ENCLOSURE 1

Point Beach Units 1 and 2

License Amendment Request to Revise Technical Specifications
to Adopt Risk Informed Completion Times TSTF-505, Revision 2,
"Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

List of Revised Required Actions to Corresponding PRA Functions and Additional Supporting Information

1. INTRODUCTION

In accordance with the NRC's revised model application for TSTF 505, Revision 2 (Reference 1), Enclosure 1 of the amendment request is to provide confirmation of the PRA models, including the necessary scope of structures, systems, and components (SSC) and their functions for each proposed application of the Risk-Informed Completion Time (RICT) Program to the TS LCO Conditions and Required Actions.

This enclosure provides confirmation that the Point Beach PRA model includes the necessary scope of SSCs and their functions to address each proposed application of the RICT Program. The enclosure addresses the applicable design basis functions, how each are modeled in the Point Beach PRA, and provides justification for the use of proposed surrogates to adequately capture configuration risk, where applicable. The enclosure also provides a comparison of the success criteria used in the PRA model to the design basis success criteria at a train and component/parameter level. The comparison addresses each of the TS LCO Conditions and associated Required Actions proposed for the Point Beach RICT Program, as identified in the TS markup pages of Attachment 2 of this amendment request. Also provided are additional justifications for the specific TS Required Actions recommended in the NRC's Final Revised Model Safety Evaluation for TSTF-505, Revision 2 (Reference 5) and information to support instrumentation redundancy and diversity, as also recommended in the NRC's Revised Model Safety Evaluation.

1.0 SCOPE

Table E1-1 below lists each TS Required Action Condition proposed for the Point Beach RICT Program and documents information regarding the associated SSCs credited in plant safety analyses, the analogous PRA functions, and the results of the comparison. The 'Comments' column provides where applicable, a disposition of inconsistencies between the TS and PRA functions regarding the scope of SSCs and the success criteria. Where the PRA scope of SSCs is not consistent with the TS, additional information is provided to describe how the Required Action Condition can be evaluated using appropriate surrogate events which permit a risk evaluation to be completed using the Configuration Risk Management Program (CRMP) tool for the RICT Program. Differences in success criteria typically arise due to the RG 1.200 (Reference 2) requirement to employ realistic as-built, as-operated criteria, whereas design basis criteria are necessarily conservative and bounding. These differences are addressed to demonstrate that the PRA criteria provide a realistic estimate of the risk of the TS condition as required by NEI 06-09 (Reference 3).

References:

1. NRC Revised TSTF 505, Revision 2, Model Application, (ADAMS Accession No. ML18115A482)

2. Regulatory Guide (RG) 1.200, Revision 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, March 2009 (ADAMS Accession No. ML090410014)

3. NEI 06-09 (Revision 0), Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines', Industry Guidance Document (ADAMS Accession No. ML063390639)

4. NRC letter to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)

5. NRC Safety Evaluation, "Final Revised Model Safety Evaluation of Traveler TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'", dated November 21, 2018 (ADAMS Accession No. ML18269A041)

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.3.1, RPS Instrumentation (Table 3.3-1)						
Condition B One Manual Reactor Trip channel inoperable (Modes 1,2)	(FU1) Two Manual Rx Trip channels	Reactor Trip Initiation	No	1 of 2 Manual Rx Trip channels	Not Modeled – See comments	The operator actions for failure to actuate a manual reactor trip will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Condition D One channel inoperable	(FU2a) Four Power Range Neutron Flux High channels	Reactor Trip Initiation	No	2 of 4 Power Range Neutron Flux- High channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU2b) Four Power Range Neutron Flux Low channels	Reactor Trip Initiation	No	2 of 4 Power Range Neutron Flux- Low channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU5) Four Overtemperature ΔT channels	Reactor Trip Initiation	No	2 of 4 Overtemperature ΔT channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU6) Four Overpower ΔT channels	Reactor Trip Initiation	No	2 of 4 Overpower ΔT channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU7b) Three Pressurizer Pressure - High channels	Reactor Trip Initiation	No	2 of 3 Pressurizer Pressure - High channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU13) Three SG Water Level Low-Low channels per SG	Reactor Trip Initiation	No	2 of 3 SG Water Level Low-Low channels on any SG	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU14) Two SG Water Level Low; Coincident w/ Steam-Flow/ Feed-Flow Mismatch channels per SG	Reactor Trip Initiation	No	One SG Water Level Low coincident w/ one Steam-Flow/ Feed-Flow Mismatch channel on any SG	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
Condition E One channel inoperable	(FU12) Two Underfrequency Bus A01, A02 channels per bus	Reactor Trip Initiation	No	1 of 2 Underfrequency Bus channels on both A01 and A02	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
Condition K One channel inoperable	(FU7a) Four Pressurizer Pressure - Low channels	Reactor Trip Initiation	No	2 of 4 Pressurizer Pressure - Low channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU8) Three Pressurizer Water Level - High channels	Reactor Trip Initiation	No	2 of 3 Pressurizer Water Level - High channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU9b) Three Reactor Coolant Flow - Low (Two loops) channels per RCS Loop	Reactor Trip Initiation	No	2 of 3 Reactor Coolant Flow - Low channels on both RCS Loops when above P8	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU11) Two Undervoltage Bus A01, A02 channels per electrical bus	Reactor Trip Initiation	No	1 of 2 Undervoltage Bus channels on A01 and A02	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
Condition L One Reactor Coolant Flow-Low (Single Loop) channel inoperable	(FU9a) Three Reactor Coolant Flow - Low (Single Loop) channels per RCS Loop	Reactor Trip Initiation	No	2 of 3 Reactor Coolant Flow - Low channels on any RCS Loop when below P8	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
Condition M One Reactor Coolant Pump Breaker Position (Single Loop) channel inoperable	(FU10a) One RCP Breaker Position (Single Loop) channel per RCP	Reactor Trip Initiation	No	One RCP Breaker Position channel on any RCP when below P8	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
Condition N One inoperable channel	(FU10b) One RCP breaker position (two loops) channel per RCP	Reactor Trip Initiation	No	One RCP breaker Position channel on both RCPs when above P8	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
Condition O One turbine trip channel inoperable	(FU15a) Three Turbine trip - Low Auto-Stop Oil Pressure channels	Reactor Trip Initiation	No	2 of 3 Turbine trip - Low Auto-Stop Oil Pressure channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU15b) Two Turbine trip - Stop Valve Closure channels	Reactor Trip Initiation	No	2 of 3 Turbine trip - Stop Valve Closure channels	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
Condition P One train inoperable (Modes 1,2)	(FU16) Two SI input from ESFAS trains	Reactor Trip Initiation	No	1 of 2 SI input from ESFAS trains	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
	(FU21) Two Automatic Trip Logic trains	Reactor Trip Initiation	No	1 of 2 Automatic Trip Logic trains	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
Condition Q One RTB inoperable (Modes 1,2)	(FU18) Two RTB trains	Reactor Trip Initiation	No	1 of 2 RTB trains	Not Modeled – See comments	This SSC is used as a surrogate for other TS 3.3.1 RPS Instrumentation Conditions.
Condition U One trip mechanism inoperable for one RTB (Modes 1,2)	(FU19) One RTB Undervoltage and Shunt Trip Mechanism per RTB train	Reactor Trip Initiation	No	One RTB Undervoltage and Shunt Trip Mechanism on any RTB train	Not Modeled – See comments	The condition of one of two inoperable reactor trip breakers is used as a surrogate for this condition.
TS 3.3.2, ESFAS Instrumentation (Table 3.3-2)						
Condition B One channel inoperable	(FU1a) Two Manual Initiation channels	Safety Injection Initiation	No	1 of 2 Manual Initiation channels	Not Modeled – See comments	The operator actions for failure to manually actuate SI will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
	(FU3a) Two Manual Initiation channels	Containment Isolation Initiation	No	1 of 2 Manual Initiation channels	Not Modeled – See comments	The condition of manual SI function inoperable is used as a surrogate for this condition since an SI signal generates a CI signal.

