel assemblies were deleted. The differentiation between 2000 psia and 2250 psia operation was deleted.

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.1, Reactor Protection System, and Section 2.8.5.0, Non-LOCA Analyses Introduction.

Basis for the change: The NSSS design parameters provide the RCS and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for the design transients, systems, structures, components, accidents, and fuel analyses and evaluations. One of the major input parameters and assumptions used in the calculation of the four cases of NSSS design parameters established for EPU is 14x14 422V+ fuel. Therefore, 422V+ is the only fuel that has been evaluated for EPU. In addition, 2000 psia operation was not analyzed and will not be allowed under EPU conditions.

2) The $f(\Delta I)$ function description was revised to delete the following:

"and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within –[*], +[*] percent, $f(\Delta I) = 0$ for cores not containing 422V+ fuel assemblies; for $q_t - q_b$ within –[*], +[*] percent, $f(\Delta I) = 0$ for cores containing 422V+ fuel assemblies.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power for cores not containing 422V+ fuel assemblies and reduced by an equivalent of [*] percent of rated power for cores containing 422V+ fuel assemblies.

(c) for cores not containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds -[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds -[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power."

and add the following:

"f (ΔI) = [*] {[*] – ($q_t - q_b$)}	when ($q_t - q_b$) ≤ [*]% RTP
0% of RTP	when [*]% RTP < ($q_t - q_b$) ≤ [*]% RTP
[*] {($q_t - q_b$) – [*]}	when ($q_t - q_b$) > [*]% RTP

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and ($q_t + q_b$) is the total THERMAL POWER in percent RTP."

Licensing Report Sections: None

Basis for the change: The f (ΔI) description is being modified to delete references to a fuel type that has not been evaluated for use at EPU conditions. The revised f(ΔI) equation is consistent with Reference (3).

3) An * was added prior to the note that reads "The values denoted with [*] are specified in the COLR."

Licensing Report Sections: None

Basis for the change: This is an administrative change. The * is added to be consistent with Reference (3).

4) The differentiation of the Rosemont or equivalent and Sostman or equivalent RTDs for τ_3 and τ_4 was deleted.

Licensing Report Sections: None

Basis for the change: Only Rosemont RTDs or equivalent are used at PBNP.

5) The Nominal Trip Setpoint is added as "Refer to Note 1 (Page 3.3.1-18)."

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.1, Reactor Protection System, and Section 2.8.5.0, Non-LOCA Analyses Introduction.

Basis for the change: The new Nominal Trip Setpoint column value refers to the same note as the AV. This is consistent with the "multiple column" format in Reference (3) for the function.

j. Function 6., Overpower ΔT, Note 2

1) Note 2 states that the T', K_4, and K_6 values are applicable for operation at both 2000 psia and 2250 psia operation. The differentiation of these values for cores containing 422V+ fuel assemblies and cores not containing 422V+ fuel assemblies and differentiation between 2000 psia and 2250 psia operation were deleted. PBNP operates at 2250 psia with 422V+ fuel being the only fuel type evaluated for operation at EPU conditions.

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.8.5.0, Non LOCA Analyses Introduction, and Section 2.4.1.2.3.2.1, Reactor Protection System.

Basis for the change: The NSSS design parameters provide the RCS and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for the design transients, systems, structures, components, accidents, and fuel analyses and evaluations. One of the major input parameters and assumptions used in the calculation of the four cases of NSSS design parameters established for EPU is 14x14 422V+ fuel. Therefore, 422V+ is the only fuel that has been evaluated at EPU conditions.

The analyses and evaluations performed for the EPU and discussed in the LR Section 1.0 only considered a nominal RCS pressure of 2250 psia.

2) "Rated power" was revised to "RTP" for ΔT_0.

Licensing Report Sections: None

Basis for the change: This is an editorial change.

3) The differentiation of the Rosemont or equivalent and Sostman or equivalent RTDs for τ_3 and τ_4 was deleted.

Licensing Report Sections: None

Basis for the change: Only Rosemont RTDs or equivalent are used at PBNP.

4) An * was added prior to the note that reads "The values denoted with [*] are specified in the COLR."

Licensing Report Sections: None

Basis for the change: This is an administrative change. The * is added to be consistent with Reference (3).

5) The Nominal Trip Setpoint is added as "Refer to Note 2 (Page 3.3.1-20)."

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.8.5.0, Non LOCA Analyses Introduction, and Section 2.4.1.2.3.2.1, Reactor Protection System.

Basis for the change: The new Nominal Trip Setpoint column value refers to the same note as the AV. This is consistent with the "multiple column" format in Reference (3) for this function.

k. Function 7.a., Pressurizer Pressure – Low

1) The AV is revised from Footnote "(h)" which states "$\geq$ 1905 psig during operation at 2250 psia, or $\geq$ 1800 psig during operation at 2000 psia" to "$\geq$ 1860 psig." Footnote (h) is deleted and replaced with a new footnote (h) for Function 17.d.

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.1, Reactor Protection System and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being implemented to support operation at EPU conditions. Operation at 2000 psia was not analyzed and will not be allowed under EPU conditions. The OPTOAX code analysis assumes that the RPS is actuated at 1840 psig. The AV is established by calculation to avoid exceeding the new analytical limit, taking all instrument uncertainties into account.

2) The Nominal Trip Setpoint is added as "1925 psig."

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.1, Reactor Protection System and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

l. Function 7.b., Pressurizer Pressure - High

1) The AV is revised from Footnote "(i)" which states "$\leq$ 2385 psig during operation at 2250 psia, or $\leq$ 2210 psig during operation at 2000 psia," to "$\leq$ 2385 psig." Footnote (i) is deleted and replaced with a new footnote (i) for Function 17.d.

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1, (Enclosure 6).

Basis for the change: This change is being implemented to support operation at EPU conditions. Operation at 2000 psia was not analyzed and will not be allowed under EPU conditions. The Loss of External Load / Turbine Trip analysis assumes that the RPS is actuated at 2403 psig. The AV is established by calculation to avoid exceeding the new analytical limit, taking all instrument uncertainties into account.

2) The Nominal Trip Setpoint is added as "2365 psig."

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1, (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

m. Function 8., Pressurizer Water Level - High

1) The AV is revised from "$\leq$ 95% of span" to "$\leq$ 85%"

Licensing Report Sections: Appendix E, Supplement to LR Section 2.4.1, (Enclosure 6) and Section 2.4.1.2.3.2.1, Reactor Protection System.

Basis for the change: This change is being made to replace current values that are not consistent with the AV calculations and contained in this submittal. A process limit of 100% of instrument span is used in the calculation of AV taking all instrument uncertainties into account. The phrase "of span" is being removed to be consistent with Reference (3). This value is applicable at both current licensed power level and EPU conditions.

2) Nominal Trip Setpoint is added as "80%."

Licensing Report Sections: Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6) and Section 2.4.1.2.3.2.1, Reactor Protection System.

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

n. Function 9.a., Reactor Coolant Flow-Low – Single Loop

   Nominal Trip Setpoint is added as "93%."

   Licensing Report Sections: Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6) and Section 2.4.1.2.3.2.1, Reactor Protection System.

   Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

o. Function 9.b., Reactor Coolant Flow-Low – Two Loops

   Nominal Trip Setpoint is added as "93%."

   Licensing Report Sections: Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6) and Section 2.4.1.2.3.2.1, Reactor Protection System.

   Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

p. Functions 10.a. and 10.b., Reactor Coolant Pump (RCP) Breaker Position – Single Loop and Two Loops

   The Nominal Trip Setpoint is added as "NA" in two places.

   Licensing Report Sections: None

   Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

q. Function 11., Undervoltage Bus A01 & A02

   The Nominal Trip Setpoint is added as "3170 V."

   Licensing Report Sections: Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6) and Section 2.4.1.2.3.2.1, Reactor Protection System.

   Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

r. Function 12., Underfrequency Bus A01 & A02

1) The Nominal Trip Setpoint is added as "57 Hz."

Licensing Report Sections: Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6) and Section 2.4.1.2.3.2.1, Reactor Protection System.

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

s. Function 13., Steam Generator (SG) Water Level - Low Low

1) The AV is revised from "≥ 20% of span" to "≥ 29.3%"

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, Section 2.8.5.2.3, Loss of Normal Feedwater Flow and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being implemented to support operation at EPU conditions. The Loss of Normal Feedwater Flow/Loss of All AC Power analyses assume that the RPS is actuated at 20% of the narrow range span. The AV is established by calculation to avoid exceeding the new analytical limit, taking all instrument uncertainties into account. The phrase "of span" is being removed consistent with NUREG-1431 (Reference 3).

2) The Nominal Trip Setpoint is added as "30%."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, Section 2.8.5.2.3, Loss of Normal Feedwater Flow and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint required for EPU for this function. The function is changing from "25%" to "30%." The NTSP is being added to conform to the "multiple column" format in Reference (3).

t. Function 14., SG Water Level - Low

1) The AV for SG Water Level - Low is revised from "NA" to "≥ 10%"

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to incorporate the AV determined by calculation. Because Steam Generator Water Level - Low is a backup trip function, a process limit of 0% of instrument span is used in the AV calculation rather than an Analytical Limit.

2) The Nominal Trip Setpoint is added as "30%."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

u. Function 14., SG Water Level - Low Coincident with Steam Flow/Feedwater Flow Mismatch

The Nominal Trip Setpoint is added as "0.8 E6 lbm/hr."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

v. Function 15.a., Turbine Trip – Low Autostop Oil Pressure

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

w. Function 15.b., Turbine Trip - Turbine Stop Valve Closure

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

x. Function 16., Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

y. Function 17., Reactor Trip System Interlocks

1.) 17.a. Intermediate Range Neutron Flux, P-6 – AV is revised from "> 1E-10 amp" to "≥ 4E-11 amp."

17.b. (1) Low Power Reactor Trips Block, P-7, Power Range Neutron Flux - AV is revised from "< 10% RTP" to "≤ 13% RTP."

17.b. (2) Low Power Reactor Trips Block, P-7, Turbine Impulse Pressure - AV is revised from "< 10% turbine power" to "≤ 12.8% turbine power."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: These changes are being made to replace existing values that are not consistent with the AV calculations. The AV are calculated based on the nominal setpoint and as-found tolerance values and are valid at current licensed power level and EPU conditions.

The proposed AV setpoint for the P-7, Turbine Impulse Pressure submitted in the EPU LAR 261 Attachment 2 Reference (1) as "≤ 13% turbine power" is revised to "≤ 12.8% turbine power" to match the recommended AV from the Turbine Impulse Pressure Low Power Permissive P-7 Instrument Scaling and Uncertainty Calculation (See Enclosure 6 Calculation Summary).

2.) 17.a. Intermediate Range Neutron Flux, P-6 – Nominal Trip Setpoint is added as "1E-10 amp."

17.b. (1) Low Power Reactor Trips Block, P-7, Power Range Neutron Flux -Nominal Trip Setpoint is added as "10% RTP."

17.b. (2) Low Power Reactor Trips Block, P-7, Turbine Impulse Pressure -Nominal Trip Setpoint is added as "10% turbine power."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

3.) Function 17.c., Power Range Neutron Flux, P-8

The AV for the P-8 function is revised from "< 50% RTP" to "≤ 38% RTP."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Section 2.4.2.1, Plant Operability.

Basis for the change: The P-8 permissive setpoint was changed by EPU to the nominal steady-state power level the reactor can operate with one RCS loop inactive without violating core thermal limits at uprated conditions. The AV is calculated based on the nominal setpoint and as-found tolerance value.

4.) Function 17.c., Power Range Neutron Flux, P-8

The Nominal Trip Setpoint is added as "35% RTP."

Licensing Report Sections: Section 2.4.1.2.3.2.1, Reactor Protection System, and Section 2.4.2.1, Plant Operability.

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

5.) Function 17.d., Power Range Neutron Flux, P-9

The AV for the P-9 function is revised from "< 50% RTP" to footnote "(h)" New footnote (h) will read "$\leq$ 38% RTP for full design power T_{avg} < 572°F or $\leq$ 53% RTP for full design power T_{avg} $\geq$ 572°F. For end of EOC coastdown, P-9 is not reset if T_{avg} decreases to < 572°F."

Licensing Report Sections: LR Section 2.4.1.2.3.2.1, Reactor Protection System, LR Section 2.4.2.1, Plant Operability, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The permissive nominal setpoint to reinstate two reactor trips on turbine trip is changed by EPU and is now related to full design power T_{avg}. The AV are calculated based on the nominal setpoint and as-found tolerance values. The phrase "full design power" is used in the footnote to clarify that the note is not referring to a specific operating $T_{avg,}$ but rather a nominal T_{avg} at 100% power.

6.) Function 17.d., Power Range Neutron Flux, P-9

The Nominal Trip Setpoint for the P-9 function is added as footnote "(i)." New footnote (i) will read "35% RTP for full design power T_{avg} < 572°F or 50% RTP for full design power T_{avg} $\geq$ 572°F. For EOC coastdown, P-9 is not reset if T_{avg} decreases to < 572°F."

Licensing Report Sections: LR Section 2.4.1.2.3.2.1, Reactor Protection System, LR Section 2.4.2.1, Plant Operability, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The permissive Nominal Trip Setpoint to reinstate two reactor trips on turbine trip is changed by EPU. The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

7.) Function 17.e., Power Range Neutron Flux, P-10

The AV for the P-10 function is revised from "> 8% RTP and < 10% RTP" to "$\geq$ 6% RTP and $\leq$ 12% RTP."

Licensing Report Sections: LR Section 2.4.1.2.3.2.1, Reactor Protection System, LR Section 2.4.2.1, Plant Operability, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The AV range for the P-10 permissive is revised to coincide with the upper and lower as-found limits for a Nominal Trip Setpoint of 9% (decreasing) as allowed during a Channel Operational Test (COT). Both limits are listed to be consistent with Reference (3), although only the lower limit provides the AV for the P-10 safety function of automatically clearing the reactor trip bypass on decreasing power.

8.) Function 17.e., Power Range Neutron Flux, P-10

The Nominal Trip Setpoint for the P-10 function is added as "9% RTP."

Licensing Report Sections: LR Section 2.4.1.2.3.2.1, Reactor Protection System, LR Section 2.4.2.1, Plant Operability, and Appendix E, Supplement to LR Section 2.4.1

(Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

z. Function 18., Reactor Trip Breakers (RTBs)

The Nominal Trip Setpoint is added as "NA" (two places).

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

aa. Function 19., Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The Nominal Trip Setpoint is added as "NA" (two places).

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

bb. Function 20., Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism

The Nominal Trip Setpoint is added as "NA" (two places).

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

cc. Function 21., Automatic Trip Logic

The Nominal Trip Setpoint is added as "NA" (two places).

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

2.2 Proposed TS 3.3.2 Changes

The following replaces the subsection entitled, "7. Technical Specification 3.3.2, ESFAS Instrumentation." included in Reference (1), Attachment 1, Section 3.1, Pages 1.0-14 through 1.0-22.

7. Technical Specification 3.3.2, ESFAS Instrumentation.

   This section contains changes to AV for ESFAS setpoints and adds Nominal Trip Setpoints. In summary, TS Table 3.3.2-1 (ESFAS Instrumentation) is being revised as follows:

   1) A new column entitled NOMINAL TRIP SETPOINT is added to convert the table to a "multiple column" format allowed by Reference (3). Values in the NTSP column are

predetermined nominal field trip setpoints contained in channel calibration procedures and evaluated in setpoint calculations.

2) Values in the column entitled ALLOWABLE VALUE are revised for individual functions identified below, either to reflect changes resulting from EPU or to provide a setpoint that accounts for instrument uncertainty to replace existing values that are not consistent with the AV calculations and contained in this submittal.

3) Two notes are added to the table to specify channel operability criteria related to comparing the as-found setting value to as-left and as-found tolerances during surveillance testing.

Specific changes to Table 3.3.2-1 are described below:

a. Table 3.3.2-1 adds a new column heading identified as "NOMINAL TRIP SETPOINT" and populates the column with nominal field trip setpoint values.

   Licensing Report Sections: LR Section 2.4.1.2.3.2, NSSS Analyses and Evaluations, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

   Basis for the change: To convert the table to a "multiple column" format as allowed by Reference (3).

b. Table 3.3.2-1 revises the values in the ALLOWABLE VALUE column.

   Licensing Report Sections: LR Section 2.4.1.2.3.2, NSSS Analyses and Evaluations, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

   Basis for the change: As defined in 10 CFR 50.36, AV are limiting settings for automatic protective devices related to those variables having significant safety functions. 10 CFR 50.36 requires that limiting settings be included in the Technical Specifications.

   The AV for trip functions are calculated based on limits from the safety analyses, process limits for the instrumentation, and instrument loop uncertainties calculated with 95% probability and 95% confidence to industry standard methodology. The AV for operating bypasses (interlocks) are calculated based on nominal setpoints used in the analyses and as-found acceptance criteria. The methods used to determine AV and summaries of calculations are provided in Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

   The AV proposed for TS Table 3.3.2-1 are limiting values for the Channel Operational Test and Channel Calibration of adjustable setpoints, and are calculated such that when as-found settings are within the AV, there is 95% probability and 95% confidence that the protective action will occur prior to the process variable exceeding the established limit. For interlocks, AV ensure the interlock, permissive or block function will occur in accordance with the assumptions of the analyses. Therefore, the assumptions of the safety analyses and results are protected by the proposed AV.

   Table 1.0-2 below identifies functions for which the AV were either affected by EPU or are changed to obtain consistency with setpoint calculations not impacted by EPU. These AV have been evaluated using the methods described in Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6). ESFAS Function 6.d in Table 3.3.2-1

(Auxiliary Feedwater-Undervoltage Bus A01 and A02) was evaluated and found to not require an AV change. The remaining ESFAS functions with adjustable setpoints have revised AV, for the reasons described below.

c. Footnote (f) which reads, "Table 3.3.2-1 Notes 1 and 2 are applicable" and new Notes 1 and 2 are added to Table 3.3.2-1.

Licensing Report Sections: LR Section 2.4.1.2.3.2, NSSS Analyses and Evaluations, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: New Notes 1 and 2 are added to TS Table 3.3.2-1 to include guidance based on joint NRC and industry efforts to clarify the application of setpoint methodology for LSSS functions. The Notes are applied to the Channel Operational Test (COT) surveillance and the Channel Calibration surveillance for those ESFAS functions with adjustable setpoints, with the exception of the Function 7, SI Block – Pressurizer Pressure.

Note 1 states:

"If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service."

Note 2 states:

"The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in FSAR Section 7.2."

Notes 1 and 2 apply to the following functions as indicated:

Table 1.0-2
ESFAS Instrumentation Allowable Value TS Changes and Notes 1 and 2 Application

Item	Function	Notes 1 & 2 Apply	EPU Related	Allowable Value Change
1.c.	Safety Injection on Containment Pressure- High	X		X
1.d.	Safety Injection on Pressurizer Pressure- Low	X		X
1.e. and note (c)	Safety Injection on Steam Line Pressure- Low	X	X	X
2.c.	Containment Spray on Containment Pressure - High High	X		X
4.c.	Steam Line Isolation – Containment Pressure – High High	X		X
4.d.	Steam Line Isolation on High Steam Flow Coincident with Safety Injection and T_{avg}- Low	X	X	X
4.e.	Steam Line Isolation on High High Steam Flow Coincident with Safety Injection	X	X	X
5.b.	Feedwater Isolation on SG Water Level – High	X		X
6.b.	Auxiliary Feedwater on SG Water Level- Low Low	X	X	X
6.d.	Undervoltage Bus A01 and A02	X		
6.e.	AFW Pump Suction Transfer on Suction Pressure Low	X	X	(Function added by EPU)
7. (was 8.)	SI Block-Pressurizer Pressure			X

Surveillance limits are established to verify that ESFAS instrumentation with an ALLOWABLE VALUE in TS Table 3.3.2-1 operates within the boundaries of applicable safety analyses, considering all instrument uncertainties. These limits are implemented in plant procedures in accordance with Notes 1 and 2 above. The determination of as-left setting tolerance and as-found criteria is described in Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Implementation of as-left and as-found limits verifies that the instrument loops are performing in accordance with uncertainty calculation assumptions and that out-of-tolerance conditions are evaluated. If a channel cannot be set within the as-left tolerance band, the channel is declared inoperable and Notes 1 and 2 apply. FSAR Section 7.2 will be revised during implementation of these changes to describe the methodology used to determine the as-found and as-left tolerances.

The new notes will apply to each of those ESFAS functions that contain specific values in the ALLOWABLE VALUE column in marked-up TS Tables 3.3.2-1 for surveillances where a specific value is measured (i.e., COT or Channel Calibration). The notes are not applied to those functions that contain "NA" in the AV column and are not applied to Function 7, SI Block – Pressurizer Pressure.

d. Function 1.a., Safety Injection – Manual Initiation

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

e. Function 1.b., Safety Injection – Automatic Actuation Logic and Actuation Relays

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

f. Function 1.c., Safety Injection - Containment Pressure - High

1) The AV is revised from "≤ 6 psig" to "≤ 5.3 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to replace an existing value that is not consistent with the AV calculation and presentation in this submittal. An analytical limit of 6 psig for the steam line break inside containment analysis is used in the calculation of AV taking all instrument uncertainties into account. This value is applicable at current licensed power level and EPU conditions.

2) The Nominal Trip Setpoint is added as "5 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

g. Function 1.d., Safety Injection - Pressurizer Pressure - Low

1) The AV is revised from "≥ 1715 psig" to "≥ 1725 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to replace existing values that are not consistent with the AV calculations and contained in this submittal. A process limit of 1700 psig (the lower limit of the instrument span) is used in the calculation of AV, taking all instrument uncertainties into account. This value is applicable at current licensed power level and EPU conditions.

2) The Nominal Trip Setpoint is added as "1735 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP is being added to conform to the "multiple column" format in Reference (3).

3) Note (a): The pressurizer pressure at which the Safety Injection - Pressurizer Pressure - Low function applies in MODES 1, 2 and 3 is revised from "> 1800 psig" to "> 2000 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The change from "> 1800 psig" to "> 2000 psig" is being made to coincide with the change in the SI Block - Pressurizer Pressure function (Function 7 in Table 3.3.2-1) AV from "≤ 1800 psig" to "≤ 2005 psig." The previous 1800 psig value in Note (a) was based on the previous normal plant operating pressure of 1985 psig and a corresponding nominal SI Block setpoint of 1800 psig that automatically removed the manual bypass of the SI Pressurizer Pressure - Low signal on increasing pressurizer pressure above 1800 psig during a normal plant heatup. The new 2000 psig value in Note (a) is based on the current (and EPU) normal plant operating pressure of 2235 psig and a revised SI unblock nominal setting of 2000 psig (increasing) shown in the NTSP column of Function 7. Below this pressurizer pressure, the manual SI Block function prevents the SI Pressurizer Pressure - Low signal from occurring during normal plant shutdown/cooldown.

h. Function 1.e., Safety Injection - Steam Line Pressure- Low

1) The AV is revised from "≥ 500[(c)] psig" to "≥ 520[(c)] psig."

Licensing Report Sections: Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being implemented to support operation at EPU conditions. The steam line failure at hot zero power and full power analysis assumes that SI actuation would occur at an analytical limit of 395.3 psig. The AV is established by calculation to avoid exceeding the new analytical limit, taking all instrument uncertainties into account.

2) The Nominal Trip Setpoint is added as "530 psig."

Licensing Report Sections: Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP value is being added to conform to the "multiple column" format in Reference (3).

3) Note (b): The pressurizer pressure at which the Safety Injection - Steam Line Pressure - Low function applies in MODES 1, 2, and 3 is revised from "> 1800 psig" to "> 2000 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The change from "> 1800 psig" to "> 2000 psig" is being made to coincide with the change in the SI Block - Pressurizer Pressure function (Function 7 in Table 3.3.2-1) AV from "≤ 1800 psig" to "≤ 2005 psig." The previous 1800 psig value in Note (a) was based on the previous normal plant operating pressure of 1985 psig and a corresponding nominal SI Block setpoint of 1800 psig that automatically removed the manual bypass of the SI Steam Line Pressure - Low signal on increasing pressurizer pressure above 1800 psig during a normal plant heatup. The new 2000 psig value in Note (b) is based on the current (and EPU) normal plant operating pressure of 2235 psig and a revised SI unblock nominal setting of 2000 psig (increasing) shown in the NTSP column of Function 7. Below this pressurizer pressure, the manual SI Block function prevents the SI Steam Line Pressure - Low signal from occurring during normal plant shutdown/cooldown.

4) Note (c): The lead time constant (t_1) was revised from "≥ 12 seconds" to "≥ 18 seconds."

Licensing Report Sections: Section 2.8.5.1.2.2.2, Steam System Piping Failures Inside and Outside Containment.

Basis for the change: The lead dynamic compensation time constant (t_1) assumed in the Steam System Piping Failure at Hot Full Power safety analysis for Safety Injection on Steam Line Pressure - Low was revised from 12 seconds to 18 seconds to obtain acceptable results. This change is being implemented to allow operation at EPU conditions. The NTSP column for the Safety Injection - Steam Line Pressure – Low Function 1.e reflects the lead time constant field trip setpoint to satisfy this change in the AV.

i. Function 2.a., Containment Spray – Manual Initiation

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

j. Function 2.b., Containment Spray – Automatic Actuation Logic and Actuation Relays

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

k. Function 2.c., Containment Spray - Containment Pressure - High High

1) The AV is revised from "$\leq$ 30 psig" to "$\leq$ 28 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to replace an existing value that is not consistent with the AV calculation and presentation in this submittal. An analytical limit of 30 psig is established in the steam line break inside containment analysis and used in the calculation of AV, taking all instrument uncertainties into account. This value is applicable at current licensed power level and EPU conditions.

2) The Nominal Trip Setpoint is added as "25 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

l. Function 3.a., Containment Isolation – Manual Initiation

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

m. Function 3.b., Containment Isolation – Automatic Actuation Logic and Actuation Relays

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

n. Function 4.a., Steam Line Isolation – Manual Initiation

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

o. Function 4.b., Steam Line Isolation – Automatic Actuation Logic and Actuation Relays

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

p. Function 4.c., Steam Line Isolation - Containment Pressure - High High

1) The AV is revised from "$\leq$ 20 psig" to "$\leq$ 18 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to replace an existing value that is not consistent with the AV calculation and presentation in this submittal. The setpoint is not credited in any accident analysis, and a process limit of 20 psig has been established and is used in the calculation of AV, taking all instrument uncertainties into account. This value is applicable at current licensed power level and EPU conditions.