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
Condition C One train inoperable.	(FU1b) Two Automatic Actuation Logic and Actuation Relay trains	Safety Injection Initiation	No	1 of 2 Automatic Actuation Logic and Actuation Relay trains channels	Not Modeled – See comments	The failure of the automatic SI signals will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
	(FU3b) Two Automatic Actuation Logic and Actuation Relay trains	Containment Isolation Initiation	No	1 of 2 Automatic Actuation Logic and Actuation Relay trains	Not Modeled – See comments	The condition of automatic SI function inoperable is used as a surrogate for this condition since an SI signal generates a CI signal.
Condition D One channel inoperable.	(FU1c) Three Containment Pressure - High channels	Safety Injection Initiation	No	2 of 3 Containment Pressure - High channels	Not Modeled – See comments	The condition of automatic SI function inoperable is used as a surrogate for this condition since it is the same function (SI initiation).
	(FU1d) Three Pressurizer Pressure - Low channels	Safety Injection Initiation	No	2 of 3 Pressurizer Pressure - Low channels	Not Modeled – See comments	The condition of automatic SI function inoperable is used as a surrogate for this condition since it is the same function (SI initiation).
	(FU1e) Three Steam Line Pressure - Low channels per steam line	Safety Injection Initiation	No	2 of 3 Steam Line Pressure - Low channels per main steam line	Not Modeled – See comments	The condition of automatic SI function inoperable is used as a surrogate for this condition since it is the same function (SI initiation).
	(FU4c) Three Containment Pressure - High, High channels	Steam Line Isolation Initiation	No	2 of 3 Containment Pressure - High, High channels	Not Modeled – See comments	The failure of the model logic for steam generator isolation will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
	(FU4d) Two High Steam Flow channels; Coincident with SI and Coincident with three T_{avg} - Low, Low channels per RCS loop	Steam Line Isolation Initiation	No	1 of 2 High Steam Flow channels coincident with SI and coincident with 2 of 3 T_{avg} - Low, Low channels	Not Modeled – See comments	The condition of steam generator isolation function inoperable is used as a surrogate for this condition.

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
	(FU4e) Two High, High Steam Flow channels per steam line; Coincident with SI	Steam Line Isolation Initiation	No	1 of 2 High, High Steam Flow channels per steam line coincident with SI	Not Modeled – See comments	The condition of steam generator isolation function inoperable is used as a surrogate for this condition.
	(FU5b) Three SG Water Level - High channels per SG	Feedwater Isolation Initiation	No	2 of 3 SG Water Level - High channels per SG	Not Modeled – See comments	The condition of steam generator isolation function inoperable is used as a surrogate for this condition.
	(FU6b) Three SG Water Level - Low, Low channels per SG	Auxiliary Feedwater Initiation	No	2 of 3 SG Water Level - Low, Low channels on any SG	Not Modeled – See comments	The failure of the model logic for these relays will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Condition F One channel inoperable	(FU4a) One Manual Initiation channel per RCS loop	Steam Line Isolation Initiation	No	One Manual Initiation channel per RCS loop	Not Modeled – See comments	The condition of steam generator isolation function inoperable is used as a surrogate for this condition.
Condition G One train inoperable	(FU4b) Two Automatic Actuation Logic and Actuation Relay trains	Steam Line Isolation Initiation	No	1 of 2 Automatic Actuation Logic and Actuation Relay trains	Not Modeled – See comments	The condition of steam generator isolation function inoperable is used as a surrogate for this condition.
	(FU5a) Two Automatic Actuation Logic and Actuation Relay trains	Feedwater Isolation	No	1 of 2 Automatic Actuation Logic and Actuation Relay trains	Not Modeled – See comments	The condition of AFW initiation is used as a surrogate for this condition.
	(FU6a) Two Automatic Actuation Logic and Actuation Relay trains	Auxiliary Feedwater	No	1 of 2 Automatic Actuation Logic and Actuation Relay trains	Not Modeled – See comments	The failure of the model logic for these relays will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
Condition H One channel inoperable	(FU6d) Two Undervoltage Bus A01 and A02 channels per electrical bus	Auxiliary Feedwater	No	1 of 2 Undervoltage Bus channels on A01 and A02	Not Modeled – See comments	The failure of the model logic for starting all four AFW pumps will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.4, Reactor Coolant System						
TS 3.4.11 Condition B One PORV inoperable and not capable of being manually cycled	Two Power Operated Relief Valves (PORVs)	RCS pressure control	Yes	1 of 2 PORVs	Same	
TS 3.4.11 Condition C One block valve inoperable	Two PORV Block Valves	RCS integrity	Yes	Associated Block Valve closure	Same	
TS 3.5, Emergency Core Cooling System (ECCS)						
TS 3.5.2 Condition A One ECCS train inoperable	Two ECCS trains each comprised of one SI pump, one RHR pump, one RHR heat exchanger and associated RWST and containment sump flowpaths	Emergency core cooling and post-accident (long-term) core cooling	Yes	1 of 2 SI pumps, and 1 of 2 RHR pumps 1 of 2 RHR pumps w/ suction from containment sump, supplying suction to 1 of 2 SI pumps for flowpath to RCS	Same	
TS 3.6, Containment Systems						
TS 3.6.2 Condition C One or more containment air locks inoperable for reasons other Condition A or B	One equipment hatch; One personnel airlock; Two emergency airlocks	Containment integrity	No	One of two containment air lock doors closed with acceptable containment leakage	Not Modeled – See comments	The failure of the model logic for containment penetrations will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.6.3 Condition A One or more penetration flow paths with one containment isolation valve inoperable (applicable to penetration flow paths with two containment isolation valves)	Two isolation valves on each containment penetration	Containment integrity	No	1 of 2 isolation valves per penetration isolate	Not Modeled – See comments	The failure of the model logic for containment penetrations will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
TS 3.6.3 Condition C One or more penetration flow paths with one containment isolation valve inoperable (applicable to penetration flow paths with one containment isolation valve and a closed system)	One isolation valve and one closed system on each containment penetration	Containment integrity	No	Each isolation valve per penetration isolates	Not Modeled – See comments	The failure of the model logic for containment penetrations will be used as a surrogate to conservatively bound the risk increase associated with this function as permitted by NEI 06-09.
TS 3.7, Plant Systems						
TS 3.7.2 Condition A One Steam Generator flowpath with one or more inoperable valves in MODE 1	Two Main Steam Lines equipped with one Main Steam Isolation Valve (MSIV) and one Non-Return Check Valve	Steam Line Isolation from Faulted Steam Line	Yes	MSIV on affected steam line isolates	Same	

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.7.4 Condition A One required ADV flowpath inoperable	Two Atmospheric Dump Valves (ADVs)	Facilitate Unit cooldown to SDC conditions	Yes	1 of 2 ADV flowpaths	Same	
TS 3.7.5 Condition A Turbine driven AFW pump system inoperable due to one inoperable steam supply, OR Turbine driven AFW pump system inoperable in MODE 3 following refueling	One turbine-driven AFW pump and one motor-driven AFW pump and associated CST and SW suction piping	Feedwater supply to SGs upon loss of main feedwater	Yes	One motor driven AFW pump supplies CST feedwater to both SGs	Same	
TS 3.7.5 Condition B One AFW pump system inoperable in MODE 1, 2 or 3 for reasons other than Condition A	One turbine-driven AFW pump and one motor-driven AFW pump and associated CST and SW suction piping	Feedwater supply to SGs upon loss of main feedwater	Yes	One turbine driven AFW pump supplies CST feedwater to both SGs	Same	
TS 3.7.7 Condition A One CC pump inoperable	Two Component Cooling Water (CCW) trains each consisting of one CCW pump and one CCW heat exchanger, and one common CCW heat exchanger capable of aligning to either train	Heat sink for removing process and operating heat from safety-related components	Yes	1 CCW pump and 1 CCW HX provide heat sink to safety related equipment	Same	

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.7.7 Condition B One required CC heat exchanger inoperable	Two Component Cooling Water (CCW) trains each consisting of one CCW pump and one CCW heat exchanger, and one common CCW heat exchanger capable of aligning to either train	Heat sink for removing process and operating heat from safety-related components	Yes	1 CCW pump and 1 CCW HX provide heat sink to safety related equipment	Same	
TS 3.7.8 Condition A One SW pump inoperable AND Both units in Modes 1, 2, 3, or 4	Six Service Water (SW) system pumps, one common ring header, non-essential flowpath isolation valves and associated SW intake piping	Heat sink for removing process and operating heat from safety-related components	Yes	2 SW pumps and 1 SW Ring Header provide heat sink to CCW system and essential loads; auto-isolate non-essential flowpaths	Same	
TS 3.7.8 Condition C SW ring header continuous flowpath interrupted	Six Service Water (SW) system pumps, one common ring header, non-essential flowpath isolation valves and associated SW intake piping	Heat sink for removing process and operating heat from safety-related components	Yes	2 SW pumps and 1 SW Ring Header provide heat sink to CCW system and essential loads; auto-isolates non-essential flowpaths	Same	
TS 3.7.8 Condition D One or more non-essential-SW-load flowpath(s) with one required automatic isolation valve inoperable. AND Affected non-essential flowpath(s) not isolated	Six Service Water (SW) system pumps, one common ring header, non-essential flowpath isolation valves and associated SW intake piping	Heat sink for removing process and operating heat from safety-related components	Yes	3 SW pumps and 1 SW Ring Header provide heat sink to CCW system and essential loads; auto-isolates non-essential flowpaths	Same	

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.8, Electrical Power Systems						
TS 3.8.1 Condition A Associated unit 345/13.8 kV (X03) transformer inoperable. OR Gas turbine not in operation when utilizing opposite unit's 345/13.8 kV (X03) transformer.	Two 13.8 kV station auxiliary transformers; one 13.8 kV gas turbine; two Class 1E 4.16 kV electrical buses; four 4.16 kV diesel generators (DG)	Power onsite safeguards buses from offsite and onsite transmission networks to support normal, safe shutdown and accident mitigation conditions	Yes	Automatically power associated ESF busses	Same	
TS 3.8.1 Condition B Associated unit's 13.8/4.16kV (X04) transformer inoperable	Two 13.8 kV station auxiliary transformers; one 13.8 kV gas turbine; two Class 1E 4.16 kV electrical buses; four 4.16 kV diesel generators (DG)	Power onsite safeguards buses from offsite and onsite transmission networks to support normal, safe shutdown and accident mitigation conditions	Yes	Automatically power associated ESF busses	Same	

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
TS 3.8.1 Condition C Associated unit's required offsite power source to buses A05 and A06 inoperable. OR Required offsite power source to buses 1A05 and 2A06 inoperable	Two 13.8 kV station auxiliary transformers; one 13.8 kV gas turbine; two Class 1E 4.16 kV electrical buses; four 4.16 kV diesel generators (DG)	Power onsite safeguards buses from offsite and onsite transmission networks to support normal, safe shutdown and accident mitigation conditions	Yes	Automatically power associated ESF busses	Same	
TS 3.8.1 Condition D One or more required offsite power source(s) to one or more required Class 1E 4.16 kV bus(es) inoperable	Two 13.8 kV station auxiliary transformers; one 13.8 kV gas turbine; two Class 1E 4.16 kV electrical buses; four 4.16 kV diesel generators (DG)	Power onsite safeguards buses from offsite and onsite transmission networks to support normal, safe shutdown and accident mitigation conditions	Yes	Automatically power associated ESF busses	Same	
TS 3.8.1 Condition F One or more required offsite power source to one or more Class 1E 4.16 kV safeguards bus(es) inoperable AND Standby emergency power inoperable to	Two 13.8 kV station auxiliary transformers; one 13.8 kV gas turbine; two Class 1E 4.16 kV electrical buses; four 4.16 kV diesel generators (DG)	Power onsite safeguards buses from offsite and onsite transmission networks to support normal, safe shutdown and accident	Yes	Automatically power associated ESF busses	Same	