2) The Nominal Trip Setpoint is added as "15 psig."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

q. Function 4.d., Steam Line Isolation on High Steam Flow Coincident with Safety Injection and T_{avg} - Low, and Function 4.e., Steam Line Isolation on High High Steam Flow Coincident with Safety Injection.

1) The AV for Function 4.d., Steam Line Isolation on High Steam Flow Coincident with Safety Injection and T_{avg}- Low, was revised from "$\leq \Delta P$ corresponding to 0.66×10^6 lb/hr at 1005 psig" to "$\leq \Delta P$ corresponding to 0.8×10^6 lbm/hr at 1005 psig" and Coincident with Safety Injection and T_{avg} - Low was revised from "$\geq$ 540°F" to "$\geq$ 542°F."

The AV for Function 4.e., Steam Line Isolation on High High Steam Flow Coincident with Safety Injection, was revised from "$\leq \Delta P$ corresponding to 4×10^6 lb/hr at 806 psig" to "$\leq \Delta P$ corresponding to 4.9×10^6 lbm/hr at 586 psig."

Licensing Report Sections: Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the Change: This change is being implemented to support operation at EPU conditions. The Steam Line Break Mass and Energy Releases Outside of Containment analysis assumed that Steam Line Isolation on High Steam Flow Coincident with Safety Injection and T_{avg}- Low was initiated at a steam flow of 1.07×10^6 lb/hr (analytical limit). The Steam Line Break Mass and Energy Releases Outside of Containment and Steam line Break (core response) analyses established a process (and analytical) limit for the Steam Line Isolation on High-High Steam Flow Coincident with Safety Injection of 5.0×10^6 lb/hr, based on the upper range limit of the steam flow instrumentation span. The AV for these functions are calculated based on the analytical and process limits, taking all instrument uncertainties into account. The main steam line flow transmitters are currently calibrated for a range of 0 - 4.0 x 10^6 lb/hr, which is near the predicted EPU nominal steam flow of 3.7 x 10^6 lbm/hr at a feedwater temperature of 390°F. The main steam flow transmitters will be recalibrated for a range of 0 - 5.0 x 10^6 lbm/hr for the EPU, with the upper range limit being used as the process/analytical limit for the High High Steam Flow setpoint. This rescaling has been accounted for in the AV calculations.

2) The Nominal Trip Setpoints for Function 4.d., Steam Line Isolation on High Steam Flow Coincident with Safety Injection and T_{avg}- Low, are added as "Δp corresponding to 0.52 x 10^6 lb/hr at 1005 psig" and "543°F."

The Nominal Trip Setpoint for Function 4.e., Steam Line Isolation on High High Steam Flow Coincident with Safety Injection, is added as "Δp corresponding to 4.85 x 10^6 lb/hr at 586 psig."

Licensing Report Sections: Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the Change: The new Nominal Trip Setpoint column values are the field trip setpoints used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

r. Function 5.a., Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

s. Function 5.b., Feedwater Isolation - SG Water Level – High

1) The AV is revised from "NA" to "≤ 90%"

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being made to incorporate the AV determined by calculation. Because Feedwater Isolation – SG Water Level - High is not credited in any accident analysis, a process limit of 97% of instrument span (the maximum reliable indication of the SG narrow range level instrumentation) is used in the setpoint calculation, rather than an Analytical Limit, to determine an AV.

2) The Nominal Trip Setpoint is added as "78%."

Licensing Report Sections: Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

t. Function 6.a., Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays

The Nominal Trip Setpoint is added as "NA."

Licensing Report Sections: NA

Basis for the change: Functions for which there are no adjustable setpoints have "NA" in the NOMINAL TRIP SETPOINT column.

u. Function 6.b., Auxiliary Feedwater - SG Water Level - Low Low

1) The AV is revised from "≥ 20% of span" to "≥ 29.3%"

Licensing Report Sections: LR Section 2.8.5.2.3, Loss of Normal Feedwater Flow, LR Section 2.8.5.2.2, Loss of Non-Emergency AC Power to Station Auxiliaries, Section 2.4.1.2.3.2.2, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: This change is being implemented to support operation at EPU conditions. The loss of normal feedwater flow and the loss of non-emergency AC power to station auxiliaries assume that Auxiliary Feedwater is initiated at an analytical limit of 20% of span. The AV is established by calculation to avoid exceeding this new analytical limit, taking all instrument uncertainties into account. The phrase "of span" is being removed consistent with NUREG-1431 (Reference 3).

2) The Nominal Trip Setpoint is added as "30%."

Licensing Report Sections: LR Section 2.8.5.2.3, Loss of Normal Feedwater Flow, LR Section 2.8.5.2.2, Loss of Non-Emergency AC Power to Station Auxiliaries, Section 2.4.1.2.3.2.2, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

v. Function 6.d., Auxiliary Feedwater – Undervoltage Bus A01 and A02

The Nominal Trip Setpoint is added as "3255 V."

Licensing Report Sections: Starting Auxiliary Feedwater on bus A01 & A02 undervoltage is not credited in any accident analysis. The function provides a backup signal for AFW actuation in the Loss of AC Power event. LR Section 2.8.5.2.2, Loss of Non-Emergency AC Power to Station Auxiliaries.

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

w. Function 6.e., Auxiliary Feedwater – AFW Pump Suction Transfer on Suction Pressure Low

1) This new function is being added for EPU, as discussed in Section 2.1 of Enclosure 2 to Reference (2). The function includes new Condition J when one channel is inoperable.

Licensing Report Section: Section 2.5.4.5.2, Auxiliary Feedwater.

Basis for the change: The AFW system is being upgraded to increase the capability of the system. A description of the associated Technical Specification changes is provided in Reference (2). The AFW system upgrade adds a provision for automatic switchover of pump suction to service water on loss of the condensate storage tank (CST) suction source.

2) The new AV is added as "≥ 5.7 psig."

Licensing Report Section: Section 2.5.4.5.2, Auxiliary Feedwater, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The AV for new Function 6.e is determined by calculation based on an analytical limit of 5.241 psig as the minimum suction pressure at each individual AFW pump, with all instrument uncertainties taken into account.

3) The new Nominal Trip Setpoint is added as "6 psig."

Licensing Report Section: Section 2.5.4.5.2, Auxiliary Feedwater, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6)..

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in NUREG-1431 (Reference 3).

x. Old Function 7., Condensate Isolation

The Condensate Isolation function is being deleted from the Technical Specifications. Function 8., SI Block - Pressurizer Pressure, is renumbered as Function 7.

Licensing Report Sections: Section 2.6.1.2.4, Containment Response to Main Steam Line Break, and Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System.

Basis for the change: With the addition of the new Main Feedwater Isolation Valves, the function to isolate condensate upon a Containment Pressure High and the Automatic Actuation Logic and Actuation Relays are no longer credited in the accident analysis for Main Steam Line Break Inside Containment.

y. New Function 7., SI Block - Pressurizer Pressure

1) The AV is revised from "≤ 1800 psig" to "≤ 2005 psig."

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The AV change for the SI Block function from "≤ 1800 psig" to "≤ 2005 psig" is based on nominal plant operating pressure now being controlled at 2235 psig. At the previous nominal operating pressure of 1985 psig, a nominal SI Block setpoint of 1800 psig was necessary to automatically remove the manual bypass of both the SI Pressurizer Pressure - Low and SI Steam Line Pressure - Low actuation functions on increasing pressurizer pressure during a normal plant heatup/pressurization. With the current licensed power level and planned EPU nominal operating pressure of 2235 psig, the nominal SI Block setting can be increased to 2000 psig to provide adequate operating margin to manually block and automatically unblock SI actuation on either low pressurizer pressure or low steam line pressure during normal plant cooldown and heatup. The AV is calculated based on the nominal 2000 psig setpoint plus as-found tolerance and is valid at current licensed power level and EPU conditions. PBNP is not analyzed for and will not operate at 2000 psia RCS pressure at EPU conditions. An SI Block function at 1800 psig for 2000 psia operation is no longer applicable.

2) The Nominal Trip Setpoint is added as "2000 psig."

Licensing Report Sections: Section 1.1, Nuclear Steam Supply System Parameters, Section 2.4.1.2.3.2.2, Section 2.4.1.2.3.2.2, Engineered Safety Features Actuation System, and Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6).

Basis for the change: The new Nominal Trip Setpoint column value is the field trip setpoint used in calibration procedures for this function. The NTSP values are being added to conform to the "multiple column" format in Reference (3).

3.0 TECHNICAL EVALUATION

Source of Allowable Values in Revised TS Tables 3.3.1-1 and 3.3.2-1

The AV column in the attached TS RPS and ESFAS table markups (Enclosure 3) contains not-to-exceed values for adjustable setpoints that are derived from setpoint calculations. The AV establishes the limit for the as-found value of the adjustable bistable or relay setting when tested during the quarterly Channel Operational Test (COT) and 18-month Channel Calibration surveillances. For primary protective functions credited in the safety analyses, verifying that the as-found value is within the AV assures that the protective action will occur within the assumptions of the analyses, while accounting for all instrument uncertainties.

The AV fall into three categories, depending on the type of protective function provided by the adjustable bistable or relay setting.

1) For RPS/ESFAS functions that are credited in the accident analyses, the AV is based on an Analytical Limit taken directly from the safety analysis. Analytical Limits are established in safety analyses to protect Safety Limits, typically to prevent the breach of radioactive barriers, such as the fuel cladding, the reactor coolant pressure boundary, and containment. The analyses demonstrate that if the protective action occurs at or before the Analytical Limit is reached, the Safety Limit is protected.

The setpoint calculation starts at the Analytical Limit, and adds or subtracts the Total Loop Error (TLE) of the instrumentation, depending on the direction of the trip/actuation. The resulting value is called the Limiting Trip Setpoint (LTSP).

The AV is either the exact LTSP value, or may be the LTSP value after rounding in the conservative direction. By calculating the LTSP and AV in this manner, a bistable or relay setting found during surveillance testing to be on the conservative side of the AV assures that the trip function will protect the Analytical Limit, because the Analytical Limit is protected by at least the full uncertainty of the instrument loop.

If the bistable as-found setting during surveillance testing does not exceed the AV, then the channel is capable of protecting the Analytical Limit with all credible uncertainties, and the channel is OPERABLE. Conversely, if an as-found setting exceeds the AV then the channel is INOPERABLE.

2) For RPS/ESFAS functions that are not credited in the accident analyses, the AV may be based on a Process Limit.

   For a backup or anticipatory trip, a Process Limit may be chosen as the span limit in the direction of the trip, to assure that the trip will occur within the instrument span. Similar to primary trips, the setpoint calculation starts at the Process Limit, and then adds or subtracts the TLE of the instrumentation, depending on the direction of the trip/actuation, to determine the Limiting Trip Setpoint (LTSP). The Limiting Trip Setpoint is then converted to the AV, either directly or by rounding in the conservative direction.

3) For RPS/ESFAS operating bypasses (also called permissives or blocks) that are not specifically credited in the accident analyses, the AV may be based on the Nominal Trip Setpoint and its as-found tolerance.

   Because operating bypasses typically lack either an Analytical Limit or a Process Limit, the AV is determined by adding the as-found tolerance to the Nominal Trip Setpoint (field trip setpoint) in the direction of interest. The direction of interest for an operating bypass is defined by IEEE 279-1968 as the direction that requires the operating bypass to be automatically removed to restore the protective function(s) that are bypassed. Removing the bypass (unblocking) is the safety function of the operating bypass.

<u>Source of Nominal Trip Setpoint Values in Revised TS Tables 3.3.1-1 and 3.3.2-1</u>

The new Nominal Trip Setpoint (NTSP) column in Tables 3.3.1-1 and 3.3.2-1 is added to conform to the "multiple column" format of NUREG-1431 (Reference 3). The values in this column represent nominal field trip setpoints for the adjustable RPS/ESFAS setpoint functions.

The NTSP values are taken from plant calibration procedures that are used to satisfy the RPS/ESFAS surveillance requirements for 92-day Channel Operational Tests and 18-month Channel Calibrations. The NTSP values are included in the setpoint calculations that determine

AV, because one of the purposes of each setpoint calculation is to demonstrate that the NTSP does not exceed the Limiting Trip Setpoint/AV. To show this, the calculation determines that positive margin exists between the NTSP and the LTSP. With margin between these values, the calibration procedure assures that even if full instrument uncertainty is applied to the NTSP, there will be margin before the Analytical Limit or Process Limit is approached.

Application of New Notes regarding as-left and as-found tolerances

The two new notes added to TS Tables 3.3.1-1 and 3.3.2-1 apply to the results of the 92-day Channel Operational Test (COT) and the 18-month Channel Calibration surveillances defined in the Technical Specifications. The notes require actions based on the as-found channel setpoint value recorded during the surveillance relative to the as-left tolerance band and the as-found tolerance band that are established in the calibration procedures.

The notes generally apply to any adjustable setpoint for which an as-found value is recorded during the surveillance. An exception to this statement is that the notes do not apply to operating bypasses, such as permissives and blocks. This is based on guidance from joint NRC and industry efforts to clarify the application of setpoint methodology for LSSS functions.

Enclosure 6, Appendix E, Supplement to LR Section 2.4.1 (Enclosure 6) describes the actions that will be taken to evaluate a channel for operability based on the as-found setpoint value, as required by the new notes.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

PBNP was designed and constructed to comply with the intent of the draft AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967 (ML003674718). PBNP was licensed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50. As such PBNP was not licensed to 10 CFR 50, Appendix A.

The origin of the PBNP General Design Criteria (GDC) relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.3.

The PBNP GDC comparable to Appendix A GDC 13, Instrumentation and Control, and 20, Protection System Functions, are PBNP GDC 12 and GDC 20, respectively. Therefore, the applicable regulatory requirements are:

- PBNP GDC 12, "Instrumentation and Control Systems," states,

  "Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential to reactor facility operating variables."

- PBNP GDC 20, "Protection System Redundancy and Independence," states,

  "Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection function to be served."

- 10 CFR 50.36(c)(1)(ii)(A) states:

  "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor."

The proposed change clarifies the Technical Specification requirements to ensure that the automatic protection action will correct the abnormal situation before a safety limit is exceeded. The proposed change also revises the Technical Specifications to enhance the controls used to maintain the variables and systems within the prescribed operating ranges, in order to ensure that automatic protection actions occur as necessary to initiate the operation of systems and components important to safety as assumed in the accident analysis.

4.2 Significant Hazards Consideration

NextEra has evaluated whether or not a significant hazard is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below. The No Significant Hazards Consideration evaluations contained in Reference (1) for the proposed Technical Specification RPS and ESFAS Instrumentation Tables 3.3.1-1 and 3.3.2-1 contained in Reference (2) for the addition of the ESFAS Function 6.e for AFW Pump Suction Transfer on Suction Pressure-Low, and contained in Reference (4) are still applicable. The evaluation below addresses the revised proposed column headings, the two-column format with the addition of Nominal Trip Setpoint values, and the added notes applicable to COT and Channel Calibration Surveillances.

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

   Response: No.

   The proposed changes to the TS will ensure that the results of previously evaluated accidents at the uprated conditions remain within the acceptance criteria.

Changes to the TS are being proposed, including changes to the RPS and ESFAS AV, the addition of a new column in TS Tables 3.3.1-1 (RPS Instrumentation) and 3.3.2-1 (ESFAS Instrumentation) labeled "Nominal Trip Setpoint," and the addition of two new Notes that govern surveillance actions based on the as-found setting of adjustable setpoints during routine surveillance testing. Of these changes, the AV changes are the only changes linked directly to the plant safety analyses Analytical Limits. The safety analyses demonstrate that the applicable acceptance criteria are met at the uprated power conditions, considering the proposed TS changes.

The proposed RPS and ESFAS setpoint changes provide appropriate values for operation of PBNP at EPU conditions. The revised TS AV have been calculated to account for new EPU analytical limits, instrument uncertainties and drift. The proposed RPS and ESFAS setpoint changes are considered in the safety analyses for the affected RPS and ESFAS functions, and do not significantly increase the probability or consequences of the accidents previously evaluated and the setpoint changes considered in the safety analysis continue to meet the applicable acceptance criteria. The safety analyses for these accidents have been performed at the EPU power level and provide acceptable results.

The proposed changes will ensure that the instruments actuate as assumed to mitigate accidents previously evaluated. The proposed changes will not significantly affect accident initiators or precursors and will not alter or prevent the ability of systems, structures or components from performing the intended safety function to meet the applicable acceptance limits for the accidents and events.

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

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

Response: No.

The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis but ensures that the instruments behave as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

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

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

Response: No.

The proposed change clarifies the requirements for instrumentation to ensure the instrumentation will activate as assumed in the accident analysis. No change is made to the accident analysis assumptions

The proposed changes to TS provide adequate margin such that PBNP Units 1 and 2 can be operated in a safe manner at the EPU conditions. No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed changes. All systems, structures and components previously assumed for the mitigation of an event remain capable of fulfilling their intended design function. The proposed changes will not have any significant effect on the margin to safety.

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

4.3 Conclusions

Based on the above, the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly a finding of "no significant hazard consideration" is justified. The PBNP Plant Operations Review Committee has reviewed this change and concurs with this conclusion.

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

5.0 ENVIRONMENTAL CONSIDERATION

NextEra has evaluated the proposed changes and has concluded that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

(1.) FPL Energy Point Beach, LLC, Letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)

(2.) NextEra Energy Point Beach, LLC, Letter to NRC, dated June 17, 2009, License Amendment Request 261 Supplement 1, Extended Power Uprate (ML091690090)

(3.) NUREG-1431, Revision 3, dated June 30, 2004, Standard Technical Specifications Westinghouse Plants (ML041830612 and ML041830205)

(4.) NextEra Energy Point Beach, LLC, Letter to NRC, dated June 17, June 2009, License Amendment Request 261 Supplement 2, Extended Power Uprate (ML091690087)

ENCLOSURE 3

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261 SUPPLEMENT 3
EXTENDED POWER UPRATE

PROPOSED TECHNICAL SPECIFICATION CHANGES

13 pages follow

Table 3.3.1-1 (page 1 of ~~8~~ 9)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7(m) SR 3.3.1.11(m)	≤ ~~108%~~ 109% RTP	107% RTP
b. Low	1(b),2	4	D	SR 3.3.1.1 SR 3.3.1.8(m) SR 3.3.1.11(m)	≤ ~~25%~~ 28% RTP	20% RTP
3. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8(m) SR 3.3.1.11(m)	≤ ~~40%~~ 43% RTP	25% RTP
4. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8(m) SR 3.3.1.11(m)	~~within span of instrumentation~~ ≤ 4.0 E5 cps	2.0 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	~~within span of instrumentation~~ ≤ 4.0 E5 cps	2.0 E5 cps
5. Overtemperature ΔT	1,2	4	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7(m) SR 3.3.1.11(m)	Refer to Note 1 (Page 3.3.1-18)	Refer to Note 1 (Page 3.3.1-18)
6. Overpower ΔT	1,2	4	D	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	Refer to Note 2 (Page 3.3.1-20)	Refer to Note 2 (Page 3.3.1-20)

(continued)

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

(b) Below the P-10 (Power Range Neutron Flux) interlock~~s~~.

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

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

(m) Table 3.3.1-1 Notes 3 and 4 are applicable

Table 3.3.1-1 (page 2 of ~~8~~ 9)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
7.	Pressurizer Pressure						
	a. Low	$1^{(e)}$	4	K	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	~~(h)~~ ≥ 1860 psig	1925 psig
	b. High	1,2	3	D	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	~~(i)~~ ≤ 2385 psig	2365 psig
8.	Pressurizer Water Level — High	$1^{(e)}$	3	K	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	≤ ~~95% of span~~ 85%	80%
9.	Reactor Coolant Flow-Low						
	a. Single Loop	$1^{(f)}$	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	≥ 90%	93%
	b. Two Loops	$1^{(g)}$	3 per loop	K	SR 3.3.1.1 SR 3.3.1.7(m) SR 3.3.1.11(m)	≥ 90%	93%
10.	Reactor Coolant Pump (RCP) Breaker Position						
	a. Single Loop	$1^{(f)}$	1 per RCP	M	SR 3.3.1.13	NA	NA
	b. Two Loops	$1^{(g)}$	1 per RCP	N	SR 3.3.1.13	NA	NA
11.	Undervoltage Bus A01 & A02	$1^{(e)}$	2 per bus	K	SR 3.3.1.9 SR 3.3.1.10(m)	≥ 3120 V	3170 V

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

(g) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

~~(h) ≥ 1905 psig during operation at 2250 psia, or ≥ 1800 psig during operation at 2000 psia.~~

~~(i) ≤ 2385 psig during operation at 2250 psia, or ≤ 2210 psig during operation at 2000 psia.~~

(m) Table 3.3.1-1 Notes 3 and 4 are applicable

Table 3.3.1-1 (page 3 of ~~8~~ 9)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
12.	Underfrequency Bus A01 & A02	1[e]	2 per bus	E	SR 3.3.1.10[m]	≥ 55.0 Hz	57 Hz
13.	Steam Generator (SG) Water Level — Low Low	1,2	3 per SG	D	SR 3.3.1.1 SR 3.3.1.7[m] SR 3.3.1.11[m]	≥ ~~20% of span~~ 29.3%	30%
14.	SG Water Level — Low	1,2	2 per SG	D	SR 3.3.1.1 SR 3.3.1.7[m] SR 3.3.1.11[m]	~~NA~~ ≥ 10%	30%
	Coincident with Steam Flow/Feedwater Flow Mismatch	1,2	2 per SG	D	SR 3.3.1.1 SR 3.3.1.7[m] SR 3.3.1.11[m]	≤ 1 E6 lb/hr	0.8 E6 lb/hr
15.	Turbine Trip						
	a. Low Autostop Oil Pressure	1[j]	3	O	SR 3.3.1.14	NA	NA
	b. Turbine Stop Valve Closure	1[j]	2	O	SR 3.3.1.14	NA	NA
16.	Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	P	SR 3.3.1.13	NA	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(j) Above the P-9 (Power Range Neutron Flux) interlock.

(m) Table 3.3.1-1 Notes 3 and 4 are applicable

Table 3.3.1-1 (page 4 of ~~8~~ 9)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	~~> 1E-10~~ amp ≥ 4E-11	1E-10 amp
b. Low Power Reactor Trips Block, P-7						
(1) Power Range Neutron Flux	1	4	S	SR 3.3.1.11 SR 3.3.1.12	~~< 10%~~ RTP ≤ 13%	10% RTP
(2) Turbine Impulse Pressure	1	2	S	SR 3.3.1.11 SR 3.3.1.12	~~< 10%~~ turbine ≤ 12.8% power	10% turbine power
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	~~< 50%~~ RTP ≤ 38%	35% RTP
d. Power Range Neutron Flux, P-9	1(k)	4	S	SR 3.3.1.11 SR 3.3.1.12	~~< 50% RTP~~ (h)	(i)
e. Power Range Neutron Flux, P-10	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	~~> 8% RTP and < 10% RTP~~ ≥ 6% RTP and ≤ 12% RTP	9% RTP
18. Reactor Trip Breakers (RTBs)	1,2	2 trains	Q	SR 3.3.1.4	NA	NA
	3(a), 4(a), 5(a)	2 trains	T	SR 3.3.1.4	NA	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	U	SR 3.3.1.4	NA	NA
	3(a), 4(a), 5(a)	1 each per RTB	T	SR 3.3.1.4	NA	NA

(continued)

(a) With the RTBs closed and the Rod Control System capable of rod withdrawal.

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

(h) ≤ 38% RTP for full design power T_{avg} < 572°F or ≤ 53% RTP for full design power T_{avg} ≥ 572°F. For EOC coastdown, P-9 is not reset if T_{avg} decreases to < 572°F.

(i) 35% RTP for full design power T_{avg} < 572°F or 50% RTP for full design power T_{avg} ≥ 572°F. For EOC coastdown, P-9 is not reset if T_{avg} decreases to < 572°F.

(k) With 1 of 2 circulating water pump breakers closed and condenser vacuum ≥ 22 "Hg.

Table 3.3.1-1 (page 5 of ~~8~~ 9)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
20.	Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism	1(l), 2(l)	1	V	SR 3.3.1.4	NA	NA
		3(l), 4(l), 5(l)	1	W	SR 3.3.1.4	NA	NA
21.	Automatic Trip Logic	1, 2,	2 trains	P	SR 3.3.1.5 SR 3.3.1.15	NA	NA
		3(a), 4(a), 5(a)	2 trains	X	SR 3.3.1.5	NA	NA

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

(l) When Reactor Trip Bypass Breakers are racked in and closed and the Rod Control System is capable of rod withdrawal.

Table 3.3.1-1 (page 6 of ~~8~~ 9)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

$$\Delta T\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o\left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P - P') - f(\Delta I)\right)$$

Where:~~where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)~~

ΔT_o	=	indicated ΔT at ~~rated power~~ RTP, °F
T	=	average temperature, °F
T'	≤	[*]°F ~~(for cores containing 422V+ fuel assemblies)~~
~~T'~~	~~≤~~	~~[*]°F (for cores not containing 422V+ fuel assemblies)~~
P	=	pressurizer pressure, psig
P'	=	[*] psig ~~(for 2250 psia operation)~~
~~P'~~	~~=~~	~~[*] psig (for 2000 psia operation and cores not containing 422V+ fuel assemblies)~~
K_1	≤	[*] ~~(for 2250 psia operation and cores containing 422V+ fuel assemblies)~~
~~K_1~~	~~≤~~	~~[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)~~
~~K_1~~	~~≤~~	~~[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)~~
K_2	=	[*] ~~(for 2250 psia operation and cores containing 422V+ fuel assemblies)~~
~~K_2~~	~~=~~	~~[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)~~
~~K_2~~	~~=~~	~~[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)~~
K_3	=	[*] ~~(for 2250 psia operation and cores containing 422V+ fuel assemblies)~~
~~K_3~~	~~=~~	~~[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)~~
~~K_3~~	~~=~~	~~[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)~~
τ_1	=	[*] sec
τ_2	=	[*] sec
τ_3	=	[*] sec ~~for Rosemont or equivalent RTD~~
	~~=~~	~~[*] sec for Sostman or equivalent RTD~~
τ_4	=	[*] sec ~~for Rosemont or equivalent RTD~~
	~~=~~	~~[*] sec for Sostman or equivalent RTD~~

~~and f(ΔI) is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:~~

~~(a) for $q_t - q_b$ within –[*], +[*] percent, f(ΔI) = 0 for cores not containing 422V+ fuel assemblies; for $q_t - q_b$ within –[*], +[*] percent, f(ΔI) = 0 for cores containing 422V+ fuel assemblies.~~

~~(b) for each percent that the magnitude of $q_t - q_b$ exceeds +[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power for cores not containing 422V+ fuel assemblies and reduced by an equivalent of [*] percent of rated power for cores containing 422V+ fuel assemblies.~~

Table 3.3.1-1 (page 7 of ~~8~~ 9)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT (continued)

~~(c) for cores not containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds [*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds [*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power.~~

$f(\Delta I) \equiv$	[*] {[*] – ($q_t - q_b$)}	when ($q_t - q_b$) ≤ [*]% RTP
	0% of RTP	when [*]% RTP < ($q_t - q_b$) ≤ [*]% RTP
	[*] {($q_t - q_b$) – [*]}	when ($q_t - q_b$) > [*]% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and ($q_t + q_b$) is the total THERMAL POWER in percent RTP.