Table E1-1 - List of Revised Required Actions to Corresponding PRA Functions

Required Action Condition Description	Applicable SSCs	SSC Function(s)	PRA Modeled	Design Success Criteria	PRA Success Criteria	Comments
redundant equipment		mitigation conditions				
TS 3.8.4 Condition A One DC electrical power subsystem inoperable	4 battery banks and associated chargers and motor control centers (MCC)	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	Yes	Automatically power associated ESF busses	Same	
TS 3.8.7 Condition A One required inverter inoperable	Four 120 VAC Instrument Inverters	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	Yes	Two 120 VAC Instrument Inverters per electrical train	Same	

2.0 MODEL CONFIRMATION

Table E1-2 provides the calculated RICT for each individual condition to which the RICT applies (assuming no other structures, systems, and components (SSCs) modeled in the PRA are unavailable). Table E1-2 confirms that the PRA models include the necessary SSCs and their functions to address each proposed application of the RICT Program to the TS Required Actions. The RICT estimates are based on the internal events, internal flooding, and internal fire PRA model calculations with seismic CDF and LERF penalties and are the most limiting of the Point Beach Unit 1 and Unit 2 results. Actual RICT values will be calculated based on the actual plant configuration using a current revision of the PRA model which represents the as-built, as-operated condition of the plant, as required by NEI 06-09-A (Reference 1) and the NRC safety evaluation (Reference 2), and may differ from the RICTs presented in Table E1-2 below. RICTs calculated to be greater than 30 days are capped at 30 days based on NEI 06-09-A. RICTs not capped at 30 days are rounded to nearest number of days. TS Required Action Conditions with insufficient TS operable equipment to meet the specified safety function of the system are not eligible for RICT Program application. Consistent with NEI 06-09-A, cases where the total CDF or LERF is greater than 1E-03/year or 1E-04/year, respectively, are not eligible for RICT Program application.

References:

1. NRC letter to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)

2. NRC letter to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)

Table E1-2 - In-Scope TS/LCO Conditions RICT Estimate

TS/LCO	Required Action Condition Description	RICT Estimate (days)
Reactor Protection System (RPS) Instrumentation		
3.3.1.B	One Manual Reactor Trip channel inoperable	30
3.3.1.D	One channel inoperable	30
3.3.1.E	One channel inoperable	30
3.3.1.K	One channel inoperable	30
3.3.1.L	One Reactor Coolant Flow-Low (Single Loop) channel inoperable	30
3.3.1.M	One Reactor Coolant Pump Breaker Position, (Single Loop) channel inoperable	30
3.3.1.N	One channel inoperable	30
3.3.1.O	One turbine trip channel, inoperable	30
3.3.1.P	One train inoperable, (Modes 1,2)	30
3.3.1.Q	One RTB inoperable, (Modes 1,2)	30
3.3.1.U	One trip mechanism, inoperable for one RTB (Modes 1,2)	30
Engineered Safety Feature Actuation System (ESFAS) Instrumentation		
3.3.2.B	One channel inoperable	30
3.3.2.C	One channel inoperable	30
3.3.2.D	One channel inoperable	30
3.3.2.F	One channel inoperable	30
3.3.2.G	One train inoperable	30
3.3.2.H	One channel inoperable	18
Pressurizer Power Operated Relief Valves (PORVs)		
3.4.11.B	One PORV inoperable and not capable of being manually cycled	30
3.4.11.C	One block valve, inoperable	30
ECCS - Operating		
3.5.2.A	One ECCS train inoperable	30
Containment Air Locks		
3.6.2.C	One or more, containment air locks inoperable for reasons other than Condition A or B	9
Containment Isolation Valves		
3.6.3.A	One or more penetration flow paths with one containment isolation valve inoperable	9
3.6.3.C	One or more penetration flow paths with one containment isolation valve inoperable	9
Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves		
3.7.2.A	One Steam Generator, flowpath with one or more inoperable valves in MODE 1	30
Atmospheric Dump Valve (ADV) Flowpaths		
3.7.4.A	One required ADV flowpath inoperable	30
Auxiliary Feedwater (AFW)		
3.7.5.A	Turbine driven AFW pump system inoperable due to one inoperable steam supply, OR Turbine driven AFW pump system inoperable in MODE 3 following refueling	30
3.7.5.B	One AFW pump system inoperable in MODE 1, 2 or 3 for reasons other than Condition A	30

Table E1-2 - In-Scope TS/LCO Conditions RICT Estimate

TS/LCO	Required Action Condition Description	RICT Estimate (days)
Component Cooling Water (CC) System		
3.7.7.A	One CC pump, inoperable	23
3.7.7.B	One required CC heat exchanger inoperable	30
Service Water (SW) System		
3.7.8.A	One SW pump inoperable AND Both units in Modes 1, 2, 3, or 4	30
3.7.8.C	SW ring header, continuous flowpath, interrupted	30
3.7.8.D	One or more non-essential-SW-load flowpath(s) with one required automatic isolation valve inoperable AND Affected non-essential flowpath(s), not isolated.	30
AC Sources - Operating		
3.8.1.A	Associated unit 345/13.8 kV (X03) transformer inoperable OR Gas turbine not in operation when utilizing opposite unit’s 345/13.8 kV (X03) transformer.	30
3.8.1.B	Associated unit's 13.8/4.16kV (X04) transformer, inoperable	30
3.8.1.C	Associated unit's required offsite power source to buses A05 and A06 inoperable OR Required offsite power source to buses 1A05 and 2A06 inoperable.	30
3.8.1.D	One or more required offsite power source(s) to one or more required Class 1E 4.16 kV bus(es) inoperable.	30
3.8.1.F	One or more required offsite power source to one or more Class 1E 4.16 kV safeguards bus(es) inoperable AND Standby emergency power inoperable to redundant equipment.	8
DC Sources - Operating		
3.8.4.A	One DC electrical power subsystem inoperable	7
Inverters - Operating		
3.8.7.A	One required inverter inoperable	30

3.0 ADDITIONAL JUSTIFICATION FOR SPECIFIC ACTIONS

The NRC Final Revised Model Safety Evaluation for TSTF-505, Revision 2 (Reference 1) provides a listing of TS Conditions for inclusion in licensee RICT Programs that are recommended for additional technical justification, as specified in Table 1, *Conditions Requiring Additional Technical Justification*, for NUREG-1431, Westinghouse STS plants. The discussion below addresses the Point Beach TS Conditions that are applicable to the NRC's additional technical justification recommendations along with the bases for the conclusion that the proposed changes do not result in any loss of a specified safety function.

1. TS 3.3.1, Table 3.3.1-1, FU 2a, Power Range Neutron Flux - High channels

   Condition D: One channel inoperable.
   Required ACTION: Place channel in trip

   Four power range neutron flux - high channels are provided for overpower protection. The trip logic accounts for the power range nuclear flux - high channels also providing a rod control function whereby the system must be capable of withstanding a second channel failure to satisfy the single failure criterion. The 2 out of 4 trip logic defaults to a 2 out of 3 logic in the event of a failed or bypassed power range neutron flux-high channel. The unavailability of a single power range neutron flux-high channel will not impair reactor core power distribution monitoring or control rod regulation since the rod control signals are based on the average of the power range neutron flux-high signals and the rod control system only responds to rapid changes in neutron flux. All neutron flux power range currents are indicated in the control room. If a power range channel failure occurs, switches are provided to permit the failed power range channel's overpower rod stop function to be bypassed and its average power signal to the reactor control system replaced by a signal derived from an active channel. This allows normal power operation to continue while the failed channel is repaired. Alarms are also provided to alert the operator of deviations from normal operating conditions so that corrective action can be taken prior to reaching a reactor trip setting. Thereby, the inoperability of a power range neutron flux-high channel does not result in a loss of function.

2. TS 3.3.1, Table 3.3.1-1, FU 18, Condition Q for the Reactor Trip Breakers (RTBs), MODES 1, 2

   Condition Q: One RTB inoperable.
   Required ACTION: Restore RTB to OPERABLE status

   An RTB train consists of all trip breakers associated with a single RTS logic train that are racked in and capable of supplying power to the rod control system. Two devices in each breaker receive RPS signals, either of which will trip the RTBs via an undervoltage trip device which will trip the reactor on loss of breaker control power or the receipt of an RPS trip signal, or a shunt trip device which provides a backup if the passive undervoltage trip fails to open the reactor trip breakers upon receipt of a trip signal. In addition, an Anticipated Transient Without Scram Mitigating System Actuation Circuitry (AMSAC) system is provided in the event of an anticipated trip without scram (ATWS) condition to further mitigate the effects of a failed RPS by tripping the turbine and starting AFW in addition to inserting the control rods. Thereby, sufficient redundancy and diversity exists in the RTB system to assure the unavailability of a single RTB does not result in a loss of function.

3. TS 3.5.2, ECCS - Operating

   Condition A: One ECCS train inoperable.
   Required ACTION: Restore train to OPERABLE status

   Additional justification is not needed for this TS Condition since Point Beach TS 3.5.2, Condition A, addresses only one inoperable ECCS train.

4. TS 3.6.2, Containment Air Locks

Condition C: One or more containment air locks inoperable for reasons other than Condition A or B.

Required ACTION:
C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1, AND
C.2 Verify a bulkhead door and associated equalizing valve are closed in the affected air lock, AND
C.3 Restore air lock to OPERABLE status

The containment airlocks are comprised of a containment penetration chamber isolated by inner and outer bulkheads equipped with double o-ring seals. In the event of an inoperable containment airlock, ACTION C.1 requires the condition to be immediately assessed in accordance with TS 3.6.1 (i.e., immediately initiate action to evaluate overall primary containment leakage). In addition, compliance TS 3.6.2, ACTION C.2, assures the presence of at least one physical barrier on the affected airlock penetration with acceptable barrier leakage in accordance with the Point Beach Containment Leakage Rate Testing Program. Thereby, containment integrity function is maintained throughout the period of airlock inoperability consistent with plant safety analyses.

5. TS 3.7.2, Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves

Condition A: One Steam Generator flowpath with one or more inoperable valves in MODE 1
Required ACTION: Restore valve to OPERABLE status.

TS 3.7.2 specifies requirements for the MSIVs, which mitigate a steam line break by automatically closing upon receipt of a safety injection (SI) signal coincident with a Hi-Hi steam flow signal. Rapid MSIV closure limits RCS cooldown and the resulting reactivity addition. Automatic MSIV closure also prevents the uncontrolled blowdown of a SG resulting from a downstream steam line break. In addition, TS 3.7.2 specifies requirements for the main steam non-return check valves, which close on reverse flow to prevent the blowdown of a non-faulted SG in the event of a main steam line break coincident with the failure of a MSIV to close properly. Main steam non-return check valve closure assures that for any steam break location coupled with an MSIV single failure, both SGs will not blow down and cause core damage due to excess positive reactivity from the RCS cooldown transient. Thereby, the loss of a single MSIV will not result in a loss of the steam line isolation function.

References:

1. NRC Safety Evaluation, “Final Revised Model Safety Evaluation of Traveler TSTF-505, Revision 2, ‘Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b’”, dated November 21, 2018 (ADAMS Accession No. ML18269A041)

4.0 INFORMATION TO SUPPORT INSTRUMENTATION REDUNDANCY AND DIVERSITY

The NRC’s revised model application for TSTF 505, Revision 2 (Reference 1), recommends for the proposed changes to the protective instrumentation features in TS Section 3.3, "Instrumentation," a demonstration that at least one redundant or diverse means remains available to accomplish the associated safety function(s) during application of a RICT, consistent with the defense-in-depth philosophy of Regulatory Guide (RG) 1.174 (Reference 2). The request is in recognition that while in an ACTION statement, redundancy of the protective feature, and thereby system reliability, is temporarily reduced. Table E1.1 of this enclosure provides a description of the PRA modeling for RPS and ESFAS instrumentation, including the scope of the TS functions to which a RICT would be applied. Sections 4.1 and 4.2 below demonstrate that diversity and redundancy is maintained during the application of a RICT by demonstrating the existence of at least one additional means to accomplish the safety function(s) .

4.1 TS 3.3.1 - Reactor Protection System (RPS) Instrumentation

The Point Beach RPS consists of four instrument channels that monitor up to four various plant parameters, depending on the coincidence logic required for the specific trip. Each protection channel terminates at a channel trip bistable in the analog protection racks. Each channel trip bistable controls two independent and redundant logic relays associated with the two independent and redundant trains ("A" and "B"). The logic relays for each train are combined in a coincidence logic network (e.g., two-out-of-four). Two independent and redundant reactor trip breakers in series provide power to the control rod drive mechanisms. In addition, two independent and redundant bypass breakers are provided in parallel with the reactor trip breakers to allow for continued reactor operation during testing of the reactor trip breakers. When the required number of channels (e.g., two-out-of-four) indicate that a plant parameter is outside its acceptable operating limit, their associated channel bistables are tripped. The tripping of the channel bistables result in the tripping of their associated coincidence logic relays for each train, which in turn results in de-energizing the reactor trip relays. De-energizing the reactor trip relays causes the associated train trip breaker to open by de-energizing its undervoltage trip coil and by energizing its shunt trip coil through an interposing relay. De-energizing the reactor trip relays also causes the opposite train bypass breaker to open by de-energizing its undervoltage trip coil. When the reactor trip breakers are tripped, power to the control rod drive mechanisms is interrupted, which allows the control rods to insert into the core by gravity.