* The values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (page 8 of ~~8~~ 9)
Reactor Protection System Instrumentation

Note 2: Overpower ΔT

$$\Delta T\,(\frac{1}{1+\tau_3 S}) \leq \Delta T_o[K_4 - K_5(\frac{\tau_5 S}{\tau_5 S+1})(\frac{1}{1+\tau_4 S})T - K_6[T(\frac{1}{1+\tau_4 S}) - T']]$$

Where: ~~where (values are applicable to operation at both 2000 psia and 2250 psia)~~

ΔT_o	=	indicated ΔT at ~~rated power~~ RTP, °F
T	=	average temperature, °F
T'	≤	[*]°F ~~(for cores containing 422V+ fuel assemblies)~~
~~T'~~	~~≤~~	~~[*]°F (for cores not containing 422V+ fuel assemblies)~~
K_4	≤	[*] ~~of rated power (for cores containing 422V+ fuel assemblies)~~
~~K_4~~	~~≤~~	~~[*] of rated power (for cores not containing 422V+ fuel assemblies)~~
K_5	=	[*] for increasing T
	=	[*] for decreasing T
K_6	=	[*] for T ≥ T' ~~(for cores containing 422V+ fuel assemblies)~~
~~K_6~~	~~=~~	~~[*] for T ≥ T' (for cores not containing 422V+ fuel assemblies)~~
	=	[*] for T < T'
τ_5	=	[*] sec
τ_3	=	[*] sec ~~for Rosemont or equivalent RTD~~ ~~[*] sec for Sostman or equivalent RTD~~
τ_4	=	[*] sec ~~for Rosemont or equivalent RTD~~ ~~[*] sec for Sostman or equivalent RTD~~

* The values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (page 9 of 9)
Reactor Protection System Instrumentation

Note 3:

If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Note 4:

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in FSAR Section 7.2.

Table 3.3.2-1 (page 1 of ~~3~~ 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	NA	NA
c. Containment Pressure—High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≤ 6 psig~~ ≤ 5.3 psig	5 psig
d. Pressurizer Pressure—Low	1,2,3(a)	3	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≥ 1715 psig~~ ≥ 1725 psig	1735 psig
e. Steam Line Pressure—Low	1,2,3(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≥ 500(c) psig~~ ≥ 520(c) psig	530 psig
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2	E	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	NA	NA
c. Containment Pressure—High High	1,2,3	2 sets of 3	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≤ 30 psig~~ ≤ 28 psig	25 psig

(continued)

(a) Pressurizer Pressure > ~~1800~~ 2000 psig.

(b) Pressurizer Pressure > ~~1800~~ 2000 psig, except during Reactor Coolant System hydrostatic testing.

(c) Time constants used in the lead/lag controller are $t_1 \geq$ ~~12~~ 18 seconds and $t_2 \leq 2$ seconds.

(f) Table 3.3.2-1 Notes 1 and 2 are applicable

Table 3.3.2-1 (page 2 of ~~3~~ 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Containment Isolation						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.4 SR 3.3.2.5	NA	NA
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements, except Manual SI initiation.					
4. Steam Line Isolation						
a. Manual Initiation	1,2(d),3(d)	1/loop	F	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(d),3(d)	2 trains	G	SR 3.3.2.2 SR 3.3.2.5	NA	NA
c. Containment Pressure—High High	1,2(d),3(d)	3	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≤ 20 psig~~ ≤ 18 psig	15 psig
d. High Steam Flow	1,2(d),3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	≤ Δp corresponding to ~~0.66~~ 0.8 x 10^6 lb/hr at 1005 psig	Δp corresponding to 0.52 x 10^6 lb/hr at 1005 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
and						
Coincident with T_{avg}—Low	1,2(d),3(d)	3	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≥ 540°F~~ ≥ 542°F	543°F
e. High High Steam Flow	1,2(d),3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	≤ Δp corresponding to 4.9 x 10^6 lb/hr at ~~806~~ 586 psig	Δp corresponding to 4.85 x 10^6 lb/hr at 586 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(d) Except when all MSIVs are closed and de-activated.

(f) Table 3.3.2-1 Notes 1 and 2 are applicable

Table 3.3.2-1 (page 3 of ~~3~~ 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2(e),3(e)	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5	NA	NA
b. SG Water Level—High	1,2(e),3(e)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~NA~~ ≤ 90%	78%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2	NA	NA
b. SG Water Level— Low Low	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	~~≥ 20%~~ ≥ 29.3%	30%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Undervoltage Bus A01 and A02	1,2	2 per bus	H	SR 3.3.2.6 SR 3.3.2.8(f)	≥ 3120 V	3255 V
e. AFW Pump Suction Transfer on Suction Pressure Low	1,2,3	1 per pump	J	SR 3.3.2.1 SR 3.3.2.3(f) SR 3.3.2.8(f)	≥ 5.7 psig	6 psig
~~7. Condensate Isolation~~						
~~a. Containment Pressure—High~~	~~1,2(e),3(e)~~	~~3~~	~~D~~	~~SR 3.3.2.1~~ ~~SR 3.3.2.3~~ ~~SR 3.3.2.8~~	~~≤ 6 psig~~	
~~b. Automatic Actuation Logic and Actuation Relays~~	~~1,2(e),3(e)~~	~~2 trains~~	~~G~~	~~SR 3.3.2.2~~ ~~SR 3.3.2.4~~ ~~SR 3.3.2.5~~	~~N/A~~	
~~8.~~7. SI Block - Pressurizer Pressure	1,2,3	3	I	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.8	~~≤ 1800 psig~~ ≤ 2005 psig	2000 psig

(e) Except when all MFIVs, MFRVs and associated bypass valves are closed and de-activated.

(f) Table 3.3.2-1 Notes 1 and 2 are applicable

Table 3.3.2-1 (page 4 of 4)
Engineered Safety Feature Actuation System Instrumentation

Note 1:

If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Note 2:

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in FSAR Section 7.2.

ENCLOSURE 4

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261 SUPPLEMENT 3
EXTENDED POWER UPRATE

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

(FOR INFORMATION ONLY)

83 pages follow

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during Anticipated Operational Occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as specifying LCO's on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulations as "...settings for automatic protective devices..., so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs before or upon reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative that the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

~~The LSSS, defined in this specification as the Allowable Value Setpoints, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).~~

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.1-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g. calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors

BASES

BACKGROUND (continued)

which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS.

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in the Technical Specifications as "...being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the 'as-found' value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel to the as-left tolerance around the NTSP to account for further drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence.

BACKGROUND (continued)

Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS instrumentation is segmented into four distinct but interconnected modules as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including Analog Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system ~~devices~~ channels, and control board/control room/miscellaneous indications;

3. Relay Logic System, including input, logic, and output devices: initiates proper unit shutdown in accordance with the defined logic, which is based on bistable, setpoint comparators, or contact outputs from the signal process control and protection systems; and

4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the NTSP and Allowable Value~~s~~. The OPERABILITY of each transmitter or sensor ~~can be evaluated when its~~ is determined by either "as found" calibration data ~~are compared against its documented acceptance criteria.~~ evaluated during the CHANNEL CALIBRATION or by qualitative assessment of the field transmitter or sensor as related to the channel behavior observed during performance of the CHANNEL CHECK.

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BACKGROUND (continued)

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with ~~setpoints~~ NTSPs derived from Analytical Limits (ALs) established by safety analyses. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the relay logic system and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1968 (Ref. 3). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RPS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

Allowable Values and Nominal Trip Setpoints

The trip setpoints used in the bistables are based on analytical limits established in the safety analyses. The selection of the Nominal Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must

BASES

BACKGROUND (continued)

function in harsh environments as defined by 10 CFR 50.49 (Ref. 4), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservative~~ly adjusted~~ with respect to the analytical limits. A ~~detailed~~ description of the methodology used to calculate the ~~Trip Setpoints, including their explicit uncertainties, is provided in DGI-01, "Instrument Setpoint Methodology" (Ref. 5).~~ Allowable Values, NTSPs, and as-left and as-found tolerance bands, is provided in FSAR Chapter 7 (Reference 1). The magnitudes of the uncertainties are factored into the determination of each NTSP and corresponding Allowable Value in design basis calculations. The ~~actual nominal T~~trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for ~~changes in random~~ measurement errors detectable by a COT.~~One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.~~

The NTSP is the value at which the bistable is set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on the stated channel uncertainties. Any bistable is considered to be properly adjusted when the as-left NTSP value is within the as-left tolerance band for CHANNEL CALIBRATION uncertainty allowance (i.e. ± rack calibration and comparator setting uncertainties). The NTSP value is therefore considered a "nominal" value (i.e., expressed as a value without inequalities) for the purposes of COT and CHANNEL CALIBRATION.

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the operability of this instrument must be verified based on the field trip setpoint and not the NTSP.

~~Setpoints in accordance with~~ Nominal Trip Setpoints, in conjunction with the use of as-found and as-left tolerances, together with the requirements of the Allowable Value, ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). ~~Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.~~

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of

BACKGROUND (continued)

service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

~~The Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 5, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Allowable Value. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.~~

Relay Logic System

The Relay Logic System equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of Relay Logic System, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip for the unit. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The Relay Logic System performs the decision logic for actuating a reactor trip, generates the electrical output signal that will initiate the required trip, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the Relay Logic System equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

BASES

BACKGROUND (continued)

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. During normal operation the output from the relay logic system is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the relay logic system output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each RTB is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the relay logic system. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

The RPS functions to preserve ~~maintain~~ the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis described in Reference 2 takes credit for most RPS trip Functions. RPS trip Functions that are retained yet not specifically credited in the accident analysis are implicitly ~~qualitatively~~ credited in the safety analysis and the NRC staff-approved licensing basis for the unit. These RPS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RPS trip Functions that were credited in the accident analysis.

Permissive and interlock setpoints allow the blocking of trips during plant start-ups and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventative or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values with regard to measurement accuracy.

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

The LCO requires all instrumentation performing an RPS Function, listed in Table 3.3.1-1 ~~in the accompanying LCO,~~ to be OPERABLE. The Allowable Value specified in Table 3.3.1-1 is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST (COT). As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the channel (NTSP) will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond that expected during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

If the actual setting of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE but degraded. The degraded condition will be further evaluated during performance of the SR. If the channel is functioning as required and is expected to pass the next surveillance, then the channel can be restored to service at the completion of the surveillance. If the evaluation determines that the channel is not performing as expected, the channel operability status cannot be verified. Therefore, it is inoperable because it may not perform its protective function(s) if needed before the next surveillance test.

If the channel setpoint cannot be restored to the NTSP as-left tolerance, or if the actual setting of the channel is found to be non-conservative with respect to the Allowable Value, the channel is inoperable. For these conditions, after the surveillance is completed, the channel's as-found setting will be entered into the Corrective Action Program for further evaluation.

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the operability of the channel must be verified based on the field trip setpoint and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, one channel of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are generally required when one RPS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RPS action. In this case, the RPS will still provide protection, even with random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RPS trip and disable one RPS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Protection System Functions

The safety analyses and OPERABILITY requirements applicable to each RPS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using one of four reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Allowable Value.

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel consists of two reactor trip switches (one in each train). Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE with the RTBs closed and the Rod Control System capable of rod withdrawal. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the Rod Control System

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

is not capable of withdrawing the shutdown rods or control rods. If the rods cannot be withdrawn from the core or all of the rods are inserted, there is no need to be able to trip the reactor. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires all four of the Power Range Neutron Flux-High channels to be OPERABLE.

In MODE 1 or 2, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux - High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RPS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux-Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux-High Setpoint trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

4. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup.

This trip Function provides redundant protection to the Power Range Neutron Flux-Low trip Function. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RPS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two operable channels are sufficient to ensure no single random failure will disable this trip function.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical and control rod ejection events.

In MODE 2 when below the P-6 setpoint, and in MODES 3, 4 and 5 when there is a potential for an uncontrolled RCAA bank rod withdrawal accident, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low Setpoint

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are de-energized.

In MODES 3, 4 and 5 with the Rod Control System not capable of rod withdrawal, and in MODE 6, this Function is not required to be OPERABLE. The requirements for the NIS source range detectors to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution are addressed in LCO 3.9.2, "Nuclear Instrumentation," for MODE 6.

5. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature-the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;

- pressurizer pressure-the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution — f(ΔI), the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

The Overtemperature ΔT trip Function is calculated for each channel as described in Note 1 of Table 3.3.1-1. Reactor Trip occurs if Overtemperature ΔT is indicated in two channels. Because the pressure and temperature signals are used for other control

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RPS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

6. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature.

The Overpower ΔT trip Function is calculated for each channel as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two channels. The temperature signals are used for other control functions. The actuation logic must be able to withstand an input failure to the control system, which may then

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)	

require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RPS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

7. <u>Pressurizer Pressure</u>

The same sensors provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. <u>Pressurizer Pressure-Low</u>

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure-Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent). On decreasing

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

power, this trip Function is automatically blocked below P-7. Below the P-7 interlock, no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires three channels of the Pressurizer Pressure-High to be OPERABLE.

~~For operation at 2250 psia~~The Pressurizer Pressure-High ~~LSSS~~NTSP is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

~~For operation at 2000 psia, a 50% load rejection with steam dump results in a peak pressure below the Pressurizer Pressure-High LSSS. Therefore, even though the PORV setting is above the reactor trip, the transient will not result in PORV actuation or a reactor trip on high Pressurizer Pressure.~~

In MODE 1 or 2, the Pressurizer Pressure-High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

8. Pressurizer Water Level—High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 interlock, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

9. Reactor Coolant Flow-Low

a. Reactor Coolant Flow-Low (Single Loop)

The Reactor Coolant Flow—Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately ~~50~~35% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE in MODE 1 above P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint, a loss of flow in two loops is required to actuate a reactor trip (Function 9.b) because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Above the P-7 interlock and below the P-8 setpoint, a loss of flow in two loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 interlock and below the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip must be OPERABLE. Below the P-7 interlock, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the reactor trip on low flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

10. Reactor Coolant Pump (RCP) Breaker Position

Both RCP Breaker Position trip Functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These Functions anticipate the Reactor Coolant Flow-Low trips to avoid RCS heatup that would occur before the low flow trip actuates.

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Single Loop) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. A channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix Relay. One OPERABLE channel is sufficient for this trip Function because the RCS Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Pump Breaker Position (Two Loops)

The RCP Breaker Position (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two RCS loops. The position of each RCP breaker is monitored. Above the P-7 interlock and below the P-8 setpoint, a loss of flow in two loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. A channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix Relay. One OPERABLE channel is sufficient for this Function because the RCS Flow—Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 interlock and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip must be OPERABLE. Below the P-7 interlock, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the reactor trip on loss of flow in two RCS loops is automatically enabled. Above the

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

11. Undervoltage Bus A01 and A02

The Undervoltage Bus A01 and A02 reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops. The voltage to Bus A01 and A02 is monitored. Above the P-7 interlock, a loss of voltage detected on both buses will initiate a reactor trip. This trip Function will generate a reactor trip independent of Reactor Coolant Flow—Low (Two Loops) Trip Setpoint. Time delays are incorporated into the Undervoltage Bus A01 and A02 channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Undervoltage channels per bus to be OPERABLE. An Undervoltage channel consists of the A01/A02 Bus Undervoltage Relay and the associated Bus Undervoltage Matrix Relay.

In MODE 1 above the P-7 interlock, the Undervoltage Bus A01 and A02 trip must be OPERABLE. Below the P-7 interlock, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the reactor trip on loss of flow in both RCS loops is automatically enabled.

12. Underfrequency Bus A01 and A02

The Underfrequency Bus A01 and A02 RCP breaker trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 interlock, a loss of frequency detected on two RCP buses will trip both RCP breakers. Tripping both RCP breakers will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency Bus A01 and A02 channels to prevent reactor trips due to momentary electrical power transients.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

The LCO requires two Underfrequency Bus A01 channels and two Underfrequency Bus A02 channels to be OPERABLE.

In MODE 1 above the P-7 interlock, the Underfrequency Bus A01 and A02 RCP breaker trip must be OPERABLE. Below the P-7 interlock, this trip and all reactor trips on loss of flow are automatically blocked, because no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the Underfrequency Bus A01 and A02 RCP breaker trip is automatically enabled.

13. Steam Generator Water Level—Low Low

The SG Water Level—Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires three channels of SG Water Level—Low Low per SG to be OPERABLE.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level—Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level—Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

14. Steam Generator Water Level—Low, Coincident With Steam Flow/Feedwater Flow Mismatch

SG Water Level-Low, in conjunction with the Steam Flow/Feedwater Flow Mismatch, ensures that protection is provided against a loss of heat sink. In addition to a decreasing water level in the SG, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow.

With less feedwater flow than steam flow, SG level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There are two SG level channels and two Steam Flow/Feedwater Flow Mismatch channels per SG. One narrow range level channel sensing a low level coincident with one Steam Flow/ Feedwater Flow Mismatch channel sensing flow mismatch (steam flow greater than feed flow) will actuate a reactor trip.

~~Table 3.3.1-1 identifies the Technical Specification Allowable Value for this trip function as not applicable (NA), because~~LCO 3.3.1, Function 13, Steam Generator Water Level-Low Low, is used to bound the analysis for a loss of feedwater event. The nominal setting required for the Steam Generator Water Level-Low trip function is 30% of span. ~~This nominal setting was developed outside of the setpoint methodology and has been provided by the NSSS supplier.~~

The LCO requires two channels of SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch per SG.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch trip must be OPERABLE. The normal source of water for the SGs is the MFW System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the RHR System in MODE 4, 5, or 6. The MFW System is in operation only in MODE 1 or 2 and, therefore, this trip Function need only be OPERABLE in these MODES.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

15. Turbine Trip

a. Turbine Trip-Low Autostop Oil Pressure

The Turbine Trip-Low Autostop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, ~~(~~approximately 50% power (35% power with full design power Tavg < 572°F), with at least one circulating water pump breaker closed, and condenser vacuum not high, will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

Table 3.3.1-1 identifies the Technical Specification NTSP and Allowable Value for this trip function as not applicable (NA). No Analytical Value is assumed in the accident analysis for this function. The nominal setting required for the Turbine Trip – Low Autostop Oil Pressure trip function is 45 psig. This nominal setting was developed outside of the setpoint methodology and has been provided by the NSSS supplier.

The LCO requires three channels of Turbine Trip-Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip-Low Autostop Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. Any turbine trip with from a power level below the P-9 setpoint, approximately 50% power (35% power for full design power Tavg <572°F), with at least one circulating water pump breaker closed, and condenser

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

vacuum not high, will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Autostop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RPS. If both limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

No analytical value is assumed in the accident analyses for this function. The LCO requires two Turbine Trip-Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. Both channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

16. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RPS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

~~Allowable Values are not applicable to~~There are no setpoints associated with this Function. The SI Input is provided by relay in the ESFAS. Therefore, Table 3.3.1-1 identifies the Technical Specification Nominal Trip Setpoint and Allowable Value for this trip function as not applicable (NA).~~Therefore, there is no measurement signal with which to associate an LSSS.~~

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

17. Reactor Protection System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either Power Range Neutron Flux or Turbine Impulse Pressure. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Open (Two Loops);
- Undervoltage Bus A01 and A02; and
- Underfrequency Bus A01 and A02.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage Bus A01 and A02; and
- Underfrequency Bus A01 and A02.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5 or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

Power Range Neutron Flux

Power Range Neutron Flux is actuated by two-out-of-four NIS power range channels. The LCO requirement for this Function ensures that this input to the P-7 interlock is available.

The LCO requires four channels of Power Range Neutron Flux to be OPERABLE in MODE 1.

OPERABILITY in MODE 1 ensures the Function is available to perform its increasing power Functions.

Turbine Impulse Pressure

The Turbine Impulse Pressure interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by one-out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure interlock to be OPERABLE in MODE 1.

The Turbine Impulse Chamber Pressure interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately ~~50~~35% power as determined by two-out-of-four NIS power range detectors.

The P-8 interlock automatically enables the Reactor Coolant Flow-Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately ~~50~~35% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock, is actuated at approximately 50% power (35% power for full design power Tavg ≤ 572°F), as determined by two-out-of-four NIS power range detectors, if the Steam Dump System is available. The LCO requirement for this Function ensures that the Turbine Trip-Low Autostop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock, to be OPERABLE in MODE 1 with one of two circulating water pump breakers closed and condenser vacuum greater than or equal to 22 "Hg.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip;

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux-Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors;
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-Low reactor trip and the Intermediate Range Neutron Flux reactor trip.

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

18. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires two OPERABLE RTBs. Two OPERABLE RTBs ensure no single random failure can disable the RPS trip capability. These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 18 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

20. Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism

The LCO requires the Reactor Trip Bypass Breaker and its associated Undervoltage Trip Mechanism to be OPERABLE when the Reactor Trip Bypass Breaker is racked in and closed. The bypass breaker and its associated trip mechanism are not required to be OPERABLE when the bypass breaker is open or racked out.

These trip Functions must be OPERABLE in MODE 1 or 2 when a Reactor Trip Bypass Breaker is racked in and closed. In MODE 3, 4, or 5, this RPS trip Function must be OPERABLE when a Reactor Trip Bypass Breaker is racked in and closed and the Rod Control System is capable of rod withdrawal.

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 18 and 19) and Automatic Trip Logic (Function 21) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RPS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RPS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip. These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

The RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be

BASES

ACTIONS (continued) entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's as-found ~~T~~trip ~~S~~setpoint is found non-conservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RPS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours. The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, this trip Function is no longer required to be OPERABLE.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies to the Manual Reactor Trip Function in MODE 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal.

- With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. If the Reactor Manual Trip channel cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened within the next hour.

The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Manual Reactor Trip Function is no longer required.

D.1 and D.2

Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-High;
- SG Water Level-Low Low; and
- SG Water Level - Low coincident with Steam Flow/Feedwater Flow Mismatch.

A known inoperable channel must be placed in the tripped condition within 1 hour. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips.

If the inoperable channel cannot be placed in the tripped condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is

BASES

ACTIONS (continued)

a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

E.1 and E.2

Condition E applies to the Underfrequency Bus A01 and A02 trip function. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint. The 6 hours to place the channel in the tripped condition is necessary due to plant design requiring maintenance personnel to effect the trip of the channel outside of the Control Room. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel and the low probability of occurrence of an event during this period that may require the protection afforded by this trip function.

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

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ACTIONS (continued) G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range perform the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

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ACTIONS (continued)

J.1 and J.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition.

K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure-Low;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Two Loops);
- Undervoltage Bus A01 and A02.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 1 hour. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 interlock and below the P-8 setpoint. These Functions do not have to be OPERABLE below the P-7 interlock because there are no loss of flow trips below the P-7 interlock. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

L.1 and L.2

Condition L applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 1 hour. If the channel cannot be restored to OPERABLE status or the channel placed

BASES

ACTIONS (continued)

in trip within the 1 hour, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This trip Function does not have to be OPERABLE below the P-8 setpoint because other RPS trip Functions provide core protection below the P-8 setpoint.

M.1 and M.2

Condition M applies to the RCP Breaker Position (Single Loop) reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel(s) must be restored to OPERABLE status within 1 hour. If the channel cannot be restored to OPERABLE status within the 1 hour, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours.

This places the unit in a MODE where the LCO is no longer applicable. This Function does not have to be OPERABLE below the P-8 setpoint because other RPS Functions provide core protection below the P-8 setpoint.

N.1 and N.2

Condition N applies to the RCP Breaker Position (Two Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 1 hour. If the channel cannot be restored to OPERABLE status in 1 hour, then THERMAL POWER must be reduced below the P-7 interlock within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable. This function does not have to be OPERABLE below the P-7 interlock because there are no loss of flow trips below the P-7 interlock. The Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER to below the P-7 interlock from full power in an orderly manner without challenging unit systems.

O.1 and O.2

Condition O applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 1 hour. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the

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ACTIONS (continued)

channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours.

P.1 and P.2

Condition P applies to the SI Input from ESFAS reactor trip and the RPS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RPS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action P.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action P.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action P.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train for up to 8 hours for surveillance testing, provided the other train is OPERABLE.

Q.1 and Q.2

Condition Q applies to the RTBs in MODES 1 and 2. With one RTB inoperable, 1 hour is allowed to restore the RTB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RPS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

The Required Actions have been modified by a Note allowing one channel to be bypassed for up to 8 hours provided the other channel is OPERABLE.

R.1 and R.2

Condition R applies to the P-6 interlock (in MODE 2) and the P-10 interlock. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours.

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ACTIONS (continued)

Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RPS Function.

<u>S.1 and S.2</u>

Condition S applies to the P-7, P-8, and P-9 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

<u>T.1 and T.2</u>

Condition T applies to the RTBs and the RTB Undervoltage and Shunt Trip Mechanisms in MODES 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal.

With one trip mechanism or RTB inoperable, the inoperable trip mechanism or RTB must be restored to OPERABLE status within 48 hours. The Completion Time is reasonable considering that the remaining OPERABLE trip mechanism or RTB is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

If the RTB or trip mechanism cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE in which the requirement does not apply. This is accomplished by opening the RTBs within the next hour (49 hours total time). The Completion Time of 1 hour provides sufficient time to accomplish this action in an orderly manner and takes into account the low probability of an event occurring in this interval.

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ACTIONS (continued)

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

With the unit in MODE 3, Condition T would apply to any inoperable RTB trip mechanisms. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 8 hours for the reasons stated under Condition Q.

The Completion Time of 48 hours is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1 and V.2

Condition V applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODE 1 or 2, when the RTBB is racked in and closed. With the required RTBB inoperable, 1 hour is allowed to restore the RTBB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour completion times are equal to the time allowed by LCO 3.0.3 for shutdown action in the event of a complete loss of RPS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

W.1 and W.2

Condition W applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODES 3, 4, or 5, when an RTBB is racked in and closed and the Rod Control System is capable of rod withdrawal. With the required RTBB inoperable, 48 hours is allowed to restore the RTBB to OPERABLE status or the unit must be placed in a MODE in which the requirement does not apply.

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ACTIONS (continued)

To achieve this status, the RTBs and RTBBs must be opened within the next 1 hour (49 hours total time). The Completion Time of 1 hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs and RTBBs open, this Function is no longer required.

X.1 and X.2

Condition X applies to the RPS Automatic Trip Logic in MODES 3, 4 or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With one train inoperable, 48 hours are allowed to restore the train to an OPERABLE status. The Completion Time of 48 hours is reasonable considering that in this condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring in this interval.

If the RPS Automatic Trip Logic cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE where this Function is not required to be OPERABLE. To achieve this status, the RTBs must be opened within the next 1 hour (49 hours total time). The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Automatic Trip Logic is no longer required.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RPS Functions.

Note that each channel of process protection supplies both trains of the RPS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations

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SURVEILLANCE REQUIREMENTS (continued)

between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hour is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate. The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. SR 3.3.1.3 is performed by means of the moveable incore

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SURVEILLANCE REQUIREMENTS (continued)

detection system. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%.

Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 50% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 50% RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

<u>SR 3.3.1.4</u>

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. The independent test for bypass breakers is included in SR 3.3.1.13. The bypass breaker test shall include an undervoltage trip. A Note has been added to SR 3.3.1.4 to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

<u>SR 3.3.1.5</u>

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST, every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of

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SURVEILLANCE REQUIREMENTS (continued)

every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5 is modified by two Notes. Note 1 provides an 8 hour delay in the requirement to perform this Surveillance for the Source Range Neutron Flux trip function instrumentation when power is reduced to below P-6. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.5 is no longer required to be performed. If the unit is to be in MODE 2 below P-6 for > 8 hours, this Surveillance must be performed prior to 8 hours after reducing power below P-6.

Note 2 excludes the RCP Breaker Position (Two Loop), Reactor Coolant Flow-Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions, and the P-6, P-7, P-8, P-9 and P-10 Interlocks. These functions/interlocks are tested at an 18 month frequency via SR 3.3.1.15.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

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SURVEILLANCE REQUIREMENTS (continued)

The difference between the current "as found" values and the ~~previous test "as left" values~~NTSP must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and verified to be within the required limits.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3.

SR 3.3.1.7 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with the safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing

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SURVEILLANCE REQUIREMENTS (continued)

unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels.

The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.8 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient

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SURVEILLANCE REQUIREMENTS (continued)

margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 31 days.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as-found" values and the NTSP ~~previous test "as left" values~~ must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time delays are adjusted to the prescribed values where applicable.

SR 3.3.1.10 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the

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SURVEILLANCE REQUIREMENTS (continued)

as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.11 is modified by two Notes as identified in Table 3.3.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative

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than the NTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RPS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, SI Input from ESFAS, and the Condenser Pressure-High and Circulating Water Pump Breaker Position inputs to the P-9 Interlock. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function for the Reactor Trip Breakers and the undervoltage trip circuits for the Reactor Trip Bypass Breakers.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

SR 3.3.1.15

SR 3.3.1.15 is the performance of an ACTUATION LOGIC TEST on the RCP Breaker Position (Two Loop), Reactor Coolant Flow-Low (Two

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REFERENCES	1. FSAR, Chapter 7. 2. FSAR, Chapter 14. 3. IEEE-279-1968. 4. 10 CFR 50.49. ~~5. DG-I01, Instrument Setpoint Methodology.~~

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

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The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the ESFAS, as well as specifying LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulations as "…settings for automatic protective devices…, so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a protective action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative that the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Nominal Trip Setpoint (NTSP) specified in Table 3.3.2-1 is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the NTSP accounts for uncertainties in setting the channel (e.g. calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the NTSP ensures that SLs are not exceeded. Therefore, the NTSP meets the definition of an LSSS.

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. Operable is defined in the Technical Specifications as "…being capable of performing its safety functions(s)." Relying solely on the NTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for

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the 'as-found' value of a protection channel setting during a surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the NTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the NTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protection channel. Therefore, the channel would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the channel to the NTSP to account for further drift during the next surveillance interval.

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB).
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR50 and 10CFR 00 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences of the event.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;

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- Signal processing equipment including analog protection system, field contacts, and protection channel sets: provide signal conditioning, compatible electrical signal output to protection system channels, and control board/control room/miscellaneous indications; and

- Relay Logic Racks including input, logic and output devices: initiates proper Engineered Safety Feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Protection System (RPS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the NTSP and Allowable Values~~.~~ The OPERABILITY of each transmitter or sensor is determined by either~~can be evaluated when its~~ "as found" calibration data ~~are compared against its documented acceptance criteria~~evaluated during the CHANNEL CALIBRATION or by qualitative assessment of field transmitter or sensor, as related to the channel behavior observed during performance of the CHANNEL CHECK.

Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with ~~setpoints~~NTSPs derived from Analytical Limits established by safety analyses. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still

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OPERABLE with a two-out-of-two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the Relay Logic Racks and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1968 (Ref. ~~2~~3).

~~Allowable Values~~NTSPs and ESFAS Setpoints

The trip setpoints used in the bistables are based on analytical limits established in the safety analyses. The selection of these analytical limits is such that adequate protection is provided when all sensors and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, ~~and~~ instrument drift, and severe environments as defined by 10 CFR 50.49 (Reference 4), the Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservativ~~ely adjusted~~ with respect to the analytical limits. A ~~detailed~~ description of the methodology used to calculate the Allowable Values and ~~including their explicit uncertainties, is provided in DGI-01, Instrument Setpoint Methodology (Ref. 4).~~ Nominal Trip Setpoints (including their explicit uncertainties), as well as the as-left and as-found tolerances, is provided in FSAR Chapter 7, Reference 1. The magnitude of the uncertainties are factored into the determination of each NTSP and corresponding Allowable Value in design basis calculations. The ~~actual nominal~~ field trip setpoint entered into the bistable is equal to or more conservative than that specified by the ~~Allowable Value~~NTSP to account for changes in random measurement errors detectable by a COT. ~~If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.~~ The Allowable Value serves as the as-found trip setpoint Technical Specification OPERABILITY limit for the purpose of the COT.

The NTSP is the value at which the bistables are set and is the expected value to be achieved during calibration. The NTSP value is the LSSS and ensures the safety analysis limits are met for the surveillance interval selected when a channel is adjusted based on stated channel uncertainties. Any bistable is considered to be properly adjusted when the "as-left" NTSP value is within the as-left tolerance for

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CHANNEL CALIBRATION uncertainty allowance (i.e. + rack calibration and comparator setting uncertainties). The NTSP value is therefore considered a "nominal value" for the purposes of the COT and CHANNEL CALIBRATION.

Nominal Trip Setpoints, in conjunction with the use of the as-found and as-left tolerances together with the requirements of the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

Note that the Allowable Values listed in Table 3.3.2-1 are the least conservative value of the s-found setpoint that a channel can have during a periodic CHANNEL Calibration, COT, or a TADOT that requires trip setpoint verification.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

Each channel can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

~~The Allowable Values listed in Table 3.3.2-1 are based on the methodology described in Reference 4, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Allowable Value. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.~~

Relay Logic Racks

The Relay Logic Rack equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of Relay Logic Racks, each performing the same functions, are provided.

The Relay Logic Racks perform the decision logic for most ESF equipment actuation; generates the electrical output signals that initiate

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the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the Relay Logic Rack equipment and combined into logic matrices that represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

The actuation of ESF components is accomplished through master and slave relays. The Relay Logic Racks energize the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the ~~end devices~~channels.

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Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure-Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are ~~qualitatively~~ implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. ~~1~~2).

Permissive and interlock setpoints allow the blocking of trips during plant start-ups and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventative or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, they are generally considered as nominal values with regard to measurement accuracy, (i.e. the value indicated is sufficiently

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

close to the necessary value to ensure proper operation of the safety system to turn the AOO).

The LCO requires all instrumentation performing an ESFAS Function listed in Table 3.3.2-1 in the accompanying LCO, to be OPERABLE. The Allowable Value specified in Table 3.3.2-1 is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST (COT). As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the channel (NTSP) will ensure that a SL is not exceeded at any given point of time as long as the channel has not drifted beyond that expected during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

If the actual setting of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE but a degraded condition has been identified. During the SR performance, the condition of the channel will be evaluated. If the channel is functioning as required and is expected to pass the next surveillance, then the channel can be restored to service at the completion of the surveillance. If any of the above described evaluations determine that the channel is not performing as expected, the channel is degraded because it may not pass its next surveillance test. If the channel setpoint cannot be reset to the as-left tolerance around the NTSP, it is inoperable. After the surveillance is completed, the channels as-found setting will be entered into the *Corrective Action Program for further evaluation.*

A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the operability of this instrument must be verified based on the field trip setpoint and not the NTSP. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

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If the actual setting of the channel is found to be non-conservative with respect to the Allowable Value, the channel would be considered inoperable. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protection channels do not function as required. The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F); and

2. Boration to ensure recovery and maintenance of SDM (k_{eff} < 1.0).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Containment Isolation;
- Containment Ventilation Isolation;
- Reactor Trip;
- Feedwater Isolation;
- Start of motor driven auxiliary feedwater (AFW) pumps; and
- Control room ventilation isolation.

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses;
- Start of AFW to ensure secondary side cooling capability; and
- Isolation of the control room to ensure habitability.

a. Safety Injection-Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals with the exception of Containment Isolation.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet. Each push button actuates both trains. This configuration does not allow testing at power.

b. Safety Injection-Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

large number of components actuated on a SI, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection-Containment Pressure-High

This signal provides protection against the following accidents:

- SLB inside containment; and
- LOCA.

Containment Pressure-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters and electronics are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations.

Thus, the high pressure Function will not experience any adverse environmental conditions and the ~~Allowable Value~~NTSP reflects only steady state instrument uncertainties.

Containment Pressure-High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection-Pressurizer Pressure-Low

This signal provides protection against the following accidents:

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

- Inadvertent opening of a steam generator (SG) relief or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. However, two independent PORV open signals must be present before a PORV can open. Therefore, a single pressure channel failing high will not fail a PORV open and trigger a depressurization/SI event. Additionally, in the event of a failed open spray valve, RCS depressurization would be slow enough to be recognized by the operator and mitigated through manual actions to close the spray valve and energize the pressurizer heaters prior to reaching saturated conditions in the RCS. Therefore, there would be no uncontrolled loss of RCS inventory and no need for boron injection. Therefore, only three protection channels are necessary to satisfy the protective requirements.

This Function must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the Pressurizer Pressure interlock. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure-High signal.

This Function is not required to be OPERABLE in MODE 3 below the Pressurizer Pressure interlock. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

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e. Safety Injection-Steam Line Pressure-Low

Steam Line Pressure-Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure-Low provides a signal for control of the main steam atmospheric steam dump valves. However, a failure in a steam line pressure channel will not create a control failure that would result in a low steamline pressure SI event. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

With the transmitters located in the fan rooms and in the fuel pool area, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the ~~Allowable Value~~NTSP reflects both steady state and adverse environmental instrument uncertainties.

This Function is anticipatory in nature and has a lead/lag ~~ratio~~ time constant of ~~12~~18/2 seconds.

Steam Line Pressure-Low must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the Pressurizer Pressure interlock. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

2. Containment Spray

Containment Spray provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;
2. Reduces the amount of radioactive iodine in the containment atmosphere; and

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3. Adjusts the pH of the water in the containment recirculation sump after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

The containment spray actuation signal starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps and mixed with a sodium hydroxide solution from the spray additive tank. When the RWST reaches the low low level setpoint, the spray pump suctions are shifted to the containment sump if continued containment spray is required. Containment spray is actuated automatically by Containment Pressure-High High.

a. Containment Spray-Manual Initiation

The operator can initiate containment spray at any time from the control room by simultaneously depressing two containment spray actuation pushbuttons. Because an inadvertent actuation of containment spray could have such serious consequences, two pushbuttons must be pushed simultaneously to initiate both trains of containment spray.

The LCO requires two channels to be OPERABLE. Each channel consists of one pushbutton and two sets of contacts, with one set of contacts in each train. Therefore an inoperable channel fails both trains of manual initiation.

b. Containment Spray-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b. Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to

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containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

c. Containment Spray-Containment Pressure-High High

This signal provides protection against a LOCA or a SLB inside containment. The transmitters are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the ~~Trip Setpoint~~NTSP reflects only steady state instrument uncertainties.

This is one of the only Functions that require the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious.

The Containment Pressure-High High Function consists of two sets with three channels in each set. Each set is a two-out-of-three logic where the outputs are combined so that both sets tripped initiates Containment Spray. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energized to trip. Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to

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pressurize the containment and reach the Containment Pressure-High High setpoints.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

Containment Isolation signals isolate all automatically isolable process lines, except component cooling water (CCW), main feedwater lines and main steam lines. The main feedwater and main steam lines are isolated by other functions because forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating CCW may force the use of feed and bleed cooling, which could prove more difficult to control.

a. Containment Isolation

(1) Containment Isolation-Manual Initiation

The LCO requires two channels to be OPERABLE. A channel consists of one pushbutton and two sets of contacts, with one set of contacts in each train.

Manual Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Containment Isolation also actuates Containment Ventilation Isolation.

(2) Containment Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to

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manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Containment Isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Containment Isolation-Safety Injection

Containment Isolation is also initiated by all Functions that initiate SI except Manual SI initiation. The Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For a SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For a SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

a. Steam Line Isolation-Manual Initiation

The LCO requires one channel per loop to be OPERABLE. A channel consists of the control switch and two sets of contacts, with one set of contacts in each train.

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are two switches in the control room, one for each MSIV.

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

b. Steam Line Isolation-Automatic Actuation Logic and Actuation Relays

The LCO requires two trains to be OPERABLE. Actuation logic consists of two trains, with each train providing output to each MSIV through individual relays.

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have a SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience a SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation-Containment Pressure-High High

This Function actuates closure of the MSIVs in the event of a LOCA or a SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters are located outside containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. Containment Pressure-High High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the ~~Allowable Values~~NTSP reflects only steady state instrument uncertainties.

Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High High setpoint.

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

d. Steam Line Isolation-High Steam Flow Coincident With Safety Injection and Coincident With T_{avg}-Low

This Function provides closure of the MSIVs during a SLB or inadvertent opening of an SG relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one-out-of-two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation.

The High Steam Flow ~~Allowable Value~~NTSP is a ΔP corresponding to approximately 20% of full steam flow at no load steam pressure.

With the transmitters (d/p cells) located inside containment, it is possible for them to experience adverse environmental conditions during a SLB event. Therefore, the ~~Allowable Value~~NTSP reflect both steady state and adverse environmental instrument uncertainties.

The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

The T_{avg}-Low Function consists of four channels (two in each loop), providing input to both trains in a two-out-of-four logic configuration. Three channels of T_{avg} are required to be OPERABLE. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of three OPERABLE channels ensures no single random failure disables the T_{avg}-Low Function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore,

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

additional channels are not required to address control protection interaction issues.

With the T_{avg} resistance temperature detectors (RTDs) located inside the containment, it is possible for them to experience adverse environmental conditions during a SLB event. Therefore, the ~~Trip Setpoint~~NTSP reflects both steady state and adverse environmental instrumental uncertainties.

This Function must be OPERABLE in MODES 1 and 2, and in MODE 3, when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

e. Steam Line Isolation-High High Steam Flow Coincident With Safety Injection

This Function provides closure of the MSIVs during a steam line break (or inadvertent opening of a relief or safety valve) to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the Function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements.

The ~~Allowable Value~~NTSP for high steam flow is a ΔP, corresponding to approximately 120% of full steam flow at full steam pressure.

With the transmitters located inside containment, it is possible for them to experience adverse environmental conditions during a SLB event. Therefore, the NTSP ~~Allowable Value~~ reflects both steady state and adverse environmental instrument uncertainties.

The main steam lines isolate only if the high-high steam flow signal occurs coincident with an SI signal. The Main Steam Line

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

5. Feedwater Isolation

The primary function of the Feedwater Isolation signal is to stop the excessive flow of feedwater into the SGs. This Function is necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated on an SI signal, or when the level in either SG exceeds the high setpoint.

An SI signal results in the following actions:

- ~~MFW pumps trip (causes subsequent closure of the MFW pump discharge vales)~~MFIV isolation; and
- MFRVs and the bypass regulating valves close.

A SG Water Level-High in either SG results in the closure of the MFRVs and the bypass regulating valves.

a. Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Feedwater Isolation-Steam Generator Water Level-High

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

the SG Water Level Control System. If this input to the SG Water Level Control System fails low, it would cause a control action to open the Feedwater Control Valves for the affected SG. The remaining channels, in a two-out-of-two configuration, would be required to detect a high SG Water Level condition and initiate a Feedwater Isolation to prevent an overfill condition. Therefore this configuration does not meet the single failure criteria of Reference ~~1~~2. However, justification for a two-out-of-three Feedwater Isolation-SG Water Level-High Function is provided in NUREG-1218, Reference ~~5~~4.

~~Table 3.3.2-1 identifies the Technical Specification Allowable Value for the Feedwater Isolation- SG Water Level - High function as not applicable (NA). No Analytical Value is assumed in the accident analysis for this function. The nominal setting required for the Feedwater Isolation - SG Water Level - High function is 78% of span. This nominal setting was developed outside of the setpoint methodology and has been provided by the NSSS supplier~~

c. Feedwater Isolation-Safety Injection

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function.

Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 and 3 except when all MFRVs, and associated bypass valves are closed and de-activated. In MODES 4, 5, and 6, the MFW System is not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has ~~two~~a motor driven pumps and a turbine driven pump, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST) (not safety related). Upon a low ~~level in the CST,~~ pressure in the

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

AFW pump suction piping, the suction source will automatically ~~the operators can manually~~ realign the pump suctions to the Service Water System, which is the safety related water source. The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately.

a. Auxiliary Feedwater-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Auxiliary Feedwater-Steam Generator Water Level-Low Low

SG Water Level-Low Low provides protection against a loss of heat sink. A loss of MFW would result in a loss of SG water level. SG Water Level-Low Low in either SG will cause both motor driven pumps to start. The system is aligned so that upon start of the pumps, water immediately begins to flow to the SGs. SG Water Level-Low Low in both SGs will cause the turbine driven AFW pump to start.

~~With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Allowable Values reflects the inclusion of both steady state and adverse environmental instrument uncertainties.~~

c. Auxiliary Feedwater-Safety Injection

An SI signal starts ~~the motor driven~~both AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

Functions 6.a through 6.c must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. SG Water Level-Low Low in any operating SG will cause the motor driven and turbine driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. ~~SG Water Level-Low Low in both SGs will cause the turbine driven pump to start.~~ These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

d. Auxiliary Feedwater-Undervoltage Bus A01 and A02

The LCO requires two channels per bus to be OPERABLE. A channel consists of an undervoltage relay and one set of associated contacts.

A loss of power on the A01 and A02 buses provides indication of a pending loss of both Main Feedwater pumps and the subsequent need for some method of decay heat removal. A loss of power to Buses A01 and A02 will start ~~the turbine driven~~ both AFW pumps to ensure that ~~at least one~~ the SGs contain~~s~~ enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

Function 6.d must be OPERABLE in MODES 1 and 2. This ensures that ~~at least one~~the SGs are~~is~~ provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, and 5, the MFW pumps may be normally shut down, and thus a pump trip is not indicative of a condition requiring automatic AFW initiation.

e. AFW Pump Suction Transfer on Suction Pressure-Low

A low pressure signal in the AFW pump suction lines protects the AFW pumps against a loss of the normal supply of water for the pumps. The pressure switches are located on the AFW pump suction lines from the CSTs. A low pressure signal sensed by the pump suction switches will cause the safety-related source of water, Service Water, to be automatically aligned to the AFW pumps. The alignment of the Service Water System to maintain at least one of the SGs per unit as the heat sink for reactor decay heat and sensible heat removal.

Table 3.3.2-1 Notes 1 and 2 are applicable.

This function must be OPERABLE in MODES 1,2 and 3 to ensure a safety grade supply of water for the AFW system to maintain the SGs as the heat sink for the reactor. In MODE 4, AFW automatic suction transfer does not need to be

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

OPERABLE because the RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

~~7. Condensate Isolation~~

~~The Condensate Isolation Function serves as a backup protection function in the event of a Main Steam Line Break inside containment with a failure of the Main Feedwater lines to isolate. An evaluation of IE Bulletin 80-04 showed that a single failure of a MFRV to close on a SI signal could allow feedwater addition to the faulted SG, leading to containment overpressure.~~

~~a. Containment Pressure-Condensate Isolation (CPCI)~~

~~The Condensate Isolation Function is actuated when containment pressure exceeds the high setpoint, and performs the following functions:~~

- ~~Trips the condensate pumps; and~~
- ~~Trips the heater drain pumps.~~

~~The Condensate Isolation Function must be OPERABLE in MODES 1, 2 and 3, except when all MFRVs and associated bypass valves are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5 and 6, because there is insufficient energy in the secondary side of the unit to have an accident.~~

~~b. Condensate Isolation - Automatic Actuation Logic and Actuation Relays~~

~~Automatic Actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.~~

~~8~~7. Pressurizer Pressure Safety Injection Block

To allow some flexibility in unit operations, the Pressurizer Pressure SI Block is included as part of the ESFAS.
The block permits a normal unit cooldown and depressurization without actuation of SI. With two-out-of-three pressurizer pressure channels (discussed previously) less than the setpoint, the operator can manually block the Pressurizer Pressure-Low and Steam Line

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)	Pressure-Low SI signals. With two-out-of-three pressurizer pressure channels above the setpoint, the Pressurizer Pressure-Low and Steam Line Pressure-Low SI signals are automatically enabled. The operator can also enable these trips by use of the respective manual reset buttons. The ~~Allowable Value~~NTSP reflects only steady state instrument uncertainties. This Function must be OPERABLE in MODES 1, 2, and 3 to allow automatic initiation of SI actuation on Pressurizer Pressure-Low or Steam Line Pressure-Low signals. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because system pressure must already be below the setpoint for the requirements of the heatup and cooldown curves to be met. The ESFAS instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement.~~10 CFR 50.36(c)2(ii))
ACTIONS	A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1. In the event a channel's ~~Trip Setpoint~~NTSP is found non-conservative with respect to the Allowable Value, or the transmitter, instrument Loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate. When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation. A.1 Condition A applies to all ESFAS protection functions. Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are

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ACTIONS (continued) those from the referenced Conditions and Required Actions.

B.1, B.2.1 and B.2.2

Condition B applies to manual initiation of:

- SI; and
- Containment Isolation.

If a channel is inoperable, 48 hours are allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. If the channel cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1 and C.2.2

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI;
- Containment Spray; and
- Containment Isolation.

If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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D.1, D.2.1 and D.2.2

Condition D applies to:

- Containment Pressure-High;
- Pressurizer Pressure-Low;
- Steam Line Pressure-Low;
- Containment Pressure-High High;
- High Steam Flow Coincident With Safety Injection Coincident With T_{avg}-Low;
- High High Steam Flow Coincident With Safety Injection;
- SG Water level-Low Low; and
- SG Water level-High.

If one channel is inoperable, 1 hour is allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Placing the channel in the tripped condition is necessary to maintain a logic configuration that satisfies redundancy requirements.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 1 hour requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

E.1, E.2.1, and E.2.2

Condition E applies to manual initiation of Containment Spray. If one or both channels are inoperable, 1 hour is allowed to return the inoperable channel(s) to OPERABLE status. The Completion Time of one hour is reasonable considering that there are OPERABLE automatic actuation functions credited to perform the safety function and the low probability of an event occurring during this interval. If the inoperable channel(s) cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the

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unit in at least MODE 3 within an additional 6 hours (7 hours total time) and in MODE 5 within an additional 30 hours (37 hours total time). The allowable Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

F.1, F.2.1, and F.2.2

Condition F applies to Manual Initiation of Steam Line Isolation.

If a channel is inoperable, 1 hour is allowed to return it to an OPERABLE status. The Completion Time of one hour is reasonable considering the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

G.1, G.2.1 and G.2.2

Condition G applies to the automatic actuation logic and actuation relays for the Steam Line Isolation, Feedwater Isolation, Condensate Isolation and AFW actuation Functions.

If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

H.1 and H.2

Condition H applies to the Undervoltage Bus A01 and A02 Function.

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If one channel is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or place it in the tripped condition. If placed in the tripped condition, this Function is then in a partial trip condition where one-out-of-two logic will result in actuation. The 6 hours to place the channel in the tripped condition is necessary due to plant design requiring maintenance personnel to effect the trip of the channel outside of the control room. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, this Function is no longer required OPERABLE.

I.1, I.2.1 and I.2.2

Condition I applies to the Pressurizer Pressure SI Block.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the block. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the block is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the Pressurizer Pressure SI block.

J.1 and J.2

Condition J applies to the AFW Pump Suction Transfer on Suction Pressure-Low

If one channel on an individual Auxiliary Feedwater pump is inoperable, 48 hours are allowed to restore the channel to OPERABLE status or declare the associated AFW pump inoperable. The 48 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 48 hour Completion Time takes into account the capacity of the remaining AFW sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. If the out of service time for the channel extends beyond 48 hours, then the automatic

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transfer of the safety-related water source is considered inoperable, making the associated AFW pump inoperable.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST on all ESFAS Automatic Actuation Logic every 31 days on a STAGGERED TEST BASIS. This test includes the application of various simulated or actual input combinations in conjunction with each possible interlock state and verification of the required logic output. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found ~~within~~conservative with respect to the Allowable Values specified in Table 3.3.2-1.

The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as-found" and "as-left" values must also be recorded and reviewed for consistency with the assumptions of the ~~surveillance interval extension analysis (Ref. 4) when applicable.~~setpoint methodology.

The Frequency of 92 days is justified in Reference ~~4~~5.