The RPS is designed so that no single failure within, or in an associated system which supports RPS operation, will prevent the intended reactor trip function. The RPS is redundant and independent for all primary inputs and functions. Each RPS channel is functionally independent of every other RPS channel and receives power from a separate AC power source. Train separation is achieved by providing separate racks and each train receives power from a separate DC power source such that each RPS train is functionally independent of the redundant train. The RPS is designed to IEEE Standard 279-1968 (Reference 3) except for some backup/anticipatory reactor protection which may not fully conform to all IEEE 279 criteria, as described in the Point Beach UFSAR.

The primary reactor trip functions are the overpower ΔT, overtemperature ΔT, and nuclear overpower trips, which define the allowable region of reactor power and coolant temperature conditions. The high pressurizer water level, loss of RCS flow, steam and feedwater flow mismatch, steam generator low-low level, turbine, safety injection, nuclear source and intermediate range, and manual trip functions are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

RPS diversity and redundancy in trip actuation capability is depicted in Table E1-3(1) below.

References:

1. NRC Revised TSTF 505, Revision 2, Model Application (ADAMS Accession No. ML18115A482)

2. Regulatory Guide 1.174. An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, January 2018 (ADAMS Accession No. ML17317A256)

3. IEEE 279, Proposed IEEE Criteria for Nuclear Power Plant Protection Systems, dated August 1968.

Table E1-3(1) - Information to Support RPS Instrumentation Redundancy and Diversity

RPS Function	Functional Unit (FU)	Channels to Trip (1)(2)(3)	Primary Transient / Accident Protection (4)	Diverse Actuation Functions() (1)	Comments
Power Range Neutron Flux (high setting)	FU2a	2 out of 4	– Uncontrolled RCCA Withdrawal at Power – Uncontrolled RCCA Withdrawal from a Subcritical Condition – Steam Line Break – Loss of External Electrical Load – Loss of RCS Coolant	– Overtemperature ΔT – Overpower ΔT – Pressurizer Pressure - High – Pressurizer Level - High – Manual Trip	No interlocks (i.e. always active)
Power Range Neutron Flux (low setting)	FU2b	2 out of 4	– Uncontrolled RCCA Withdrawal from a Subcritical Condition – Loss of RCS Coolant	– Source Range Flux – Intermediate Range Flux – Power Range Neutron Flux - High – Manual Trip	Manual block above P10 interlock (~10% RTP) allowed. Automatically reinstated below P10.
Overtemperature ΔT	FU5	2 out of 4	– Uncontrolled RCCA Withdrawal from a Subcritical Condition – Uncontrolled RCCA Withdrawal at Power RCCA Drop – Loss of RCS Flow – RCCA Drop – Loss of External Electrical Load – Loss of Normal Feedwater – SG Tube Rupture – Loss of RCS Coolant	– Power Range Neutron Flux - High – Overpower ΔT – Pressurizer Pressure - High – Pressurizer Level - High – SG Level Low-Low – Turbine Trip – Manual Trip	No interlocks. Requires two sets of temperature measurements above trip setpoint per RCS loop
Overpower ΔT	FU6	2 out of 4	– Uncontrolled RCCA Withdrawal at Power – RCCA Drop – Steam Line Break – CVCS Malfunction	– Source Range Neutron Flux – Manual Trip	No interlocks. Requires two sets of temperature measurements above trip setpoint per RCS loop

Table E1-3(1) - Information to Support RPS Instrumentation Redundancy and Diversity

RPS Function	Functional Unit (FU)	Channels to Trip (1)(2)(3)	Primary Transient / Accident Protection (4)	Diverse Actuation Functions() (1)	Comments
Pressurizer Pressure - High	FU7a	2 out of 3	– Uncontrolled RCCA Withdrawal at Power – Loss of External Electrical Load – Loss of RCS Flow – Loss of Normal Feedwater	– Power Range Neutron Flux - High – Overtemperature ΔT – Pressurizer Level - High – SG Level Low-Low – Turbine Trip – Manual Trip	No interlocks
Pressurizer Pressure - Low	FU7b	2 out of 4	– RCCA Drop – SG Tube Rupture – Steam Line Break – Loss of RCS Coolant	– Overtemperature ΔT – Overpower ΔT – Safety Injection – Manual Trip	Automatic block below P-7 Reinstated above P-7
Pressurizer Level - High	FU8	2 out of 3	– Uncontrolled RCCA Withdrawal at Power – Loss of External Electrical Load – Loss of Normal Feedwater	– Pressurizer Pressure - High – Power Range Neutron Flux - High – Overtemperature ΔT – Manual Trip	Automatic block below P-7
RCS Flow - Low (one loop) (two loops)	FU9a FU9b	2 out of 3	– Loss of RCS Flow – Loss of All AC Power to the Auxiliaries	– Overtemperature ΔT – Pressurizer Pressure - High – Undervoltage Bus A01, A02 – Underfrequency Bus A01, A02 – RCP Breaker position – Manual Trip	One loop trip requires 2 of 3 channels on either RCS loop. Blocked below P8 interlock (~35% RTP). Reinstated above P8. Two loop trip requires 2 of 3 channels on both RCS loops. Blocked below P7 (~10% RTP). Reinstated above P7.
RCP breaker position - one loop RCP breaker position - 2 loops	FU10a FU10b	1 out of 1 per loop	– Loss of RCS Flow – Loss of All AC Power to the Auxiliaries	– Overtemperature ΔT – Pressurizer Pressure - High – Undervoltage Bus A01, A02 – RCS Flow - Low (one loop) (two loops) – Manual Trip	No interlocks. ~5 second time delay. Trips RCP on affected bus. One loop interlocked with P-8. Two loops interlocked with P-7.
Undervoltage Bus A01, A02	FU11	1 out of 2 per bus	– Loss of RCS Flow – Loss of All AC Power to the Auxiliaries	– Overtemperature ΔT – Pressurizer Pressure - High – RCP Breaker position – RCS Flow - Low (one loop) (two loops) – Manual Trip	Automatic block below P-7

Table E1-3(1) - Information to Support RPS Instrumentation Redundancy and Diversity