SR 3.3.2.3 is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with the safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint

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SURVEILLANCE REQUIREMENTS (continued)

more conservative than the NTSP is used in the plant surveillance procedures (field trip setpoint), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and as-found tolerances be in the FSAR Chapter 7, Reference 1.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay and verifying contact operation. This test is performed every 18 months.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. This test is performed every 18 months.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT every 31 days. This test is a check of the Undervoltage Bus A01 and A02 Function.

The Frequency is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.7

SR 3.3.2.7 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. It is performed every 18 months. The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle.

SR 3.3.2.8

SR 3.3.2.8 is the performance of a CHANNEL CALIBRATION.

BASES

SURVEILLANCE REQUIREMENTS (continued)

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint methodology. The difference between the current "as-found" values and the previous test "as-left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.8 is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with the safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the NTSP. Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures(field trip setpoint), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable.

The second Note also requires that the methodologies for calculating the as-left and as-found tolerances be in the FSAR, Chapter 7, Reference 1.

BASES

REFERENCES

1. FSAR Chapter 7
2. FSAR, Chapter 14.
3. IEEE-279-1968.
4. 10 CFR 50.49.

~~DGI-01, Instrument Setpoint Methodology.~~

5. NUREG-1218, April 1988.

ENCLOSURE 5

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261 SUPPLEMENT 3
EXTENDED POWER UPRATE

LICENSING REPORT SECTION 2.4.1

40 pages follow

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Feature Actuation, and Control Systems

2.4.1.1 Regulatory Evaluation

Instrumentation and control (I&C) systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse I&C systems and equipment are provided for the express purpose of protecting against potential common-mode failures of I&C protection systems. Point Beach Nuclear Plant (PBNP) conducted a review of the reactor protection system (RPS), engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse I&C systems for the proposed EPU to ensure that the systems and changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The PBNP review was also conducted to ensure that failures of the systems and changes necessary for EPU do not affect safety functions.

The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a (a)(1), 10 CFR 50.55a(h), and the following 10 CFR 50 Appendix A, General Design Criteria (GDC):

- GDC-1, insofar as it requires that structures, systems, and components (SCCs) important-to-safety are designed, fabricated, erected, and tested to quality standards commensurate with their importance to functions to be performed.
- GDC-4, insofar as it requires that SSCs be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC-13, insofar as it requires that instrumentation is provided to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and for accident conditions as appropriate to ensure safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.
- GDC-19, insofar as it requires that a control room is provided from which actions can be taken to operate the nuclear unit safely under normal conditions, and maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs).
- GDC-20, insofar as it requires protection systems be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of important-to-safety systems and components.

- GDC-21 insofar as it requires protection systems be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

- GDC-22 insofar as it requires protection systems be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis.

- GDC-23 insofar as it requires protection systems be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

- GDC-24, insofar as it requires that the protection system is separated from the control systems to the extent that a system satisfying all reliability, redundancy, and independence requirements of the protection systems is left intact in the event of a failure of any single control system component or channel, or failure or removal from service of any single control system component or channel that is common to the control and protection systems. Interconnection of the protection and control systems will be limited so as to ensure that safety is not significantly impaired.

Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

PBNP Current Licensing Basis

As noted in PBNP Final Safety Analysis Report (FSAR), Section 1.3, the GDC used during the licensing of PBNP Station predates those provided today in 10 CFR 50, Appendix A. The origin of the PBNP GDC relative to the Atomic Energy Commission proposed GDC is discussed in FSAR, Section 1.3. The GDC number in parentheses following the criterion description corresponds to the number of the Atomic Industrial Forum version of the proposed General Design Criterion that existed when PBNP was originally licensed..

The PBNP GDCs that correspond to 10 CFR 50 Appendix A GDC-1, 4, 13, 19, 20, 21, 22, 23, and 24 are as follows:

CRITERION: Those systems and components of reactor facilities which are essential to the prevention or the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (PBNP GDC 1)

PBNP GDC 1 broadly applies to plant equipment, including I&C, and is discussed in FSAR 4.1, Design Basis. Quality standards of material selection, design, fabrication, and inspection conform to the applicable provisions of recognized codes and good nuclear practice.

CRITERION: The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel. (PBNP GDC 11)

The plant is equipped with a common control room which contains those controls and instrumentation necessary for operation of each unit's reactor and turbine generator under normal and accident conditions.

CRITERION: Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables. (PBNP GDC 12)

Instrumentation and controls are provided to monitor and maintain important reactor parameters (including neutron flux, primary coolant pressure, loop flow rate, coolant temperatures, and control rod positions) within prescribed operating ranges. Other I&C systems are provided to monitor and maintain, within prescribed operating ranges, the temperatures, pressure, flow, and levels in the reactor coolant system, steam systems, containment, and other auxiliary systems. Process variables which are required on a continuous basis for the startup, power operation, and shutdown of the plant are indicated, recorded, and controlled from the control room, which is a controlled access area. The quantity and types of instrumentation provided are adequate for safe and orderly operation of all systems and processes over the full operating range of the plant.

CRITERION: Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core. (PBNP GDC 13)

Ex-core nuclear instrumentation is used primarily for reactor protection, by monitoring neutron flux and by generating appropriate trip and alarm functions for various phases of the reactor operating and shutdown conditions. Nuclear instrumentation also provides a fission process control function and indicates reactor fission process status during startup and power operation.

CRITERION: Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. (PBNP GDC 14)

If the RPS sensors detect conditions which indicate an approach to unsafe operating conditions that require core protection, the system actuates alarms, prevents control rod motion, initiates load runback, and initiates reactor trip by opening the reactor trip breakers.

CRITERION: Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features. (PBNP GDC 15)

Instrumentation and controls provided for the ESFAS are designed to automatically initiate engineered safety features (ESF) equipment during those accidents which are mitigated by automatic ESF equipment operation. Actuated ESF equipment (depending on the severity of the condition) includes the Safety Injection (SI) System, the Containment Air Recirculation Cooling System, containment isolation, and the Containment Spray System, as discussed in FSAR Section 6.0, Engineered Safety Features Criteria.

The ESFAS consists of redundant analog channels, each containing sensors for different trip parameters, channel circuitry, and trip bistables. The trip bistable outputs are combined in coincident trip logic in two redundant actuation trains. Sufficient redundancy is provided so that a single failure will not defeat the actuation function.

CRITERION: Protection systems shall be designed for high functional reliability and inservice testability necessary to avoid undue risk to the health and safety of the public. (PBNP GDC 19)

A minimum of two independent protection channels are provided in the RPS and ESFAS for each trip variable, with most variables having three or four independent channels. Protection system reliability to avoid unnecessary trips is provided by redundancy within each tripping function and the use of coincidence trip logic. Each protection channel associated with any specific trip variable is provided with an independent source of electrical power and independent circuitry from the sensor through the trip bistable. Therefore, in the event that the loss of a single protection channel occurs, only that particular protection channel is affected, and coincidence logic is not satisfied to initiate a protective action (unless a one-out-of-two coincidence logic is employed).

Most protection channels are designed so that on loss of power, the bistables fail in the tripped condition (the preferred failure direction for most protection channels).

Protection channels are designed with sufficient redundancy for individual channel calibration and testing during power operation without degrading the protection functions. To remove an analog channel from service for test, calibration, or maintenance, all of the associated channel's trip signals to the RPS or ESFAS are first placed in the tripped condition. Tripping a channel to be tested will not cause a reactor trip or ESF actuation unless a trip condition already exists in a redundant channel.

CRITERION: Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. (PBNP GDC 20)

A minimum of two independent protection channels are provided in the RPS and ESFAS for each trip variable, with most variables having three or four independent channels. The design is such that no single failure within the protection systems or their supporting systems will defeat the overall protective function or violate the protection system design criteria. The design includes redundant, independent channels extending from sensors to the trip bistable outputs, which are then combined into coincidence trip logic in two redundant logic trains that extend to the final actuated devices. Sufficient redundancy and coincidence logic is included to reliably accomplish the protective functions if a single failure should occur, while also minimizing unnecessary protective actions due to single failures.

FSAR Section 7.2, Reactor Protection System, and FSAR Section 7.3, Engineered Safety Feature Actuation System, discuss certain protection system backup trips that may not fully meet the single failure criterion. However, failure of a backup trip does not prevent proper protective action of primary trips assumed in the accident analyses, and does not represent a loss of the protective function discussed in PBNP GDC 20.

When protection system sensors supply signals for control functions, an isolation amplifier is used to fully isolate the control signal from the protection signal. Therefore, any control circuit failure is prevented from affecting the protection channel. In a few circuits which provide main control board annunciation and stop rod withdrawal, the safety and control functions are combined from the sensor through dual alarm units. In these circuits, a failure in the control portion of the circuit can cause the safety portion of the circuit to go to its trip position. This may result in initiation of protective action.

CRITERION: The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident shall not result in loss of the protection function or shall be tolerable on some other basis. (PBNP GDC 23)

Potentially adverse conditions to which redundant protection system equipment may be exposed include adverse environmental effects, fires, earthquakes, and missile hazards. The design and layout of protection system components precludes loss of the protection function as a result of adverse conditions to which the components may be exposed.

Physical and electrical separation of redundant protection system channels and trains is employed to reduce the probability of an external hazard, such as a fire or missile, impairing the protection function through a common mode failure. Separation of redundant analog channels originates at the process sensors and continues along the field wiring, through containment penetrations, to the analog protection racks. As mentioned previously under PBNP GDC 20, some sensors for pressurizer pressure and reactor coolant flow may share common sensing lines, but the consequence of a line failure (rupture) will not prevent a protective action from occurring.

Separation of redundant protection channel/train field wiring is achieved using separate wireways, cable trays, conduit runs, and containment penetrations for redundant channels and trains. Separate, dedicated racks for each channel and train are provided to terminate the field wiring, so that internal wiring within a rack is limited to a single channel or train. Power supplies to redundant channels and trains are provided from separate 120 VAC instrument buses and from separate DC buses, respectively.

FSAR Section 7.2, Rector Protection System, and FSAR Section 7.3, Engineered Safety Feature Actuation System, discuss certain protection system backup trips that may not fully meet wiring separation criteria for redundant trains. However, failure of a backup trip circuit does not prevent proper protective action of primary trips assumed in the accident analyses, and does not represent a loss of the protective function discussed in PBNP GDC 23.

Environmental qualification of electrical/electronic equipment is addressed in FSAR Section 7.2.3.5, Environmental Qualification of Reactor Protection System Equipment.

Seismic qualification of protection system components is addressed in FSAR Section 7.2.3.4, Seismic Qualification of Protection System Equipment.

CRITERION: Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred. (PBNP GDC 25)

During power operation, each reactor protection channel and logic train is capable of being calibrated and tripped independently by simulated signals to verify its operation, without tripping the plant. The testing scheme includes checking through the trip logic to the reactor trip breakers. Thus, the operability of each channel and logic train can be determined conveniently and without ambiguity.

During power operation, each ESFAS channel and logic train is capable of being calibrated and tripped independently by simulated signals to verify its operation up to the final actuation device. Because ESF equipment actuation would adversely impact plant operation at power, the final ESF actuation devices are not cycled while the reactor is at power. A resistance check of the relay coils is performed at power, but actuation of ESF equipment is performed during refueling shutdowns, rather than at power.

CRITERION: The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced. (PBNP GDC 26)

Each RPS channel and train is designed on the "de-energize to operate" principle; an open circuit or loss of power causes the respective channel or train to go into its tripped condition (the "preferred failure" direction).

The analog channels for the ESFAS, with the exception of containment spray actuation, are designed on the same "de-energize to operate" principle as the reactor protection channels. The high-high containment pressure channels for containment spray actuation are designed as energize-to-operate, to avoid spray operation on inadvertent channel power failures.

Regarding the two ESFAS actuation trains, the output relays are "energize-to-operate" and require power to actuate ESF equipment. This design prevents inadvertent ESF equipment actuation on power failure of an actuation train (the "preferred failure" direction).

FSAR Chapter 7, Instrumentation and Control, addresses the design features and functions of the RPS, ESFAS and other reactor control systems and instrumentation.

In addition to the evaluations described in the FSAR, PBNP's electrical and I&C systems were evaluated for plant license renewal. The evaluation of the electrical and I&C components, and the subsequent review and conclusions are discussed in:

- Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2, (NUREG-1839), dated December 2005.

Electrical and instrumentation and controls are described in SER Section 2.5 and SER Section 3.6.

2.4.1.2 Technical Evaluation

2.4.1.2.1 Introduction

With respect to the EPU, the RPS, ESFAS, reactor control systems, and balance of plant (BOP) instrumentation are potentially impacted by the increase in reactor thermal power.

2.4.1.2.2 Input Parameters and Assumptions

The design parameters associated with the EPU are identified in LR Section 1.1, Nuclear Steam Supply System Parameters. The initial best estimate nominal operating parameters are also identified in LR Section 1.1, Nuclear Steam Supply System Parameters. The BOP parameters are derived from the heat balance.

2.4.1.2.3 Description of Analyses and Evaluations

The effects of the EPU have been evaluated for normal operation, operational transients, and accident conditions described in the FSAR. These analyses used the most conservative combination (where appropriate) of Nuclear Steam Supply System (NSSS) design values from LR Section 1.1, Nuclear Steam Supply System Parameters, Table 1-1. In addition, these analyses included changes to specific RPS, ESFAS, control system, and BOP setpoints. The results of the transient and accident analyses are described in the following LR Section:

- LR Section 2.4.2, Plant Operability.
- LR Section 2.6, Containment Review Considerations
- LR Section 2.8.5, Accident and Transient Analyses

2.4.1.2.3.1 Balance of Plant

2.4.1.2.3.1.1 Description of Balance-of-Plant Analyses and Evaluation

Operation of the plant at EPU conditions has minimal effect on BOP system I&C devices. Based on EPU operating conditions for the power conversion and auxiliary systems, most process control valves and instrumentation have sufficient range/adjustment capability for use at the EPU conditions.

The evaluation methodology used to evaluate the BOP system instrumentation includes the following basic steps:

- Perform system analysis to determine how the EPU conditions/ranges/setpoints compare to the current operating conditions/ranges/setpoints for the BOP systems.
- For those systems (subsystems) that are impacted by the EPU, determine the major process instrumentation or board-mounted instruments from the piping and instrument diagrams (P&IDs) and instrument scaling calculations and tabulate the pre-EPU and post-EPU process data.
- Analyze the affected instruments to determine EPU instrument impact.
- For those instruments affected by the EPU, recommend new scaling, setpoints, ranges, or a suitable replacement (if required).

BOP system instrumentation evaluated (except items that are part of or provide input to the NSSS and/or the main turbine control system) included the following fluid systems:

- Main Steam system (PBNP FSAR Section 10.1)
- Condensate and feedwater system (PBNP FSAR Section 10.1)
- Auxiliary Feedwater system (PBNP FSAR Section 10.2)
- Heater drain system (PBNP FSAR Section 10.1)
- Circulating water system (PBNP FSAR Section 10.1)
- Component cooling water system (PBNP FSAR Section 9.1)
- Condenser steam dump system (PBNP FSAR Section 7.7 and 10.1)
- Turbine generator system (PBNP FSAR Section 10.1)
- Extraction steam system (PBNP FSAR Section 10.1)
- Steam Generator blowdown system (PBNP FSAR Section 10.1)
- Spent fuel pool cooling and cleanup system (PBNP FSAR Section 9.9)
- Service Water System (PBNP FSAR Section 9.6)

The EPU evaluation of BOP instrumentation and controls demonstrated that, except as noted below, the design of BOP instruments, ranges, and setpoints remains acceptable for EPU operation.

The existing BOP indicated spans for indicators located at the Alternate Shutdown Instrumentation and Control (ASIC) panels for monitoring steam line pressure and steam generator (SG) level, as identified in FSAR Section 7.5.4.2, Indication and Controls Provided Outside the Control Room, are unaffected at EPU conditions.

The existing BOP indicators located at the Auxiliary Safety Instrumentation Panels (ASIPs) for monitoring SG wide range level as identified in FSAR page 7.5-5 are unaffected at EPU conditions.

The Regulatory Guide (RG) 1.97, Rev. 2 monitored variables, as identified in FSAR Section 7.6, Instrumentation Systems, and summarized in FSAR Table 7.6-1, remain bounding at EPU conditions, with the exception of main steam flow and main feedwater flow indication which will be re-ranged for EPU conditions.

Condensate and Feedwater System

The condensate and feedwater system evaluation is described in LR Subsection 2.5.5.4, *Condensate and Feedwater*. As a result of this evaluation, the following modifications will be implemented:

- To regain operating margin when the power uprate occurs on the main feedwater system, the following setpoints will be changed:
  - Feedwater pump low feedwater pump suction pressure, open low pressure heater bypass valve. (PC-2273, adjust controller setpoint).
  - Feedwater pump low feedwater pump suction pressure, trip main feedwater pump.
- Replace Main Feedwater flow transmitters for increased range.
- Upgrade Feedwater Regulating Valve (FRV) internal trim with a trim having a higher rated Cv value, replace the pneumatic operator, and replace the analog positioner with a digital positioner. The FRVs and the FRV bypass valves will continue to close on receipt of a SI signal, SG water level high-high signal, or a low Tavg signal with a reactor trip signal. These valves close on a SI signal as backups to the new Feedwater Isolation Valves (FIVs).
- Install new Feedwater Isolation Valves (FIVs) in the main feedwater line to each Steam Generator just outside containment. These normally open valves will close on receipt of a SI Signal to isolate main feedwater flow in the event of a steam line break inside containment.
- Change the Control Room indicator scale plates and recalibrate/rescale the instrument loops for Main Feedwater flow.
- Change the Control Room indicator scale plates and recalibrate/rescale the instrument loop for Main Feedwater pumps suction flow.
- Change the Control Room indicator scale plates and recalibrate/rescale the instrument loop for Heater Drain pump discharge flow.
- Increase the low level setpoint alarm for the Condensate Storage Tank.
- Change condenser low vacuum alarm setpoint.
- Recalibrate/rescale LEFM electronics used for plant calorimetric input and adjustment for new high flow alarm setpoint.

Main Steam System

The main steam system evaluation is described in LR Subsection 2.5.5.1, Main Steam. As a result of this evaluation, the following modifications will be implemented.

- Modify the electro-hydraulic control (EHC) system for opening sequence control of the HP Turbine Control Valves to convert from partial arc admission to full arc admission to the HP Turbine.

- Replace Main Steam flow transmitters for increased range.
- Replace HP Turbine Exhaust to MSR local pressure indicators.
- Change the Control Room indicator scale plates and recalibrate/rescale Main Steam flow loops.
- Recalibrate/rescale HP turbine gland steam supply pressure transmitters.
- Change the Control Room indicator scale plates and recalibrate/rescale HP Turbine first stage pressure transmitters and the following interlocks and inputs:
  - Permissive P2 – Auto-Rod withdrawal stop at low power
  - Permissive P5 – Steam Dump Interlocks
  - Permissive P7 – Block various trips at power
  - Permissive P20 – AMSAC (Anticipated Transient Without Scram [ATWS] mitigation system actuation circuitry)
  - Feedwater Control input

Extraction Steam System

As a result of this evaluation, the following modifications will be implemented.

- Recalibrate/rescale fifth point heater pressure transmitters.
- Recalibrate/rescale T-94A preseparator tank pressure transmitters.
- Recalibrate/rescale third point heater pressure transmitters.
- Recalibrate/rescale second point heater pressure transmitters.
- Recalibrate/rescale LP turbine crossover pressure transmitters.

Condenser Steam Dump System

The condenser steam dump system evaluation is described in LR subsection 2.5.5.2, Main Condenser. As a result of this evaluation, no modifications are required.

Auxiliary Feedwater System

The Auxiliary Feedwater (AFW) system will be modified from a shared system to a unitized system. The required AFW system modifications, including the I&C changes, are described in LR Section 2.5.4.5, Auxiliary Feedwater. This modification will require changes to the Main Control boards to modify some of the existing AFW controls and to add new controls for the new auxiliary feedwater pumps, valves, and instrumentation. The design modification process will verify that all I&C changes comply with both the existing I&C licensing basis and with the revised AFW licensing basis described in LR Section 2.5.4.5, Auxiliary Feedwater.

BOP Instrumentation and Controls Results

The changes to ranges and/or setpoints for BOP instruments will not change instrument or instrument loop functions. As a result of the EPU, there are no changes to the PBNP GDC-12 current licensing basis that the quantity and types of process instrumentation provided ensures safe and orderly operation of the plant nor will the changes affect separation, redundancy, or diversity of the instrumentation and controls discussed above.

Plant Computer

The plant computer (also referred to as the plant process computer system) is described in FSAR Section 7.5.1.4, Plant Process Computer System. Plant process computer system inputs that are affected by instrumentation scaling changes will be modified during the implementation phase of the EPU. However, the plant computer safety assessment and reactor thermal output functions as described in FSAR Section 7.5.1.4.a, Safety Assessment System, and 7.5.1.4.b, Feedwater Leading Edge Flow Measurement (LEFM) System, will not change as result of the EPU.

2.4.1.2.3.2 NSSS Analyses and Evaluations

The Extended Power Uprate (EPU) analyses identified additional instrumentation and trip setpoint changes that are required to ensure Departure from Nucleate Boiling (DNB), Reactor Coolant System (RCS) pressure, and secondary system pressure remain within the allowable design margins and the response to the design basis operational transients remain acceptable. These changes are described below. The EPU analyses determined that with the exception of the changes identified in this section, the NSSS instrumentation ranges, scaling, and setpoints used in the RPS, ESFAS, and reactor control instrumentation remained adequate for EPU. The specific changes to instruments necessary to meet the EPU analyses are described in the following sections.

PBNP previously determined that certain Allowable Values (AV) in existing Technical Specification (TS) Tables 3.3.1-1 (RPS Instrumentation) and 3.3.2-1 (ESFAS Instrumentation) should be revised to include instrument uncertainties (Reference PBNP letter to the NRC, NRC 2007-0061, dated 8/8/07). This commitment has been accounted for in the determination of revised AVs in the TS Tables for functions that are affected by the EPU changes, as well as revised AVs for other functions not affected by EPU, to make the TS Tables consistent with the intent of an AV and to resolve the commitment.

In addition to revising AVs for various RPS and ESFAS functions, TS Tables 3.3.1-1 and 3.3.2-1 and their associated Bases have been revised to incorporate other changes recommended by a joint NRC and industry effort to clarify the application of setpoint methodology for Limiting Safety System Settings (LSSS), as defined in 10 CFR 50.36. These changes include addition of a column for the "Nominal Trip Setpoint" (NTSP) for those functions that have an adjustable setpoint, and addition of new notes that define actions to be taken during periodic surveillances for the Channel Operational Test (COT) and the Channel Calibration based on the as-found setting relative to as-left and as-found setting tolerances.

A discussion of the methods used to develop AVs, NTSPs, as-left tolerances, and as-found tolerances for use in TS Tables 3.3.1-1 and 3.3.2-1 is included in Attachment E to this submittal.

Overview of RPS/ESFAS Technical Specification Changes

This section provides an overview of proposed changes to TS Tables 3.3.1-1 (RPS Instrumentation) and 3.3.2-1 (ESFAS Instrumentation). Further details are provided in a later section. The RPS/ESFAS tables are being revised to:

1) Add a new column titled "Nominal Trip Setpoint";

2) Add two new Notes that apply to selected RPS/ESFAS Functions that require Channel Operational Tests and Channel Calibrations; and

3) Change Allowable Values in the tables for various RPS/ESFAS Functions.

4)

<u>New "Nominal Trip Setpoint" column</u>

Westinghouse plants that follow NUREG-1431, Standard Technical Specifications – Westinghouse Plants, are allowed to adopt a "multiple column" format for RPS and ESFAS Function Tables 3.3.1-1 and 3.3.2-1 that lists both an Allowable Value and a Nominal Trip

Setpoint for functions with adjustable setpoints. Non-adjustable functions that rely on open-close contacts will have "NA" in both columns.

The NTSP is the ideal field setting for the adjustable bistable setting, applied during routine channel surveillance/calibration. Both the as-left and as-found tolerance limits found in the calibration procedures are centered around the NTSP. By including this column in TS Tables 3.3.1-1 and 3.3.2-1, the TS control the ideal field setting, and future changes to the field setting will require prior NRC approval. However, new Note 2 (see below) does permit field settings more conservative than the NTSP, provided the as-left and as-found tolerances are applied to the actual field setpoint implemented in the calibration procedures that confirm channel performance.

Unlike the AV, the NTSP is a fixed value, rather than a not-to-exceed limit. Therefore, the inequality symbol assigned to AVs is not assigned to the respective NTSP values in the specifications.

<u>New Notes applied to RPS/ESFAS Function COT and Channel Calibrations</u>

The following notes are being added to TS Tables 3.3.1-1 and 3.3.2-1:

Note 1. If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Note 2. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in FSAR Section 7.2.

Both Notes will be applied to the COT and Channel Calibration surveillances for those RPS/ESFAS Functions shown in marked-up Tables 3.3.1-1 and 3.3.2-1. The Notes do not apply to adjustable Permissives and Interlocks.

Appendix E, Supplement to LR Section 2.4.1, provides additional information on the calibration and surveillance criteria applied to the NTSP, and how these requirements are incorporated into the design basis.

<u>Allowable Value Changes</u>

Allowable Value changes are proposed in this submittal for two reasons:

- EPU safety analyses revised the Analytical Limits for some RPS/ESFAS Functions. This in turn required recalculating the Limiting Trip Setpoint (LTSP) for those Functions that protect the safety analyses. The LTSP is separated from the Analytical Limit by the total channel uncertainty, determined by combining all appropriate uncertainties. For these

Functions, the new AVs are the revised LTSP, and may include some rounding in the conservative direction.

- Existing non-EPU-related AVs were recalculated that were found to be inconsistent with the setpoint calculations and inconsistent with the TS intent of an AV. The proposed AVs in this group are now consistent with the intent of RIS 2006-0017, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing And Calibration Of Instrument Channels."

Classification of Setpoints by Function

RPS and ESFAS setpoints are classified into one of the following three categories based on the individual setpoint function and whether the setpoint is credited in the safety analyses to protect a safety limit:

Category A — Setpoints for protective functions that have defined Analytical Limits in the safety analyses to protect safety limits. Analytical Limits establish not-to-exceed setpoint values that assure that a protective action occurs within the safety analysis assumptions to protect the reactor core and reactor coolant system Safety Limits (SL) identified in PBNP Technical Specification 2.1. Setpoints in this category are referred to as primary trips. Category A setpoints are SL-related LSSSs in the context of RIS 2006-0017. The Allowable Value for Category A setpoints is based on the Limiting Trip Setpoint to protect the safety analyses, considering all channel uncertainties.

Category B — Setpoints for protective functions that lack an Analytical Limit in the safety analyses. These protective functions are included in the protection system for defense-in-depth, but the functions are not specifically credited in the safety analyses. The setpoints for these protective functions are referred to as backup or anticipatory trips. Category B setpoints are non-SL related LSSSs in the context of RIS 2006-0017. The Allowable Value for Category B setpoints is typically based on the Limiting Trip Setpoint to protect a Process Limit rather than an Analytical Limit.

Category C — Setpoints for automatically removing protection system operating bypasses. These setpoints have no Analytical Limit in the safety analyses that protects a safety limit. Operating bypasses may also be referred to in the TSs as interlocks, permissives, or blocks. Setpoints for removing operating bypasses are not considered a protection system protective action (reactor trip or ESF actuation) that protects a safety limit. Category C setpoints are non-SL related in the context of RIS 2006-0017. The Allowable Value for Category C setpoints is typically based on the as-found acceptance criteria for the Nominal Trip Setpoint, instead of either an Analytical Limit or a Process Limit.

Summary Tables of Proposed Changes

Table 2.4.1-1, Summary of EPU-Related RPS Functions, Allowable Values, and Analytical Limits, and Table 2.4.1-2, Summary of EPU-Related ESFAS Functions, Allowable Values, and Analytical Limits, are provided at the end of this section. The tables identify:

- the RPS [or ESFAS] Function
- TS Table Function number
- Category (defined above)
- Is the Function a SL-related Limiting Safety System Setting (LSSS), in the context of RIS 2006-0017? If the answer is NO, an Analytical Limit does not exist for the setpoint.
- the proposed value in the TS Function table "Allowable Value" column

If the function is a Category A setpoint (a SL-related LSSS), the table also includes:

- the current Analytical Limit
- the EPU Analytical Limit
- the related safety analyses

The Tables do not include the NTSP listed in the TS markups in LAR Attachment 2. However, the NTSP is provided in the following section that discusses each Function that is affected by this amendment.

2.4.1.2.3.2.1 Reactor Protection System

The design bases of the PBNP Reactor Protection System (RPS) are described in FSAR Section 7.2.1, Design Bases, and includes a listing of the reactor trip functions, the purpose of each trip, and any associated protection and control permissives. The RPS automatically trips the reactor to protect against reactor coolant system damage caused by high system pressure and to protect the reactor core against fuel rod cladding damage caused by a departure from nucleate boiling. The basic reactor tripping philosophy is to define a region of power and coolant temperature and pressure conditions allowed by the primary trip functions (overpower ΔT trip, overtemperature ΔT trip, and power range flux overpower trips). The allowable operating region within these trip settings is provided to prevent any combination of power, temperature, and pressure that would result in a departure from nucleate boiling with all reactor coolant pumps in operation.

Reactor trip functions such as high and low pressurizer pressure trips, low RCS flow trip, and steam-generator low-low water level trip are additional primary trip functions credited in the safety analysis for specific accident conditions and mechanical failures. Primary trips are distinguished by having specific Analytical Limits in the safety analyses to protect safety limits. The Analytical Limit provides the starting point for determining the primary trip setpoint value. Other reactor trip functions such as a high pressurizer water level trip, turbine trip, SI trip, source and intermediate range neutron flux trips, and manual trip are provided to back up the primary trip functions and are not specifically credited in the safety analyses. As a result, backup trips do not have explicit Analytical Limits to anchor the trip setpoint values.

The following are descriptions of the RPS instrumentation and setpoint changes necessary to ensure the RPS will continue to satisfy its design functions at EPU conditions.

RCS Average Temperature Instrumentation Range Change

LR Section 2.8.5.0, Accident and Transient Analyses, made recommendations for the T_h, T_c, T_{avg} and ΔT instrument ranges and setpoints to ensure the instrumentation would provide the required indication, core DNB protection, and plant response during accidents and transients over the entire range of operation at EPU conditions. The current range of the T_h and T_c instruments (500°F– 650°F) and ΔT instruments (0 - 100°F) satisfies the recommended ranges.

However, the T_{avg} instrument channels, including indicators, will be recalibrated for a higher range to meet the LR Section 2.8.5.0 recommendations, as follows:

- T_{avg} range revised from 520°F – 620°F to 530°F – 630°F

Reactor Trip Function Changes for EPU

Power Range Neutron Flux – High Reactor Trip

For EPU, the Analytical Limit for the power range neutron flux – high reactor trip is decreased to 116% RTP from the current value of 118% RTP for the Rod Withdrawal at Power event. The proposed TS AV for this function is based on the LTSP established by calculation to avoid exceeding the EPU Analytical Limit, taking all instrument uncertainties into account. The following change is proposed for the AV in TS Table 3.3.1-1:

Parameter	Analytical Limit		TS Allowable Value		Nominal Trip Setpoint
	Current	EPU	Current	EPU	Current and EPU
Power Range Neutron Flux - High	118% RTP	116% RTP	≤ 108% RTP	≤ 109% RTP	107% RTP

The NTSP has design margin with respect to the AV.

Overtemperature ΔT (OTΔT) Reactor Trip

Typically the values for the OTΔT reactor trip setpoints constants are listed in the cycle-specific Core Operating Limits Report (COLR) for each fuel cycle. For the initial EPU startup, the OTΔT trip setpoint will be recalibrated with OTΔT constants changed as follows:

Parameter	Current	EPU
Analytical Limit	1.255	1.295
Constant K1	1.16	1.203
Constant K2	0.0149/°F	0.016/°F
Constant K3	0.00072/psi	0.000811/psi

Overpower ΔT (OPΔT) Reactor Trip

As with OTΔT reactor trip setpoint, the values for the OPΔT trip setpoints constants are listed in the cycle specific COLR for each fuel cycle. The accident and transient analyses determined the rate sensitive temperature portion of the setpoint and the f(ΔI) function are not necessary for the OPΔT trip circuit to provide the required protection for maintaining the fuel design limits. The OPΔT trip setpoint constants changed as follows:

Parameter	Current	EPU
Analytical Limit	1.14	1.165
Constant K4	1.10	1.118
Constant K5	0.0262/°F	0.0/°F
Constant K6	0.00103/psi for Tavg ≥ T' 0.0/psi for Tavg < T'	0.00123/psi for Tavg ≥ T' 0.0/psi for Tavg < T'
Constant T'	569°F	576°F

Pressurizer Pressure - Low Reactor Trip

For EPU, the Analytical Limit for the pressurizer pressure - low reactor trip is 1840 psig, an increase from the current value of 1815 psig for the OPTOAX code analysis. The proposed AV is based on the LTSP established by calculation to avoid exceeding the new Analytical Limit, taking all instrument uncertainties into account. The following change is proposed for the AV in TS Table 3.3.1-1:

Parameter	Analytical Limit		TS Allowable Value		Nominal Trip Setpoint
	Current	EPU	Current	EPU	Current and EPU
Pressurizer Pressure Low	1815 psig	1840 psig	Footnote (h)	≥ 1860 psig	1925 psig

The EPU AV is for operation at 2250 psia. Currently, a separate AV is also stated in TS 3.3.1 Function 7.a Footnote (h) for operation at 2000 psia. The proposed AV for EPU is for operation only at 2250 psia. Footnote (h) and the AV for operation at 2000 psia will be deleted.

The NTSP has design margin with respect to the AV.

Pressurizer Pressure - High Reactor Trip

For EPU, the Analytical Limit for the pressurizer pressure - high reactor trip is 2403 psig for the Loss of External Electrical Load/Turbine Trip analysis, a decrease from the current Analytical Limit of 2410 psig. The proposed AV is based on the Limiting Trip Setpoint established by calculation to avoid exceeding the new Analytical Limit, including all instrument uncertainties. For this setpoint, the proposed AV for EPU is the same as the current AV for 2250 psia operation and no TS change is necessary.

Parameter	Analytical Limit		TS Allowable Value		Nominal Trip Setpoint
	Current	EPU	Current	EPU	Current and EPU
Pressurizer Pressure High	2410 psig	2403 psig	Footnote (i)	≤ 2385 psig	2365 psig

The EPU AV is for operation at 2250 psia. Currently, a separate AV is also stated in TS 3.3.1 Function 7.b Footnote (i) for operation at 2000 psia. The proposed AV for EPU only applies for operation at 2250 psia. Footnote (i) and the AV for operation at 2000 psia will be deleted.

The NTSP has design margin with respect to the AV.

Steam Generator Narrow Range Water Level Low-Low Reactor Trip

The accident and transient analyses have determined that the Analytical Limit utilized in the Loss of Normal Feedwater/Loss of AC Power events is changed for the EPU. For these events, the steam generator water level low-low reactor trip is credited as a primary protection function with an Analytical Limit. The steam generator water level low-low reactor trip Analytical Limit for the Loss of Normal Feedwater/Loss of AC Power events is increasing from 17% to 20.0% of narrow range span (NRS). To account for the Analytical Limit change for EPU conditions and instrument channel uncertainties, the TS Allowable Value will increase from ≥ 20.0% to ≥ 29.3%.

Parameter	Analytical Limit		TS Allowable Value		Nominal Trip Setpoint	
	Current	EPU	Current	EPU	Current	EPU
Steam Generator Narrow Range Water Level Low-Low (NRS)	17.0%	20.0%	≥ 20.0%	≥ 29.3%	25.0%	30.0%

The NTSP has design margin with respect to the AV.

Reactor Trip System Interlock/Permissive Changes for EPU

Power Range Neutron Flux, P-8

The power range neutron flux P-8 interlock is an operating bypass function that is not specifically credited in the safety analyses. The interlock safety function is to automatically reinstate single loop loss of coolant flow reactor trips when the trip functions are required on increasing power. For EPU, the P-8 setpoint needs to be lowered from the current nominal 50% RTP to protect the reactor during a partial loss of flow event. Calculation CN-TA-08-52, "Partial Loss of Flow Permissive P-8 Setpoint Analysis for Point Beach Extended Power Uprate", requires that the P-8 setpoint limit reactor power to ≤ 45% RTP when the P-8 permissive block is in effect. Based on a 10% instrument uncertainty assumed in CN-TA-08-52, the P-8 Nominal Trip Setpoint is to be lowered to 35% RTP. The actual instrument uncertainty is less than this 10% assumption.

Based on a P-8 NTSP of 35%, and the proposed AV is established by calculation as the upper as-found limit for the 35% RTP NTSP. The AV for the TS is selected based on the NTSP and the expected loop performance (upper as-found tolerance) between calibrations, rounded. The following change is proposed for the P-8 AV:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	EPU	Current	EPU
Power Range Neutron Flux, P-8	< 50% RTP	≤ 38% RTP	49% RTP	35% RTP

Power Range Neutron Flux, P-9

The power range neutron flux P-9 interlock is an operating bypass function that is not specifically credited in the safety analyses. The safety function of the interlock is to remove (unblock) the operating bypass automatically to reinstate the reactor trip on turbine trip function when power is above the ability of the steam dump system to prevent a reactor trip on a load rejection. The nominal setpoint at which the trips are reinstated is being revised by EPU to be a function of full power Tavg. A nominal P-9 permissive setpoint of 50% RTP is adequate at operation with a full design power Tavg of ≥ 572°F and during end-of-cycle coastdown operation. The nominal P-9 permissive setpoint must be reduced to 35% RTP during operation when full design power Tavg is below 572°F.

The AV for the TS is selected based on the NTSP and the expected loop performance (upper as-found tolerance) between calibrations, rounded. The following changes are proposed for the P-9 Allowable Value:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	EPU	Current	EPU
Power Range Neutron Flux, P-9	< 50% RTP	Footnote (h) ≤ 38% RTP * or ≤ 53% RTP *	49% RTP	Footnote (i) 35% RTP ** or 50% RTP **

* ≤ 38% RTP for full design power Tavg < 572°F or
≤ 53% RTP for full design power Tavg ≥ 572°F and end-of-cycle coastdown

** ≤ 35% RTP for full design power Tavg < 572°F or
≤ 50% RTP for full design power Tavg ≥ 572°F and end-of-cycle coastdown

Reactor Trip Function Changes (Non-EPU)

The following changes to RPS setpoints are included to revise the Allowable Values in TS 3.3.1 at current conditions. These changes are also acceptable for EPU conditions.

Power Range Neutron Flux – Low Reactor Trip

The power range neutron flux – low reactor trip is a primary trip for safety analyses of subcritical rod withdrawal, rod ejection, and steam line break (outside containment) events. The Analytical Limit of 35% RTP applies to both current operation and EPU. The proposed AV is established by calculation to avoid exceeding the Analytical Limit to account for calculated uncertainties. The following change is proposed for the AV:

Parameter	Analytical Limit	TS Allowable Value		Nominal Trip Setpoint
	Current and EPU	Current	Proposed	Current and Proposed
Power Range Neutron Flux - Low	35% RTP	≤ 25% RTP	≤ 28% RTP	20% RTP

The NTSP has design margin with respect to the AV.

Intermediate Range Neutron Flux

The intermediate range high flux reactor trip is a backup trip that is not specifically credited in the safety analyses. Therefore, the trip function lacks an analytical limit upon which to base the AV. The AV for the TS is selected based on the NTSP and the expected loop performance between calibrations, rounded. Due to the nature of this process, the expected performance upper limit was based on setting tolerance, measurement and test equipment accuracy, and a drift allowance. This calculation method provides a more conservative AV than assuming a process limit of 100% instrument span. The following change is proposed for the AV:

Parameter	TS Allowable Value		Nominal Trip Setpoint
	Current	Proposed	Current and Proposed
Intermediate Range Neutron Flux	≤ 40% RTP	≤ 43% RTP	25% RTP

The NTSP has design margin with respect to the AV.

Source Range Neutron Flux

The source range high flux reactor trip is a backup trip that is not specifically credited in the safety analyses. Therefore, the trip function lacks an analytical limit upon which to base the AV. The AV for the TS is selected based on the Nominal Trip Setpoint plus a combination of as-left tolerance and expected rack drift between calibrations. The

NTSP is selected to remain within the upper range limit (the Process Limit) less uncertainties. The following changes are proposed to include a value for the AV instead of the phrase "within span of instrumentation" consistent with NUREG-1431, and to add the NTSP:

Parameter	TS Allowable Value		Nominal Trip Setpoint
	Current	Proposed	Current and Proposed
Source Range Neutron Flux	"within span of instrumentation"	≤ 4.0 E5 cps	2.0 E5 cps

<u>Pressurizer Water Level - High</u>

The pressurizer water level high reactor trip is a backup trip that is not specifically credited in the safety analyses and does not have an Analytical Limit. A Process Limit of 100% of the instrument span is used to calculate the AV. The following change is proposed for the AV:

Parameter	TS Allowable Value		Nominal Trip Setpoint
	Current	Proposed	Current and Proposed
Pressurizer Water Level - High	≤ 95%	≤ 85%	80%

The NTSP has design margin with respect to the AV.

<u>Reactor Coolant Flow - Low</u>

The reactor coolant flow low reactor trip is a primary trip that is credited in the Complete Loss of Flow safety analysis with an Analytical Limit of 87% of minimum measured flow (MMF). The AV is based on the Limiting Trip Setpoint established by calculation to avoid exceeding the Analytical Limit, taking all instrument uncertainties into account. Both the Analytical Limit and AV are unchanged by EPU. Adding the NTSP for this function to TS Table 3.3.1-1 will conform to the "multiple column" format allowed by NUREG-1431:

Parameter	Analytical Limit	TS Allowable Value	Nominal Trip Setpoint
	Current and Proposed	Current and Proposed	Current and Proposed
Reactor Coolant Flow - Low	87% MMF	≥ 90%	93%

The NTSP has design margin with respect to the AV.

Undervoltage Bus A01 & A02

The Undervoltage Bus A01 & A02 reactor trip is evaluated in the Complete Loss of Flow safety analysis as a backup trip that is bounded by the Reactor Coolant Flow - Low reactor trip (see Section 2.8.5.3). The NTSP for the undervoltage function is being added to TS Table 3.3.1-1 and will conform to the "multiple column" format allowed by NUREG-1431:

Parameter	TS Allowable Value	Nominal Trip Setpoint
	Current and Proposed	Current and Proposed
Undervoltage Bus A01 & A02	≥ 3120 V	3170 V

Underfrequency Bus A01 & A02

The Underfrequency Bus A01 & A02 function trips the Reactor Coolant Pump breakers, but does not provide a direct reactor trip. If underfrequency trips the RCP breakers, the RCP breaker position reactor trip (separate Function 10 in TS Table 3.3.1-1) may occur, but this is a backup reactor trip that is not credited in any safety analysis. The NTSP for the underfrequency function (Function 12 in Table 3.3.1-1) that could lead to an RCP breaker trip is being added to TS Table 3.3.1-1 and will conform to the "multiple column" format allowed by NUREG-1431:

Parameter	TS Allowable Value	Nominal Trip Setpoint
	Current and Proposed	Current and Proposed
Underfrequency Bus A01 & A02	≥ 55 Hz	57 Hz

Steam Generator Water Level - Low

The steam generator water level low signal, coincident with steam flow/feedwater flow mismatch, provides a backup reactor trip that is not specifically credited in the safety analyses and therefore does not have an Analytical Limit. The low level backup trip function is necessary in the event that a control/protection interaction failure of the SG level low-low reactor trip function disables the reactor trip on low-low level. A Process Limit of 0%, the low limit of the instrument span, is used to calculate the Allowable Value considering all instrument uncertainties.

The AV is proposed to be changed as follows:

Parameter	TS Allowable Value		Nominal Trip Setpoint
	Current	Proposed	Current and Proposed
SG Water Level - Low	"NA"	≥ 10%	30%

For this function, Point Beach has chosen to set the NTSP conservatively higher at 30% to be consistent with the Steam Generator Water Level Low-Low Reactor Trip setpoint. The NTSP has design margin with respect to the AV.

Steam Flow/Feedwater Flow Mismatch

The reactor trip on SG Water Level Low coincident with Steam Flow/Feedwater Flow Mismatch is a backup function that is not specifically credited in the safety analyses, does not have an Analytical Limit, and is not affected by EPU. The NTSP for the Steam Flow/Feedwater Flow Mismatch function is being added to TS Table 3.3.1-1 and will conform to the "multiple column" format allowed by NUREG-1431:

Parameter	TS Allowable Value	Nominal Trip Setpoint
	Current and Proposed	Current and Proposed
Steam Flow / Feedwater Flow Mismatch	≤ 1 E6 lb_m/hr	0.8 E6 lb_m/hr

Reactor Trip System Interlock/Permissive Changes (Non-EPU)

Intermediate Range Neutron Flux, P-6

The P-6 interlock is an operating bypass (permissive) function that is not specifically credited in the safety analyses. The safety function of the interlock is to remove (unblock) the P-6 operating bypass automatically to reinstate the source range neutron flux reactor trip when intermediate range power decreases to the setpoint. This unblock safety function lacks an analytical limit upon which to base the Allowable Value. The AV for the TS is selected based on the lower as-found limit for the NTSP, rounded. The following changes are proposed for the AV and the NTSP:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	Proposed	Current	Proposed
Intermediate Range Neutron Flux, P-6	> 1E-10 amp	≥ 4E-11 amp	1.5E-10 amp	1E-10 amp

The AV will ensure that the source range neutron flux reactor trip function is restored on decreasing power during shutdown. The Nominal Trip Setpoint change establishes the NTSP conservatively above the Allowable Value.

Power Range Neutron Flux, P-7

The power range neutron flux P-7 interlock is an operating bypass (permissive) function that is not specifically credited in the safety analyses. The safety function of the interlock is to remove (unblock) the P-7 operating bypass automatically to reinstate reactor trips on increasing power of approximately 10% RTP that are not required at lower power levels. The proposed AV is based on the upper as-found limit for the P-7 Nominal Trip Setpoint. The following change is proposed for the Allowable Value:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	Proposed	Current	Proposed
Power Range Neutron Flux, P-7	< 10% RTP	≤ 13% RTP	9.5%	10%

The proposed NTSP will continue to assure that the required reactor trip functions are automatically reinstated below the lower range limit of the power range instrumentation. The proposed AV reflects the upper as-found limit for the proposed NTSP.

Turbine Impulse Pressure, P-7

The turbine impulse pressure P-7 interlock is an operating bypass function that is not specifically credited in the safety analyses. The safety function of the interlock is to remove (unblock) the P-7 operating bypass automatically to reinstate reactor trips on increasing turbine power of approximately 10% that are not required at low power levels. The proposed AV is based on the upper as-found limit established by calculation for the nominal turbine impulse pressure P-7 NTSP. The following changes are proposed for the AV and NTSP:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	Proposed	Current	Proposed
Turbine Impulse Pressure, P-7	< 10% turbine power	≤ 12.8% turbine power	8% turbine power	10% turbine power

The proposed AV will assure that the required reactor trip functions are automatically reinstated at low power. The proposed AV is the upper as-found limit established by calculation for the NTSP.

Note that for EPU the input from turbine first stage pressure input will be rescaled to actuate the P-7 permissive at the value consistent with the revised 0% - 100% turbine first stage pressure range for the EPU condition.

Power Range Neutron Flux, P-10

The power range neutron flux P-10 interlock is an operating bypass function that is not specifically credited in the safety analyses. The P-10 safety function is to automatically reinstate two reactor trips (Intermediate Range Neutron flux and Power Range Neutron Flux – Low) that are required at low power levels on a decreasing power of approximately 9% RTP. Both upper and lower AVs are included for consistency with the STS. However, the lower AV is the only value that protects the P-10 safety function to automatically remove (unblock) the bypass. The proposed AVs are based on the upper and lower as-found limits established by calculation for the P-10 Nominal Trip Setpoint.

The following changes are proposed for the Allowable Value:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	Proposed	Current	Proposed
Power Range Neutron Flux, P-10	> 8% RTP and < 10% RTP	≥ 6% RTP and ≤ 12% RTP	8.5%	9%

The proposed NTSP will continue to assure that the required reactor trip functions are automatically reinstated. The proposed AVs reflect the upper and lower as-found limits that are established by calculation for the proposed NTSP.

2.4.1.2.3.2.2 Engineered Safety Feature Actuation System

The engineered safety feature actuation system (ESFAS) actuates various Engineered Safety Features (ESF) to provide protection against the release of radioactive materials in the event of a loss-of-coolant accident or a secondary line break accident. The ESF components also function to maintain the reactor in a shutdown condition. They also provide sufficient core cooling to limit the extent of fuel and fuel cladding damage and to ensure the integrity of the containment structure. These functions rely on automatic initiation by the ESFAS and associated instrumentation and controls. ESFAS instrumentation, Analytical Limits, and settings being changed for EPU and non-EPU reasons are discussed in the following sections.

EPU Changes

Main Steam Flow and Feedwater Flow Instrumentation Range Changes

The current main steam flow and main feedwater flow transmitters require changes to support EPU. The transmitters are currently calibrated with a flow range of $0 - 4.0x10^6$ lb_m/hr. The upper range limit is near the predicted EPU nominal steam flow of $3.7x10^6$ lb_m /hr with a feedwater temperature of 390 °F. The main steam and main feedwater flow transmitters will be recalibrated for a range of 0 - $5.0x10^6$ lb_m /hr. The expanded range ensures that the steam flow indication will continue to meet the required Regulatory Guide 1.97 range of 110% of design flow, plus provide additional scaling to ensure steam line isolation on high-high steam flow will occur within the flow instrumentation range.

Steam Line Pressure Low – Safety Injection (SI)

For EPU safety analyses, safety injection actuation on low steam line pressure is assumed to occur at 410 psia (395 psig) in the reactor core analyses for steam line failures at hot zero power and full power. This is an increase from the Analytical Limit of 335 psia (320.3 psig) at the current power level. The proposed AV is established by calculation to avoid exceeding the EPU Analytical Limit assuming all instrument uncertainties. The following change is proposed for the AV:

Parameter	Analytical Limit		TS Allowable Value		Nominal Trip Setpoint
	Current	EPU	Current	EPU	Current and EPU
Steam Line Pressure Low - SI	320 psig	395 psig	≥ 500[c] psig	≥ 520[c] psig	530 psig

The NTSP has design margin with respect to the AV.

Note (b) in the Applicable Modes column of Table 3.3.2-1 for this Function is modified to revise the pressurizer pressure from ">1800 psig" to ">2000 psig" at which the modes apply. This change is made to coincide with a change to the SI Block – Pressurizer Pressure function (Function 7 in Table 3.3.2-1) NTSP from 1775 psig to

2000 psig to automatically remove the SI block on increasing pressurizer pressure during plant startup. Below this pressure, the manual SI Block function prevents the SI Pressurizer Pressure – Low signal from occurring during normal plant shutdown/cooldown.

Note (c) referenced in the Allowable Value for this function is also being revised to change the lead time constant (t_1) for the lead/lag controller from "≥ 12 seconds" to "≥ 18 seconds". This change is being implemented to obtain acceptable steam line break analysis results during operation at EPU conditions.

<u>High and High-High Steam Line Flow – Steam Line Isolation (SLI)</u>

The EPU safety analysis for core response to steam line failures at hot zero power determined that the process limit for the high-high steam line flow input to steam line isolation is 5.0 E6 lb_m/hr based on the setpoint remaining within the upper range limit of the steam flow instrumentation. In addition, the EPU accident analysis for the steam line break mass and energy release analysis (outside containment) determined that the analytical limit for the high steam line flow input to steam line isolation is 1.07 E6 lb_m/hr. The Analytical/Process Limit changes and their effect on the AVs and NTSPs for both functions are shown in the following table.

Parameter	Analytical/Process Limit		TS Allowable Value		Nominal Trip Setpoint	
	Current	EPU	Current	EPU	Current	EPU
Steam Line Isolation High-High Steam Flow (in lb/hr)	4.0 E6 @ 806 psig	5.0 E6 @ 586 psig	≤ Δp corresponding to 4.0 E6 lb/hr @ 806 psig	≤ Δp corresponding to 4.9 E6 lb/hr @ 586 psig	≤ Δp corresponding to 3.86 E6 @ 806 psig	≤ Δp corresponding to 4.85 E6 @ 586 psig
Steam Line Isolation High Steam Flow (in lb/hr)	0.97 E6 @ 1005 psig	1.07 E6 @ 1005 psig	≤ Δp corresponding to 0.66 E6 lb/hr @ 1005 psig	≤ Δp corresponding to 0.8 E6 lb/hr @ 1005 psig	≤ Δp corresponding to 0.47 E6 @ 1005 psig	≤ Δp corresponding to 0.52 E6 @ 1005 psig

The NTSP have design margin with respect to the AV.