RPS Function	Functional Unit (FU)	Channels to Trip (1)(2)(3)	Primary Transient / Accident Protection (4)	Diverse Actuation Functions() (1)	Comments
Underfrequency Bus A01, A02	FU12	1 out of 1 per bus	– Loss of RCS Flow – Loss of All AC Power to the Auxiliaries	– Open RCP Breaker position – Manual Trip	No interlocks. Trips RCP on affected bus.
SG Water Level - Low, Low	FU13	2 out of 3	– Loss of Normal Feedwater – Loss of External Electrical Load – Loss of All AC Power to the Auxiliaries	– Steam/Feed flow mismatch coincident w/ low SG level – Overtemperature ΔT – Overpressure ΔT – Pressurizer Pressure - High – Pressurizer Level - High – RCS Flow - Low – Undervoltage Bus A01, A02 – Underfrequency Bus A01, A02 – Turbine Trip – Manual Trip	Requires 2 of 3 low-low level channels in either SG
SG Water Level Low; coincident w/ Steam Flow / Feed Flow Mismatch	FU14	1 out of 2 per loop*	– Loss of Normal Feedwater – Loss of All AC Power to the Auxiliaries	– SG Water Level - Low, Low – Overtemperature ΔT – Overpressure ΔT – Pressurizer Pressure - High – Pressurizer Level - High – RCS Flow - Low – Undervoltage Bus A01, A02 – Underfrequency Bus A01, A02 – Turbine Trip – Manual Trip	*Coincident w/1 out of 2 low SG water level per loop
Turbine trip on low oil pressure	FU15a	2 out of 3	– Loss of Normal Feedwater – Loss of External Electrical Load	– SG Water Level - Low, Low – Pressurizer Pressure - High – Overtemperature ΔT – Overpower ΔT – Undervoltage Bus A01, A02 – Underfrequency Bus A01, A02 – Manual Trip	Interlocked with P-7 and P-9

Table E1-3(1) - Information to Support RPS Instrumentation Redundancy and Diversity

RPS Function	Functional Unit (FU)	Channels to Trip (1)(2)(3)	Primary Transient / Accident Protection (4)	Diverse Actuation Functions() (1)	Comments
Turbine trip on stop valve closure	FU15b	2 out of 2	− Loss of Normal Feedwater − Loss of External Electrical Load	− SG Water Level - Low, Low − Pressurizer Pressure - High − Overtemperature ΔT − Overpower ΔT − Undervoltage Bus A01, A02 − Underfrequency Bus A01, A02 − Manual Trip	Interlocked with P-7 and P-9

Notes:
1. Each Functional Unit (FU) will cause a reactor trip with 1/2, 2/3 or 2/4 tripped signals.
2. Bypassed channels reduce the number of total available channels by 1, e.g. from 2/4 to 2/3, or from 2/3 to 2/2.
3. An inoperable channel may be placed in a tripped state, reducing the redundancy from 2/4 required tripped channels to 1/3 required tripped channels.
4. Each listed accident results in a reactor trip.

4.2 TS 3.3.2 - Engineering Safeguards Features Actuation System (ESFAS) Instrumentation

The Point Beach ESFAS is comprised of sensors connected to signal processing circuitry consisting of two to four redundant channels that monitor various plant parameters and digital circuitry consisting of two redundant logic trains which receive inputs from the signal processing channels. The two ESFAS actuation trains are provided to actuate the two ESF equipment trains associated with each unit. When coincidence logic for a particular ESFAS subsystem is satisfied, the trip relays in both logic trains are actuated. With redundant logic trains, a single logic train failure will not prevent a valid ESF actuation.

The analog portion of the ESFAS system is shared with the RPS. The four ESFAS channels, which share cabinets with the RPS channels, receive 120 VAC power from the four independent, battery-backed instrument buses. The logic racks for the two ESFAS trains that actuate ESF equipment receive battery-backed power from redundant 125 VDC sources. Loss of AC power to an individual ESFAS channel (except the containment spray actuation channels) will cause the associated channel's output bistables to trip. The “deenergize-to-operate” design is similar to the RPS analog channels. The analog channels are designed to fail in the trip state on a power failure with the exception of the containment spray actuation channels to preclude inadvertent spray-down of the containment on power loss. Similarly, the ESFAS output relays to individual ESF components are intentionally designed as energize-to-trip relays to avoid inadvertent actuation of ESF systems, which would disrupt plant operation.

Four steam line flow channels are used to initiate the Steam Line Isolation (SLI) logic. The four channels are divided into two pairs, with one pair assigned to each main steam line (generator). The flow signals from each main steam line are combined in a 1-out-of-2 coincidence.

Two channels of 4160V bus undervoltage relays are used to initiate the AFW pumps. Each channel contains two relays, which are combined in 1-out-of-2 coincidence. The 1-out-of-2 coincidence outputs are then combined in a taken-twice logic The coincidence logic satisfies the single failure criterion, whereby one relay will not cause unnecessary actuation, and the taken-twice logic requires a loss of both 4160V buses before AFW actuation will occur.

Three channels are used for all other ESFAS trip variables. Three sensor channels support 2-out-of-3 coincidence trip logic, which satisfies both single failure and the reliability criterion. The 2-out-of-3 logic also allows a channel in test to be tripped, while the remaining two channels provide protection in a 1-out-of-2 logic until the tested channel is restored to service

The Point Beach ESFAS is designed to IEEE Standard 279-1968 (Reference 1) with the exception of some engineered safety functions which contain equipment that are not credited in accident analyses and may not fully conform to all IEEE 279 criteria, as described in the Point Beach UFSAR.

ESFAS diversity and redundancy in trip actuation capability is depicted in Table E1-3(2) below:

Reference:

1. NRC IEEE 279, Proposed IEEE Criteria for Nuclear Power Plant Protection Systems, dated August 1968.

Table E1-3(2) - Information to Support ESFAS Instrumentation Redundancy and Diversity

ESFAS Function	Functional Unit (FU)	Channels to Trip	ESF Actuation	Primary Design Basis Accident	Diverse Actuation Function(s)	Comments
Containment Pressure - High	FU1c	2-out-of-3 (1)	Safety Injection	– Large Break LOCA – Small Break LOCA	– Pressurizer Pressure - Low SI	
Pressurizer Pressure - Low	FU1d	2-out-of-3	Safety Injection	– Large Break LOCA – Small Break LOCA – SG Tube Rupture	– Containment Pressure - High SI – SG Water Level - High, High	
SG Pressure - Low, Low	FU1e	2-out-of-3	Safety Injection	– Steam Line Break	– Pressurizer Pressure - Low SI	2-out-of-3 in either loop
Containment Pressure - High, High	FU4c	2-out-of-3	Steam Line Isolation	– Small Break LOCA – Large Break LOCA – Steam Line Break	– Pressurizer Pressure - Low SI – Containment Pressure - High SI – Overpower ΔT	
SG Flow - High, coincident with SI and Low T_{avg}	FU4d	1-out-of-2 taken twice	Steam Line Isolation	– Steam Line Break	– Pressurizer Pressure - Low SI – Containment Pressure - High SI – Overpower ΔT	1-out-of-2 in either loop and 2-out-of-4 low T_{avg} channels. Four steam flow channels are divided into two pairs, with one pair assigned to each steam loop. Flow signals from each loop are combined in a 1-out-of-2 coincidence logic.
SG Flow - High, High, coincident with SI	FU4e	1-out-of-2	Steam Line Isolation	– Steam Line Break	– Pressurizer Pressure - Low SI – Containment Pressure - High SI – Overpower ΔT	1-out-of-2 in either loop
SG Water Level - High, High	FU5b	2-out-of-3	Feedwater Isolation	– Excessive Load Increase – Reduction in feedwater enthalpy	– SI on Pressurizer Pressure - Low – SG Flow - High, High, coincident with SI – SG Flow - High, coincident with SI and Low T_{avg}	Provides backup (non-credited) feedwater isolation 2-out-of-3 in either loop