<u>AFW Pump Start on Steam Generator Narrow Range Water Level Low-Low</u>

The accident and transient analyses have determined that the Analytical Limit in the Loss of Normal Feedwater/Loss of AC Power events for AFW pump start is increased to 20% of narrow range span (NRS) for the EPU. The existing Analytical Limit for AFW initiation for a Loss of Normal Feedwater/Loss of AC Power event is 17% NRS. To account for the Analytical Limit change for EPU conditions and instrument channel uncertainties, the TS Allowable Value will increase from ≥ 20% to ≥ 29.3%, with a corresponding increase in the Nominal Trip Setpoint from 25% to 30%.

Parameter	Analytical Limit		TS Allowable Value		Nominal Trip Setpoint	
	Current	EPU	Current	EPU	Current	EPU
Steam Generator Narrow Range Water Level Low-Low	17%	20%	≥ 20%	≥ 29.3%	25%	30%

The NTSP has design margin with respect to the AV.

AFW Pump Suction Transfer on Suction Pressure Low

New Function 6.e, AFW Pump Suction Transfer on Suction Pressure Low, is being added to TS Table 3.3.2-1 for EPU, as discussed in Section 2.1 of Enclosure 2 to NextEra letter to NRC dated June 17, 2009, License Amendment Request 261 Supplement 1, Extended Power Uprate. The function will automatically transfer the AFW pump suction source to service water to provide continuity of water supplied to the AFW pumps in the event of a loss of the normal Condensate Storage Tank water source that results in low suction pressure. The Analytical Limit, AV, and NTSP for the new function are as follows:

Parameter	Analytical Limit	TS Allowable Value	Nominal Trip Setpoint
AFW Pump Suction Transfer on Suction Pressure Low	5.241 psig	≥ 5.7 psig	6.0 psig

Removal of Condensate Isolation Function from Table 3.3.2-1

For EPU, the Condensate Isolation function (Function 7 in TS Table 3.3.2-1) will no longer be credited to limit secondary system pipe break mass and energy releases to containment and will be removed from TS 3.3.2. The addition of Feedwater Isolation Valves for EPU that automatically close on an SI signal will replace the Condensate Isolation function to provide redundant isolation (with MFRV and bypass valve isolation) of the main feedwater headers during safety injection.

Non-EPU Changes

The following proposed changes to ESFAS setpoints are necessary to make the Allowable Values in TS 3.3.2 at current conditions consistent with the AV calculations and TS presentation adopted in this submittal. These changes are also acceptable at EPU conditions.

Containment Pressure High - SI

Safety injection initiation on high containment pressure is credited in the safety analyses for a steam line break inside containment. The safety analysis Analytical

Limit for the function is 6 psig increasing and is not changing for EPU. The proposed AV is based on the Limiting Trip Setpoint established by calculation to avoid exceeding the Analytical Limit, considering all instrument channel uncertainties. The following change is proposed for the AV:

Parameter	Analytical Limit	TS Allowable Value		Nominal Trip Setpoint
		Current	Proposed	Current and Proposed
Containment Pressure High - SI	6 psig	≤ 6 psig	≤ 5.3 psig	5 psig

The NTSP has design margin with respect to the AV.

Pressurizer Pressure Low - SI

Safety injection initiation on low pressurizer pressure is credited in the LOCA, steam line break, and steam generator tube rupture core response safety analyses. The most restrictive (highest) safety analysis Analytical Limit for the function is 1648 psig decreasing. However, the lower range limit for the pressurizer pressure narrow range instruments 1700 psig, which is above this limit. Therefore, the AV for this function is based on the more restrictive lower range limit (a Process Limit) to assure the function occurs within the instrument range. The proposed AV is established by calculation to prevent the field setpoint descending below the lower range limit when all appropriate uncertainties are considered. The following change is proposed for the Allowable Value:

Parameter	Process Limit	TS Allowable Value		Nominal Trip Setpoint
		Current	Proposed	Current and Proposed
Pressurizer Pressure Low - SI	1700 psig	≥ 1715 psig	≥ 1725 psig	1735 psig

The NTSP has design margin with respect to the AV.

In addition, Note (a) in the Applicable Modes column of Table 3.3.2-1 for this Function is modified to revise the pressurizer pressure from ">1800 psig" to ">2000 psig" at which the modes apply. This change is made to coincide with a change to the SI Block – Pressurizer Pressure function (Function 7 in Table 3.3.2-1) NTSP from 1775 psig to 2000 psig to automatically remove the SI block on increasing pressurizer pressure during plant startup. Below this pressure, the manual SI Block function prevents the SI Pressurizer Pressure – Low signal from occurring during normal plant shutdown/ cooldown.

Containment Pressure High High – Containment Spray

Containment spray initiation on high-high containment pressure is credited in the containment integrity safety analysis for the steam line break inside containment. The safety analysis analytical limit for the function is 30 psig increasing and is not changing for EPU. The proposed AV is based on the Limiting Trip Setpoint established by calculation to remain below the Analytical Limit. The following change is proposed for the AV:

Parameter	Analytical Limit	TS Allowable Value		Nominal Trip Setpoint
		Current	Proposed	Current and Proposed
Containment Pressure High High - CS	30 psig	≤ 30 psig	≤ 28 psig	25 psig

The NTSP has design margin with respect to the AV.

Containment Pressure High High - SLI

Steam line isolation on high-high containment pressure is not specifically credited in the containment integrity safety analyses for a steam line break inside containment. Therefore, the trip function lacks an analytical limit upon which to base the Allowable Value. The proposed AV is based on maintaining the NTSP as-found value below the historical Process Limit previously established for this setpoint (20 psig), minus instrument loop uncertainties. The proposed AV is established by calculation. The following change is proposed for the AV:

Parameter	Process Limit	TS Allowable Value		Nominal Trip Setpoint
		Current	Proposed	Current and Proposed
Containment Pressure High High - SLI	20 psig	≤ 20 psig	≤ 18 psig	15 psig

The proposed AV change establishes the upper limit on the as-found value for the NTSP to remain within the 20 psig Process Limit for steam line isolation on high-high containment pressure.

The NTSP has design margin with respect to the AV.

Low Tavg Interlock

The low Tavg interlock, coincident with safety injection and high steam flow, is credited in the safety analysis for the steam line break release analysis (outside containment) to generate a steam line isolation signal. The proposed AV is established by calculation to avoid exceeding the analytical limit, including all instrument uncertainties. The following change is proposed for the AV:

Parameter	Analytical Limit	TS Allowable Value		Nominal Trip Setpoint
		Current	Proposed	Current and Proposed
Low Tavg interlock	540°F	≥ 540°F	≥ 542°F	543°F

The proposed AV change establishes the upper limit on the as-found NTSP to remain within the assumptions of the safety analyses that credit the low Tavg interlock coincident with safety injection and high steam flow for steam line isolation.

The NTSP has design margin with respect to the AV.

Steam Generator Water Level High - Feedwater Isolation

Feedwater isolation on high steam generator level is not specifically credited in the safety analyses. Therefore, the trip function lacks an analytical limit upon which to base the AV. The proposed AV is based on the process limit of the maximum reliable indication of the level instrumentation (97% level). The proposed AV is established by calculation including all instrument uncertainties. The following change is proposed for the AV:

Parameter	Process Limit	TS Allowable Value		Nominal Trip Setpoint
		Current	Proposed	Current and Proposed
SG Water Level High – Feedwater Isolation	97%	"NA"	≤ 90%	78%

The NTSP has design margin with respect to the AV.

Auxiliary Feedwater - Undervoltage Bus A01 & A02

Auxiliary Feedwater actuation on Undervoltage Bus A01 & A02 is not credited in any accident analysis and therefore lacks an Analytical Limit. The function provides a backup signal for AFW actuation in the Loss of AC Power event. The NTSP for the function is being added to TS Table 3.3.2-1 to conform to the "multiple column" format allowed by NUREG-1431:

Parameter	TS Allowable Value	Nominal Trip Setpoint
	Current and Proposed	Current and Proposed
Undervoltage Bus A01 & A02	≥ 3120 V	3255 V

SI Block Pressurizer Pressure

The SI block pressurizer pressure function is an operating bypass that is not specifically credited in the safety analyses. The safety function of the operating bypass is to automatically remove (unblock) the bypass to reinstate SI actuation on low pressurizer pressure and low steam line pressure when the RCS repressurizes on a normal plant startup. The proposed AV is based on the upper as-found limit established by calculation for the SI unblock NTSP. The following change is proposed for the AV:

Parameter	TS Allowable Value		Nominal Trip Setpoint	
	Current	Proposed	Current	Proposed
SI Block - Pressurizer Pressure	≤ 1800 psig	≤ 2005 psig	1775 psig	2000 psig

The current NTSP and AV are based on a normal RCS operating pressure of 1985 psig (nominal), which existed prior to 2000 to address primary-to-secondary differential pressure issues. In 2000/2001, the RCS operating pressure on both units was increased to the original plant operating pressure of 2235 psig (nominal) as part of the fuel upgrade program. This setpoint change will restore the SI block NTSP and AV to values consistent with RCS operation at 2235 psig.

The proposed setpoint provides reasonable margin to both normal operating pressure and the pressurizer pressure low SI setpoint (1735 psig decreasing). The AV provides an as-found limit for the proposed NTSP that supports the required function of an operating bypass.

2.4.1.2.3.3 Control Systems

The various reactor control systems are described in FSAR Section 7.7, Control Systems. The reactor control systems are designed to limit nuclear plant transients for prescribed design load perturbations, under automatic control, within prescribed limits to preclude the possibility of a reactor trip in the course of these transients. During steady-state operation, the primary function of the reactor control is to maintain a programmed average reactor coolant temperature that rises in proportion to load. The control systems also limit nuclear plant system transients to prescribed limits about this programmed temperature for specified load perturbations. Complete supervision of both the nuclear and turbine generator plants is accomplished from the central control room. This supervision includes the capability to test periodically the operability of the RPS.

The current design basis operational transients described in FSAR Section 7.7.1, Rod Control System, are:

- Step-load change of ±10% or ramp load change of 5% per minute within the load range of 15% to 100% of rated power
- 50% load loss from any power level with steam dump
- Net loss of electrical load or turbine trip below 50% power with steam dump

The design basis 50% load loss for EPU has been changed to the equivalent of 50% of the EPU rated thermal power (RTP) at a maximum turbine unloading rate of 200% per minute. The 50% load loss at a maximum rate of 200% per minute is more realistic than a step change and is consistent with uprating projects previously performed on other Westinghouse plants.

The analyses evaluating the response to design basis operational transients at EPU conditions are described in LR Section 2.4.2, Plant Operability. The acceptable response to the design basis operation transients and accidents and transients associated with control system failures are based on the changes described for the rod control system and steam dump system being implemented.

<u>Turbine First Stage Pressure Instrumentation</u>

When the turbine generator is on line, turbine first stage pressure increases essentially linearly from 0% - 100% turbine load and provides a close correlation of secondary power to reactor power. This allows turbine first stage pressure to be used as a reliable input demand signal or permissive to the various reactor control systems between 0% and 100% reactor power. The pre-EPU 0% -100% turbine load turbine first stage pressure correlates to 0 – 544 psig. For EPU, a new HP turbine rotor is being installed which currently is expected to generate a 0% - 100% power nominal first stage turbine pressure of 0 – 664 psig. Actual full power turbine first stage pressure may change slightly as the HP turbine design is refined and instrument calibrations will be revised accordingly.

The existing turbine first stage pressure transmitters and associated indications will be recalibrated and scaled to a range of 0 – 664 psig. The span of the turbine first stage pressure transmitters is 0 - 800 psig. The inputs to each of the following systems will be recalibrated to respond at the appropriate value for the new 0 – 100% power nominal turbine first stage pressure of 0 – 664 psig.

- AMSAC – arm/disarm circuit permissive P-20 at first stage pressure equivalent to 40% turbine power
- P-2 Permissive – blocks Automatic Rod Withdrawal block at less than 15% turbine load
- P-7 Permissive- in conjunction with P-10, bypasses low pressurizer pressure and low RCS flow, undervoltage, and under frequency trips
- T_{ref} input to the Reactor Coolant T_{avg} Control program
- EHC Turbine Control

Rod Control System Changes

The rod control system responds to changes in RCS temperature and secondary load as sensed by the RCS measured T_{avg} instrumentation and turbine first stage pressure instrumentation. The rod control system is designed to maintain average RCS temperature within ±1.5°F of the 0% - 100% T_{avg} program reference value (T_{ref}) derived from 0-100% power turbine first stage pressure (0 – 664 psig). In addition, the rod control system responds to deviations between reactor power and turbine load as sensed by the mismatch between power range instruments and turbine first stage pressure instrumentation. Both the T_{avg} program and the power mismatch program control rod speed and direction during normal and transient operation.

The EPU 0 – 100% power T_{avg} temperature program (T_{ref}) is changing from the current 547°F to 570°F to 547°F to 576°F based on a 0 – 664 psig turbine first stage pressure. Once the T_{ref} program is calibrated with the turbine first stage pressure range and temperature control band, the rods are expected to respond as designed to T_{avg} temperature deviations from T_{ref}.

The power mismatch circuits will be calibrated with the new 0 – 100% turbine first stage pressure values which will ensure the power mismatch circuits will continue to provide maximum rod speed with a deviation between nuclear power and turbine power of 10%.

Pressurizer Level Program

The pressurizer level control system maintains the pressurizer level within a programmed band consistent with auctioneered high T_{avg}. The programmed level is designed to maintain a sufficient margin above the low level alarm where the heaters turn off and letdown isolation occurs while maintaining the level low enough that a sufficient steam volume is maintained to ensure the pressurizer does not go solid during accidents and transient conditions.

Analyses described in LR Section 2.4.3, Pressurizer Component Sizing, and LR Section 2.8.5, Accident and Transient Analyses, determined the nominal pressurizer level program for EPU must be changed from the current 20% - 45.8% program to a new nominal program of 20% at no load T_{avg} to 47% at the maximum analyzed T_{avg} of 577°F.

Low limit = 20% span at no-load T_{avg} of 547°F

High limit = 47% span at maximum analyzed T_{avg} of 577°F

The level control program is linear between no-load and maximum analyzed T_{avg}. The level program is clamped at the high limit if measured T_{avg} exceeds 577°F.

Steam Dump Control and Turbine Bypass Systems

The steam dump control and turbine bypass systems are comprised of the steam generator atmospheric dump valves (ADVs) and the condenser steam dump valves. The ADVs can be used to remove sensible heat stored in the RCS at shutdown and cooldown when the condenser steam dumps are not available. The condenser steam dump system removes sensible heat stored in the RCS for a large rapid load decrease or a reactor trip on a turbine trip. With condenser steam dump not available, a rapid turbine load reduction would result in a large steam pressure increase and could potentially challenge the Main Steam Safety Valves (MSSVs). Steam is dumped in order to remove the stored heat in the primary system at a rate fast enough to prevent lifting the MSSVs for a large rapid load decrease, or a reactor trip. The evaluation of the steam bypass system is described in LR Section 2.5.5.3, Turbine Bypass, and LR Section 2.4.2, Plant Operability.

With the condenser available, the condenser steam dumps (groups A – D) are armed based on a rapid decrease in turbine first stage pressure (equivalent to >10% load decrease) and the dump valves either modulate open or are tripped open based on the magnitude of error (ΔT) between the measured T_{avg} and the reference temperature (T_{ref}) programmed off turbine first stage pressure.

As described in LR Section 2.4.2, Plant Operability, the current steam dump valve capacity at EPU conditions (coincident with rod motion) is sufficient to accommodate a rapid load decrease equivalent to 50% reactor thermal power (RTP) at a rate of 200% per minute and a turbine trip without a reactor trip below the nominal permissive P-9 setpoint of 50% power, provided the full power Tavg is between 572°F to 577°F. The nominal permissive P-9 setpoint is 35% below the full power Tavg of 572°F. Tavg load rejection and Tavg turbine trip steam dump current setpoints are acceptable at EPU conditions for full power Tavg anywhere between 558°F and 577°F.

Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC)

The PBNP Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) as required by 10 CFR 50.62 is described in FSAR Section 7.4.1, ATWS Mitigation System Actuation Circuitry. The changes to this circuitry are associated with permissive P-20 which arms and disarms the circuit at a turbine first stage pressure equivalent to approximately 40% nuclear power, and recalibrating the turbine first stage pressure, steam flow, and feedwater flow inputs for the EPU full load values. The P-20 permissive will be recalibrated to arm/disarm at approximately 40% for the appropriate turbine first stage pressure consistent with the new 0% - 100% power nominal turbine first stage pressure range of 0 – 664 psig.

Additional Changes

As described in LR Section 2.8.5, Accident and Transient Analyses, and the subsequent non-LOCA sections, changes were required to setpoints and associated functions to support the EPU. Some of these changes resulted in changes to the TS. They include reducing the MSSV actuation settings for the two highest setpoints and reduction in the high pressurizer pressure reactor trip setpoint Analytical Limit from 2425 to 2418 psia. In addition, the high neutron flux reactor trip Analytical Limit was reduced from 118% to 116% RTP. The OTΔT and OPΔT reactor trip setpoints were recalculated to adequately protect core thermal limits at the uprated power level conditions. The lead/lag compensation on low steam line pressure safety injection signal was changed from the current values of 12 seconds / 2 seconds to 18 seconds / 2 seconds; the safety analysis Analytical Limit for this function was also increased from 335 psia to 410 psia. The safety analysis low-low steam generator water level reactor trip setpoint was increased from its current value of 17% to 20% NRS. These changes are identified in Table 2.8.5.0-9 of LR Section 2.8.5, Accident and Transient Analyses, and supported by non-LOCA safety analyses.

Evaluation of Impact on Renewed Plant Operating License Evaluations and License Renewal Programs

Safety related instrumentation or instrumentation that performs a function necessary to address one of the five regulated events are scoped within license renewal, however instruments typically are scoped as active components and are excluded from aging management review. Cables, connectors, pipes and tubes that service the in-scope instruments are passive and require aging management review and are addressed in other sections of this LR. No new protection or control systems were added to the scope of License Renewal as a result of the EPU. The changes to instrumentation as a result of the EPU are predominately rescaling and recalibration of existing instrumentation and introduce no added components or configuration of the instruments. The rescaling and recalibration of these instruments do not impact the design function of the instruments and do not effect the conclusions stated in the licensing renewal evaluations. Therefore, the conclusions reached in NUREG 1839, Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant Units 1 and 2, remain valid for the EPU.

For the limited number of cases discussed above, instruments or active instrument components must be changed to ensure the operability of the instruments for EPU conditions. These instrument changes are being performed in accordance with the plant modification process which evaluates the impact of the change with regard to license renewal and aging management.

2.4.1.3 Results

The changes to the instrumentation and controls for EPU are the result of accident and transient analyses and system evaluations to verify the systems and controls will continue to provide the required indication, protection actions, and plant response as originally designed. The changes ensure the DNB values remain within acceptable limits and the RCS pressure boundary, main steam pressure boundary, and containment boundary are all maintained within the design values. There are no new protection or control systems required to support EPU. The identified instrumentation recalibration and instrument rescaling will ensure the instrumentation continues to allow monitoring of plant process parameters during normal, transient and accident conditions and provide protective functions as required.

2.4.1.4 Conclusions

PBNP has reviewed the I&C systems relevant to the effects of the proposed EPU on the functional design of the RPS, ESFAS, and control systems. PBNP concludes that the evaluation has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU (including the proposed changes to the Technical Specification listed in the preceding section, Setpoint Changes and Instrument Changes) are consistent with the plant's design basis, including the revised load rejection design basis to a rapid ramp load reduction equivalent to 50% rated thermal power at a maximum unloading rate of 200% per minute. PBNP further concludes that the systems will continue to meet the PBNP current licensing basis with respect to the requirements of PBNP GDC 1, 11, 12, 13, 14, 15, 19, 20, 23, 25, and 26. Therefore, PBNP finds the proposed EPU acceptable with respect to instrumentation and controls.

Table 2.4.1-1
Summary of EPU-related RPS Functions, Allowable Values and Analytical Limits

Reactor Trip Function	TS Table 3.3.1-1 Function No.	Setpoint Category	SL-related LSSS?	Proposed AV	Current Analytical Limit	EPU Analytical Limit	Related Safety Analyses
Power Range Flux - High	2.a	A	Yes	≤ 109% RTP	118% RTP	116% RTP	Rod Withdrawal at Power
Power Range Flux - Low	2.b	A	Yes	≤ 28% RTP	35% RTP	35% RTP	Rod Withdrawal from Subcritical RCCA Ejection
Intermediate Range Flux	3	B	No	≤ 43% RTP	-	-	-
Overtemperature ΔT	5	A	Yes	Unchanged	$1.255 * \Delta T_o$	$1.295 * \Delta T_o$	Rod Withdrawal at Power
Overpower ΔT	6	A	Yes	Unchanged	$1.14 * \Delta T_o$	$1.165 * \Delta T_o$	Steam System Piping Failure – HZP Steam System Piping Failure – Full Power
Pressurizer Pressure - Low	7.a	A	Yes	≥ 1860 psig	1815 psig	1840 psig	OPTOAX
Pressurizer Pressure - High	7.b	A	Yes	≤ 2385 psig	2410 psig	2403 psig	Loss of External Electrical Load/Turbine Trip
Pressurizer Water Level – High	8	B	No	≤ 85%	-	-	-
Steam Generator Water Level – Low Low	13	A	Yes	≥ 29.3%	17%	20%	Loss of Normal Feedwater Loss of All AC to Station Auxiliaries
Steam Generator Water Level – Low	14	B	No	≥ 10%	-	-	-
Permissive P-6	17.a	C	No	≥ 4E-11 amp	-	-	-
Permissive P-7 Neutron Flux	17.b (1)	C	No	≤ 13% RTP	-	-	-
Permissive P-7 Turbine Impulse	17.b (2)	C	No	≤ 12.8% turbine power	-	-	-
Permissive P-8	17.c	C	No	≤ 38% RTP	-	-	-
Permissive P-9	17.d	C	No	(a)	-	-	-
Permissive P-10	17.e	C	No	≥ 6% RTP and ≤ 12% RTP	-	-	-

(a) ≤ 38% RTP for full design power Tavg < 572°F or ≤ 53% RTP for full design power Tavg ≥ 572°F and for end-of-cycle coastdown

Table 2.4.1-2
Summary of EPU-related ESFAS Functions, Allowable Values and Analytical Limits

ESFAS Function	TS Table 3.3.2-1 Function No.	Setpoint Category	SL-related LSSS?	Proposed AV	Current Analytical Limit	EPU Analytical Limit	Associated Safety Analyses
SI Containment Pressure - High	1.c	A	Yes	≤ 5.3 psig	6 psig	6 psig	Steam System Piping Failure – HZP Steam System Piping Failure – Full Power
SI Pressurizer Pressure - Low	1.d	A	No	≥ 1725 psig	1700 psig (Process Limit)	1700 psig (Process Limit)	(Process limit is transmitter lower range limit – LOCA, SLB, and SGTR analyses use limits below the lower range limit)
SI Steam Line Pressure - Low	1.e	A	Yes	≥ 520 psig	320 psig	395 psig	Steam System Piping Failure – HZP Steam System Piping Failure – Full Power
CS Containment Pressure – High High	2.c	A	No	≤ 28 psig	30 psig	30 psig	Steam Line Break M&E Release (Inside Containment)
SLI Containment Pressure – High High	4.c	B	No	≤ 18 psig	-	-	-
SLI High Steam Flow	4.d	A	Yes	≤ Δp corresponding to 0.8 E6 lb/hr @ 1005 psig	0.97 E6 lb_m/hr	1.07 E6 lb_m/hr	Steam Line Break M&E Release (Outside Containment)
Low Tavg	4.d	A	No	≥ 542°F	540°F	540°F	Steam Line Break M&E Release (Outside Containment)
SLI High-High Steam Flow	4.e	A	Yes	≤ Δp corresponding to 4.9 E6 lb/hr @ 586 psig	4.0 E6 lb_m/hr (Process Limit)	5.0 E6 lb_m/hr (Process Limit)	Steam System Piping Failure - HZP
FW Isolation SG High-High Level	5.b	B	No	≤ 90%	-	-	-
AFW Steam Generator Water Level – Low Low	6.b	A	Yes	≥ 29.3%	17%	20%	Loss of Normal Feedwater Loss of All AC to Station Auxiliaries
AFW Pump Suction Transfer on Suction Pressure Low	6.e	A	Yes	≥ 5.7 psig	none	5.241 psig	Loss of Normal Feedwater Loss of All AC to Station Auxiliaries
Condensate Isolation Containment Pressure	7.a	Function Deleted					
SI Block – Pressurizer Pressure	8	C	No	≤ 2005 psig	-	-	-

ENCLOSURE 6

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261 SUPPLEMENT 3
EXTENDED POWER UPRATE

APPENDIX E

15 pages follow

Appendix E

RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1

Purpose

This appendix describes the application of Limiting Safety System Settings (LSSS) in the Technical Specifications (TS) for the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS), the method used to develop the LSSS values for functions, the methods used to maintain the functions operable, and the control of the methods in design basis documents.

Limiting Safety System Settings

10 CFR 50.36 requires that Limiting Safety System Settings be included in the TS. LSSS values for RPS and ESFAS functions that actuate at an adjustable setpoint are established in TS Tables 3.3.1-1, Reactor Protection System Instrumentation, and 3.3.2-1, Engineered Safety Feature Actuation System Instrumentation. NextEra Energy Point Beach, LLC is adopting a "multiple column" format for RPS and ESFAS setpoint TS tables consistent with NUREG-1431, Standard Technical Specifications Westinghouse Plants, and setpoint terminology consistent with NUREG-1431 and joint NRC/industry efforts to clarify the application of setpoint methodology for LSSS functions. The multiple column format includes separate columns for the Allowable Value (AV) and Nominal Trip Setpoint (NTSP) for adjustable setpoint functions.

The NTSP is the nominal value chosen for the field setpoint to protect the safety analyses. The NTSP accounts for uncertainties in setting the channel (e.g., during calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of actuation of the channel over time (e.g., drift during surveillance intervals) and other factors which may affect its actual performance (e.g., a harsh sensor environment during accidents). In this manner, the NTSP ensures that Safety Limits are not exceeded. Therefore, the NTSP meets the definition of an LSSS. Field setpoints installed in the bistables may be more conservative than the NTSP as necessary in response to plant conditions.

The AV presented in TS Tables 3.3.1-1 and 3.3.2-1 for a function is the least conservative as-found value of a trip setpoint that a channel can have when tested and be considered OPERABLE in the context of the TS. If the as-found value is beyond the Allowable Value, the channel may not be able to perform its protective function within the limits of the safety analyses. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval.

Calibration and surveillance requirements defined in the TS are performed in accordance with Point Beach Nuclear Plant (PBNP) surveillance and maintenance procedures. During the Channel Operational Test (COT), instrument loops are surveilled by placing them in a bypass or trip condition and injecting a test signal in place of the sensor input signal. Calibration and surveillance limits are established to ensure that instrument channels are performing as expected and within the statistical allowances of the uncertainty terms assigned in applicable design basis uncertainty calculations. The as-left tolerance is an allowance for setting the channel since the bistable cannot be set precisely at the NTSP. The as-left setting of a channel that has been adjusted will be within the as-left tolerance for that channel. The as-found tolerance is established to detect when a channel under surveillance is performing outside of the assumptions of design basis uncertainty calculations considering instrument drift expected over the surveillance period. The as-left and as-found tolerance are two sided bands centered around the NTSP. Where field settings are set conservative with respect to the NTSP, the tolerances are applied to the field setting. Both the as-left and as-found criteria are implemented on applicable RPS and ESFAS functions through the use of two new notes in TS Tables 3.3.1-1 and 3.3.2-1 adopted from recent joint NRC/industry efforts to clarify channel operability based on as-found settings relative to the associated as-left and as-found calibration tolerances.

NTSPs, AVs, as-left tolerances and as-found tolerances are determined for all RPS functions in TS Table 3.3.1-1 and all ESFAS functions in TS Table 3.3.2-1 unless one or more of the following exclusions apply:

1. Manual actuation circuits, automatic actuation logic circuits, or instrument functions that derive input from contacts which have no associated sensor or adjustable device (e.g., limit switches, breaker position switches, float switches, proximity detectors) are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.

2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e. trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants inservice testing program.

3. Functions and Surveillance Requirements which test only digital components are excluded. There is no expected change in result between consecutive surveillances of these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.

Permissives and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but are not explicitly modeled in the Safety Analyses and therefore, lack an Analytical Limit. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative starting assumptions for the accident analysis, the permissives or interlocks are generally considered nominal values without regard to measurement accuracy. For the applicable permissives and interlocks, AVs and NTSPs are provided in TS Table 3.3.1-1 and TS Table 3.3.2-1 based on nominal values assumed in the safety analyses and expected rack performance over the calibration interval.

Design Basis Uncertainty Calculations

The bases for NTSPs, AVs, as-left tolerances, and as-found tolerances are contained in design basis uncertainty calculations. These calculations are based on industry standard ISA 67.04 as implemented at PBNP in design guideline DG-I01, Instrument Setpoint Methodology. The general methodology for determining these values in calculations will be described in

FSAR Section 7.2. The calculated values are determined such that there is 95% probability and 95% confidence level that the instrument channel will trip prior to the process variable exceeding an established limit.

Analytical Limits (AL) are established by a Safety Analysis for FSAR design basis events. The AL is the process value at which a protective action initiation is assumed to occur to ensure that a safety limit is not exceeded. If there is no AL for a specific function, a Process Limit (PL) such as the limit of an instrument span may be used in place of an AL. If the AL is outside the instrument span, the span limit is used as the PL to assure the function will occur within the instrument span. In some cases, an AL or PL is not provided for an RPS/ESFAS function. For example, RPS and ESFAS interlocks and operating bypasses (permissives) that allow bypassing of and automatically reinstate protective functions are not specifically credited in the safety analyses and have no AL or PL associated with them.

AVs for trip and actuation functions (excluding interlocks and permissives) are conservatively determined accounting for all known instrument loop errors including process measurement effects, calibration uncertainties, reference accuracies, environmental effects, and instrument drift. Random and independent uncertainty terms are combined using the square root sum of squares (SRSS) method and bias terms are included algebraically. For setpoints approached from one direction, a single-sided conversion factor is used to convert the random component of total loop uncertainty (TLU) from a two-sided error to a single-sided error prior to combining the random and bias uncertainties. For a trip or initiating function where the process variable increases towards an AL or PL, TLU is subtracted from the AL or PL. For variables that decrease towards the limit, TLU is added to the AL or PL. PBNP design basis calculations label the result as the limiting trip setpoint (LTSP). The LTSP result may be rounded in a conservative direction to arrive at the Allowable Values contained in TS Table 3.3.1-1 and TS Table 3.3.2-1. For interlocks and permissives, AVs are determined by adding a single-sided as-found tolerance, in the conservative direction, to the permissive or interlock nominal setting.

NTSPs are conservative with respect to the AV. For trip and actuation functions (excluding interlocks and permissives), NTSPs are chosen to provide margin to the Limiting Trip Setpoint (which is converted to the AV by rounding). For interlocks and permissives, the NTSP is the nominal value assumed in the analysis. In all cases, the difference between the NTSP and the AV is equal to or exceeds the channel as-found acceptance criteria.

The as-left tolerance for setting devices in instrument loops is based on rack component accuracies and operating experience (referred to in design basis calculations as rack as-left (RAL) tolerances).

Rack as-found tolerances (referred to in design basis calculations as RAF tolerances) for surveillance and calibration purposes are based on rack component accuracies combined with drift expected to occur over the span of the calibration or surveillance frequency. The calculations determine RAF for each module within a protection channel rack. The RAF applied during the COT is for those rack modules that are tested by the COT.

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RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1

Design Basis Documentation

The basis for the NTSPs, AVs, as-left tolerances, and as-found tolerances, including definition and assignment of each uncertainty term, is maintained in setpoint calculations controlled under PBNP procedures. The calculations that are basis for setpoint changes described in LR 2.4.1 are referenced and summarized (where practical) below.

A description of the basic methods used to determine TLU, LTSP, AV, as-left tolerance, and as-found tolerance will be added to Chapter 7 of the PBNP FSAR. The FSAR will describe, in general, the basis for the surveillance and calibration requirements and the basic calculation methods to combine uncertainties and apply the result to establishing the various setpoint terms. This information will become design and licensing bases and will provide a framework for 10 CFR 50.59 evaluations for plant modifications. However, the 10 CFR 50.59 evaluation process and the design and licensing bases specified in the PBNP FSAR should not prohibit making changes to the following:

- The field setpoint, provided it is equal to or conservative with respect to the NTSP.
- The TLU, provided the basic calculation methods are consistent, and the TLU does not exceed the difference between the AV and the applicable AL or PL.
- The magnitude of the as-left and as-found tolerances, provided they are consistent with the description of the basic methods included in the FSAR.

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Calculation Summaries

Summaries of setpoint calculations that provide a basis for the values presented in LR Section 2.4.1 are provided below. LR Section 2.4.1 describes the applicable changes to ALs or PLs and the basis for those changes.

Only portions of the calculations applicable to the RPS or ESFAS functions are summarized and only non-zero terms are shown. Summaries are not provided for the reactor trip functions for OTΔT, OPΔT and Steam Flow/Feedwater Flow mismatch because of the complexity of the calculations. The calculations are available for review upon request. This summary uses the following terms and acronyms which may vary from the summarized calculations:

AL	Analytical Limit
CE	Calorimetric Flow Error
FE	Flow Element Error
FTSP	Field Trip Setpoint
IR	Insulation Resistance Effect
La	Lead/Lag Accuracy
Ld	Lead/Lag Drift Allowance
Lmte	Lead/Lag Measurement & Drift Allowance
Lv	Lead/Lag Setting Tolerance
LSSS	Limiting Safety System Setpoint
LTSP	Limiting Trip Setpoint
OL^+	Upper Operability Limit
PA	Ambient Pressure Effect
PE	Process Error
PL	Process Limit
RAL	Rack As-Left Tolerance
RAF	Rack As-Found Tolerance
Rd	Rack Drift Allowance
Rmte	Rack Measurement & Test Equipment Accuracy Uncertainty
RTP	Reactor Thermal Power
Rv	Rack Setting Tolerance
Sd	Sensor Drift Allowance
St	Sensor Temperature Uncertainty
Sp	Sensor Pressure Effect
Sv	Sensor Setting Tolerance
SRSS	Square Root Sum of Squares of terms that follow in ()
TLU	Total Loop Uncertainty

The setpoint calculations use the following basic formulas:

LTSP = AL or PL + TLU	(for decreasing process)
= AL or PL - TLU	(for increasing process)
TLU = SRSS (random terms) + bias terms	(for two-sided uncertainties)
= 0.839 * SRSS (random terms) + bias terms	(for single-sided uncertainties)

RAL = +/- Rack Setting Tolerance (Rv)

RAF = +/- SRSS (Rv, Rd, Rmte)

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Parameter: Steam Generator Narrow Range Level
Summary of Calculation Note Number CN-CPS-07-6

Low-Low SGNR Water Level – RPS Reactor Trip, ESFAS Aux Feedwater

0-100% instrument span = 0-100% SGNR Water Level span

AL = 20% span
TLU = 0.839 * SRSS(Sd, Sv, St, Sspe, Rd, Rv) + PE
= 9.23% span
Sd = +/- 0.877% span
Sv = +/- 0.5% span
St = +/- 0.751% span
Sspe = +/- 0.637% span
Rd = +/- 0.212% span
Rv = +/- 0.5% span
PE = + 7.96%
LTSP = 20 + 9.23 = 29.23% span (Allowable Value rounded to ≥ 29.3%)
FTSP = 30% span (NTSP)
RAL = +/- 0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/- 0.543% of span (as-found tolerance)

Low SGNR Water Level RPS Trip

0-100% instrument span = 0-100% SGNR Water Level span

AL = None, PL =0% (Instrument Span Limit)
TLU = 0.839 * SRSS(Sd, Sv, SSpe, Rd, Rv) + PE
= 9.95% span
Sd = +/- 0.877% span
Sv = +/- 0.5% span
St = +/- 0.751% span
Sspe = +/- 0.637% span
Rd = +/- 0.222% span
Rv = +/- 0.5% span
PE = + 8.68% span
LTSP = 9.95% span (Allowable Value rounded up to ≥ 10%)
FTSP = 30% span (NTSP)
RAL = +/-0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/- 0.547% of span (as-found tolerance)

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RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1

Parameter: Steam Generator Narrow Range Level
Summary of Calculation Note Number CN-CPS-07-6

High-High SGNR Water Level ESFAS Feedwater Isolation

0-100% instrument span = 0-100% SGNR Water Level span

AL = None, PL = 97% span (Maximum Reliable Indication Limit)
TLU = 0.839 * SRSS(Sd, Sv, St, Sspe, Rd, Rv) + PE
= -6.04% span
Sd = +/- 0.877% span
Sv = +/- 0.5% span
St = +/- 0.751% span
Sspe = +/- 0.637% span
Rd = +/- 0.212% span
Rv = +/- 0.5% span
PE = - 4.77% span
LTSP = 90.96% span (Allowable Value rounded down to ≤ 90%)
FTSP = 78% span (NTSP)
RAL = +/- 0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/-0.543% span (as-found tolerance)

Parameter: Pressurizer Pressure
Summary of Calculation No: 2009-0001

Low Pressurizer Pressure Reactor Trip

0-100% instrument span = 1700-2500 psig

AL = 1840 psig
TLU = 0.839 * SRSS(Sd, Sv, Ss, St, Rd, Rv, La, Ld, Lmte, Lv, PA) + PE
= 2.334% span = 15.7 psig
Sd = +/- 0.844 % span
Sv = +/- 0.5 % span
Ss = +/- 1.0% span
St = +/- 1.538 % span
Rd = +/- 0.222% span
Rv = +/- 0.5% span
La = +/- 0.5% span
Ld = +/- 0.5% span
Lmte = +/- 0.083% span
Lv = +/- 0.5% span
PA = +/- 0.25 % span
PE = - 0.092% span
LTSP = 1840 + 15.7 = 1855.7 psig (Allowable Value rounded up to ≥ 1860 psig)
FTSP = 1925 psig (NTSP)
RAL = +/- 0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/-0.547% span (for bistable)
RAF = SRSS(Lv, Ld, Lmte) = +/-0.712% span (for L/L module) (as-found tolerance)

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RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1

<u>High Pressurizer Pressure Reactor Trip</u>

0-100% instrument span = 1700-2500 psig

AL = 2403 psig
TLU = 0.839 * SRSS(Sd, Sv, Ss, St, Rd, Rv, PA) + PE
= -1.909% span = -15.3 psig
Sd = +/-0.844 % span
Sv = +/-0.5 % span
Ss = +/-1.0% span
St = +/-1.538 % span
Rd = +/-0.222% span
Rv = +/-0.5% span
PA = +/-0.25 % span
PE = -0.092% span
LTSP = 2403 – 15.3 = 2387.7 psig (Allowable Value rounded down to ≤ 2385 psig)
FTSP = 2365 psig (NTSP)
RAL = +/- 0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/- 0.547% span (as-found tolerance)

<u>Parameter: Pressurizer Pressure</u>
Summary of Calculation No: 2009-0001

<u>Low Pressurizer Pressure SI Actuation</u>

0-100% instrument span = 1700-2500 psig

AL = 1648 psig, however PL = 1700 psig due to range limitation
TLU = 0.839 * SRSS(Sd, Sv, Ss, St, Rd, Rv, PA) + PE
= 3.406% span = 22.9 psig
Sd = +/- 0.844 % span
Sv = +/- 0.5 % span
Ss = +/- 1.0% span
St = +/- 1.538 % span
Rd = +/- 0.222% span
Rv = +/- 0.5% span
PE = - 0.092% span
PA = +/- 0.25 % span
LTSP = 1700 + 22.9 = 1722.9 psig (Allowable Value rounded up to ≥ 1725 psig)
FTSP = 1735 psig (NTSP)
RAL = +/- 0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/- 0.547% span (as-found tolerance)

<u>Pressurizer Pressure SI Unblock</u>

0-100% instrument span = 1700-2500 psig

AL = None
TLU = N/A
LSSS = ≤ 2005 psig (Allowable Value)
FTSP = 2000 psig (NTSP)
RAL = +/- 0.5% span
RAF = SRSS(Rv, Rd) = +/- 0.547% span
Rd = +/- 0.222% span
Rv = +/- 0.5% span

Parameter: Power Range Nuclear Instrumentation
Summary of Calculation No: 2009-0002

<u>Power Range Neutron Flux – High Reactor Trip</u>

0 – 100% instrument span = 0 – 120% RTP

AL = 116% RTP
TLU = 0.839 * SRSS(Ra, Rd, Rmte, Rv, PE)
= -6.342% span = -6.39 RTP
Ra = +/- 2.0% span
Rd = +/- 2.0% span
Rmte = +/- 0.596% span
Rv = +/- 0.833% span
PE = +/- 5.583% span
LTSP = 116 – 6.39 = 109.61 RTP (Allowable Value rounded down to ≤ 109% RTP)
FTSP = 107% RTP (NTSP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.247% span = +/- 2.696% RTP (as-found tolerance)
RAL = +/- 0.833% span = 1.0% RTP (as-left tolerance)

<u>Power Range Neutron Flux – Low Reactor Trip</u>

0 – 100% instrument span = 0 – 120% RTP

AL = 35% RTP
TLU = 0.839 * SRSS(Ra, Rd, Rmte, Rv, PE)
= -6.342% span = - 6.39 RTP
Ra = +/- 2.0% span
Rd = +/- 2.0% span
Rmte = +/- 0.596% span
Rv = +/- 0.833% span
PE = +/- 5.583% span
LTSP = 35 – 6.39 = 28.61 RTP (Allowable Value rounded down to ≤ 28% RTP)
FTSP = 20% RTP (NTSP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.247% span = +/- 2.696% RTP (as-found tolerance)
RAL = +/- 0.833% span = 1.0% RTP (as-left tolerance)

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RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1

<u>Parameter: Power Range Nuclear Instrumentation</u>

Summary of Calculation No: 2009-0002

<u>Power Range Neutron Flux – Interlock P-9</u>

0 – 100% instrument span = 0 – 120% RTP

AL = None
TLU = N/A
LSSS = ≤ 38% RTP for Tavg < 572°F (Allowable Value)
= ≤ 53% RTP for Tavg ≥ 572°F and end-of-cycle coastdown
FTSP = 35% RTP for Tavg < 572°F (NTSP)
= 50% RTP for Tavg ≥ 572°F and end-of-cycle coastdown
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.247% span = +/- 2.696% RTP
RAL = +/- 0.833% span = 1.0% RTP

<u>Power Range Neutron Flux – Interlock P-8</u>

0 – 100% instrument span = 0 – 120% RTP

PL = 45% RTP
TLU = N/A
FTSP = 35% RTP (NTSP)
OL^+ = FTSP + SRSS(Rv, $Rd_{3\sigma}$)
Rv = +/- 0.833% span
$Rd_{3\sigma}$ = +/- 3.0 % span
OL^+ = 35% + 3.73% RTP (Allowable Value rounded to ≤ 38% RTP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.247% span = +/- 2.696% RTP
RAL = +/- 0.833% span = 1.0% RTP

<u>Power Range Neutron Flux – Interlock P-7</u>

0 – 100% instrument span = 0 – 120% RTP

AL = None
TLU = N/A
LSSS = 13% RTP (Allowable Value)
FTSP = 10% RTP (NTSP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.247% span = +/- 2.696% RTP
RAL = +/- 0.833% span = 1.0% RTP

<u>Power Range Neutron Flux – Interlock P-10</u>

0 – 100% instrument span = 0 – 120% RTP

AL = None
TLU = N/A
LSSS = ≥ 6% RTP and ≤ 12% RTP (Allowable Value)
FTSP = 9% RTP (NTSP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.247% span = +/- 2.696% RTP
RAL = +/- 0.833% span = 1.0% RTP

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RPS/ESFAS Setpoint Methodology – Supplement to LR 2.4.1

Parameter: Intermediate Range Instruments
Summary of Calculation No: 2009 -0003

IRM High Flux Reactor Trip

AL = None
TLU = N/A
FTSP = 25% RTP
OL^{+} = FTSP + SRSS(Rv, $Rd_{3\sigma}$)
Rv = +/- 0.10% span
$Rd_{3\sigma}$ = +/- 3.0 % span
OL^{+} = 43.47% RTP (Allowable Value rounded to $\leq$ 43% RTP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 2.010% span = +/- 0.201 Vdc (as-found tolerance)
Rv = +/- 0.10% span
Rd = +/- 2.0 % span
Rmte = +/- 0.177% span
RAL = +/- 0.10% span (as-left tolerance)

IRM Permissive P-6 Unblock

AL = None
TLU = N/A
LSSS = $\geq 4 \times 10^{-11}$ amps (Allowable Value)
FTSP = 1.0×10^{-10} amps (NTSP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 5.087% span = +/- 0.6×10^{-10} amps
Rv = +/- 3.763% span
Rd = +/- 2.0% span
Rmte = +/- 2.778% span
RAL = +/- 3.763% span = +/- 0.500×10^{-10} amps

Parameter: Turbine Impulse Pressure
Summary of Calculation No: 2007 -0001

Turbine Impulse Pressure P-7 Unblock

AL = None
TLU = N/A
LSSS = $\leq$ 12.8% Turbine Power (Allowable Value)
FTSP = 10% Turbine Power = 41.1 psig (NTSP)
RAF = SRSS(Rv, Rd, Rmte) = +/- 1.667% span = +/- 10.8 psi
Rv = +/- 0.5 % span
Rd = +/- 1.59% span
Rmte = +/- 0% span
RAL = +/- 0.5% span = +/- 3.3 psi

Parameter: Pressurizer Water Level Instrumentation
Summary of Calculation Note No: CN-CPS-07-2

Narrow Range Pressurizer Water Level – High Reactor Trip

0-100% Instrument span = 0-100% Pressurizer Level

AL = None, PL = 100% instrument span
TLU = 0.839 * SRSS(Sa, Sd, Sv, Smte, Sp, St, Rd, Rv) + PE
= -14.41% span
Sa = +/- 0.25% span
Sd = +/- 1.105% span
Sv = +/- 0.5% span
Smte = +/- 0.374%
Sp = +/- 1.395% span
St = +/- 2.317% span
Rd = +/- 0.212% span
Rv = +/- 0.5% span
PE = -11.853% span

LTSP = 100 – 14.41 = 85.59% span	(Allowable Value rounded to ≤ 85% span)
FTSP = 80% span	(NTSP)
RAF = SRSS(Rv, Rd) = +/- 0.543% span	(as-found tolerance)
RAL = +/- 0.5% span	(as-left tolerance)

Parameter: Containment Pressure Low Range Instruments
Summary of Calculation No: PBNP-IC-17

High Containment Pressure – Safety Injection

0 – 100% instrument span = 0-60 psig

AL = 6 psig
TLU = 0.839 * SRSS(Sd, Sv, St, Rd, Rv)
= +/- 1.13 % span = +/- 0.678 psi
Sd = +/- 0.518% span
Sv = +/- 0.5% span
St = +/- 1.09% span
Rd = +/- 0.212% span
Rv = +/- 0.25% span

LTSP = 6 – 0.678 = 5.322 psig	(Allowable Value rounded to ≤ 5.3 psig)
FTSP = 5 psig	(NTSP)
RAF = SRSS(Rv, Rd) = +/- 0.328% span	(as-found tolerance)
RAL = +/- 0.25% span	(as-left tolerance)

High-High Containment Pressure – Containment Spray

0 – 100% instrument span = 0-60 psig

AL = 30 psig
TLU = 0.839 * SRSS(Sd, Sv, St, Rd, Rv)
= +/- 1.13 % span = +/- 0.678 psi
Sd = +/- 0.518% span
Sv = +/- 0.5% span
St = +/- 1.09% span
Rd = +/- 0.212% span
Rv = +/- 0.25% span

LTSP = 30 – 0.678 = 29.322 psig	(Allowable Value rounded to $\leq$ 28 psig)*
FTSP = 25 psig	(NTSP)
RAF = SRSS(Rv, Rd) = +/- 0.328% span	(as-found tolerance)
RAL = +/- 0.25% span	(as-left tolerance)

* The containment spray actuation function uses signals from two containment pressure ranges in a two-out-of-three taken twice coincidence logic. The Allowable Value chosen for the function is the most conservative of the LTSPs calculated for the two pressure ranges. The LTSP in calculation PBNP-IC-19 is the basis for the Allowable Value.

Parameter: Containment Pressure Intermediate Range Instruments
Summary of Calculation No: PBNP-IC-19

High-High Containment Pressure – Steam Line Isolation

0 – 100% instrument span = 0 - 90 psig

AL = None, PL = 20 psig
TLU = 0.839 * SRSS(Sd, Sv, St, Rd, Rv)
= +/-1.496% span = +/-1.347 psig
Sd = +/- 0.518% span
Sv = +/- 0.5% span
St = +/- 1.538% span
Rd = +/- 0.212% span
Rv = +/- 0.5% span

LTSP = 20 – 1.347 = 18.653 psig	(Allowable Value rounded to $\leq$ 18 psig)
FTSP = 15 psig	(NTSP)
RAF = SRSS(Rv, Rd) = +/- 0.543% span	(as-found tolerance)
RAL = +/- 0.5% span	(as-left tolerance)

High-High Containment Pressure – Containment Spray Actuation

0 – 100% instrument span = 0 - 90 psig

AL = 30 psig
TLU = 0.839 * SRSS(Sd, Sv, St, Rd, Rv)
= +/-1.496 % span = +/-1.347 psi
Sd = +/-0.518% span
Sv = +/-0.5% span
St = +/-1.538% span
Rd = +/-0.212% span
Rv = +/-0.5% span
LTSP = 30 – 1.347 = 28.653 psig (Allowable Value rounded to $\leq$ 28 psig)
FTSP = 25 psig (NTSP)
RAF = SRSS(Rv, Rd) = +/- 0.543% span (as-found tolerance)
RAL = +/- 0.5% span (as-left tolerance)

Parameter: Steam Line Pressure Instruments
Summary of Calculation No: PBNP-IC-39

Steam Line Pressure – Low Safety Injection

0 -100% instrument span = 0 -1400 psig

AL = 395.3 psig
TLU = 0.839 * SRSS(Sd, Sv, Ss, St, Rd, Rv, La, Ld, Lmte, Lv, PA) + PE
= 8.601% span = 120.41 psig
Sd = +/- 0.518% span
Sv = +/- 0.5% span
St = +/- 8.0% span
Rd = +/- 0.212% span
Rv = +/- 0.5% span
La = +/- 0.5% span
Ld = +/- 0.5% span
Lmte = +/- 0.083% span
Lv = +/- 0.5% span
IR = 1.808% span
LTSP = 395.3 + 120.41 = 515.71 psig (Allowable Value rounded to $\geq$ 520 psig)
FTSP = 530 psig (NTSP)
RAL = +/- 0.5% span (as-left tolerance)
RAF = SRSS(Rv, Rd) = +/- 0.543% span (for Bistable)
RAF = SRSS(Lv, Ld, Lmte) = +/- 0.712% span (for L/L module) (as-found tolerance)

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<u>Parameter: Low Tavg Interlock Instruments</u>
Summary of Calculation Note No: CN-CPS-07-11

<u>Tavg Low Interlock</u>

0 – 100% instrument span = 530 – 630 °F

AL = 540 °F
TLU = See Note 1
= 1.71% span = 1.71 °F
LTSP = 540 + 1.71 = 541.71 °F (Allowable Value rounded to ≥ 542 °F)
FTSP = 543 °F (NTSP)
RAF = SRSS(0.5, 0.196) = +/- 0.537% span (Bistable)(as-found tolerance)
RAF = SRSS(.05, .2, .107) = +/- 0.232 % span = +/- 0.470 mVdc (Current Source)
RAF = SRSS(0.083, 0.429, 0.033) = 0.438 % span (Amplifier)
RAF = SRSS(0.5, 0.061) = 0.504 % span = +/- 2.1 mVdc (E/I Converter)
RAL = +/- 0.5% span (Bistable) (as-left tolerance)
RAL = +/- 0.10 mVdc (Current Source)
RAL = +/- 0.083% span (Amplifier)
RAL = +/- 0.5% (E/I Converter)

Note 1 – Because of the complexity of this calculation, please refer to the actual calculation for details