Table E1-3(2) - Information to Support ESFAS Instrumentation Redundancy and Diversity

ESFAS Function	Functional Unit (FU)	Channels to Trip	ESF Actuation	Primary Design Basis Accident	Diverse Actuation Function(s)	Comments
SG Water Level - Low, Low	FU6b	2-out-of-3	AFW Start	– Loss of Normal Feedwater – Loss of all AC Auxiliaries	– Undervoltage Bus A01 and A02 – AMSAC	2-out-of-3 in either loop. AMSAC inputs are separate and independent of RPS.
Undervoltage Bus A01 and A02	FU6d	1 of 2 taken twice	AFW Start	– Loss of Normal Feedwater	– SG Water Level - Low, Low – AMSAC	Starts Turbine Driven AFW. Two channels of 4160V bus undervoltage relays are used to initiate AFW. Each channel contains two relays, which are combined in 1-out-of-2 coincidence.

Notes:
1. 2-out-of-3 logic also allows a channel in test to be tripped, while the remaining two channels provide protection in a 1-out-of-2 logic until the tested channel is restored to service

ENCLOSURE 2

Point Beach Units 1 and 2
License Amendment Request to Revise Technical Specifications
to Adopt Risk Informed Completion Times TSTF-505, Revision 2,
“Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b”

Information Supporting PRA Consistency with RG 1.200

Table of Contents

1.0 <u>Purpose</u>

The purpose of this enclosure is to provide information on the technical adequacy of the Point Beach Nuclear Plant (PBN) Probabilistic Risk Assessment (PRA) internal events (IE), internal flooding, and fire PRA models in support of the license amendment request (LAR) to adopt TSTF-505, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”, Revision 2 (Reference 1). The PBN internal events, internal flooding, and fire PRA models described within this LAR are based on those described within NextEra Energy (NEE) PBN submittals regarding adoption of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (Reference 2) with routine maintenance and updates applied.

Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, Revision 0 (Reference 3), as clarified by the NRC final safety evaluation of this report (Reference 4), defines the technical attributes of a PRA model and its associated Configuration Risk Management Program (CRMP) tool required to implement this risk-informed application. Meeting these requirements satisfies Regulatory Guide (RG) 1.174, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2 (Reference 5), requirements for risk-informed plant-specific changes to a plant's licensing basis.

NEE uses a multi-faceted approach to establishing and maintaining the technical adequacy and fidelity of PRA models for its nuclear generation sites. This approach includes both a PRA maintenance and update process procedure and the use of self-assessments and independent peer reviews.
Section 2.0 of this enclosure describes the overall approach used to perform the peer review finding closure reviews for the PBN PRAs. Section 3.0 discusses the requirements related to the scope of the PBN PRA internal events, internal flood, and internal fire models. Section 4.0 addresses the technical adequacy of the PBN PRA full power internal events and internal flood models for this application. Section 5.0 addresses the technical adequacy of the PBN Fire PRA model for this application.

No portable FLEX mitigating strategies are incorporated into the Point Beach PRA models used in this LAR.

2.0 <u>Peer Review Finding Closure Process</u>

All the PRA models described below have been peer reviewed to the requirements of RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2 (Reference 6), the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) RA-Sa-2009 PRA Standard (hereafter “ASME/ANS PRA Standard”), “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Reference 7), NEI 05-04, “Process for Performing PRA Peer Reviews Using the ASME PRA Standard (Internal Events)”, Revision 2 (Reference 8),and NEI 07-12, “Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines”, Revision 1 (Reference 9).

The review and closure of all but one of the finding-level Facts and Observations (F&Os) (LE-C9-01) from the peer reviews have been independently evaluated to confirm that the associated model changes did not constitute a model upgrade. These reviews included F&Os that were associated with “met” supporting requirements, as well as all F&Os associated with supporting requirements (SRs) that were met at the Capability Category (CC) II level. LE-C9-01 is met at CCI. The finding closures are associated with the current internal events, internal flood, and internal fire models. As an implementation item, these models will be implemented as a one top MOR for use in implementing the RICT process.

Expectations regarding preparation for the review (NEI 05-04, Section 4.2) and conduct of the self-assessment by the host utility (NEI 05-04, Section 4.3) were addressed prior to conduct of these reviews. This included documentation by PBN of resolution of the prior PRA peer review finding-level F&Os and preparation of the information required for this independent assessment. The documented bases for F&O closure provided by NEE included a written assessment whether the resolution constituted PRA maintenance or PRA upgrade.

The multi-disciplinary teams of reviewers for each closure review met the independence and relevant peer reviewer qualifications requirements in the ASME/ANS PRA Standard and related guidance. The internal events, internal flood, and fire F&Os were assessed, each of which was assigned to at least two of the reviewers.

(Reference 10), (Reference 11), (Reference 12), (Reference 13), and (Reference 14) provide additional details of the F&O closure reviews, including the approach taken:

- The process guidance in NEI 05-04, Section 4.6, was applicable to these reviews.
- The independent technical review teams reviewed the documented bases for closure of the finding-level F&Os prepared by NEE.
- The independent technical review teams determined whether the finding-level F&Os in question had been adequately addressed and could be closed out by consensus.
- As part of this process each F&O was reviewed regarding whether the closure response represented PRA maintenance or a PRA upgrade.
- Section 3 of each F&O closure report specifically states that the closure review team concluded that all SRs where the F&Os have been closed are now “met” at CC II.
- Details of the F&O Closure review assessments are documented in Appendix A of the F&O Closure Reports. The assessment for each F&O includes the determination that each closed finding meets CC II for all the applicable SRs of the ASME/ANS PRA Standard, as endorsed by RG 1.200 Revision 2.

3.0 Requirements Related to Scope of PBN Internal Events, Internal Flood and Fire PRA Models

The internal events, internal flood, and the internal fire PRA MOR are at power models. The models include both Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). As described previously, the internal events and internal flood PRA models described within this LAR are based on those described within the NEE submittal of the LAR to adopt 10 CFR 50.69 (Reference 2) with routine maintenance and updates applied.

4.0 Scope and Technical Adequacy of the PBN Internal Events and Internal Flood PRA Models

NEI 06-09-A requires that the PRA be reviewed to the guidance of RG 1.200, Revision 2 for a PRA which meets CC II for the supporting requirements of the internal events at power ASME/ANS PRA Standard. It also requires that deviations from these CCs relative to the Risk Informed Completion Time (RICT) Program be justified and documented.

The information provided in this section demonstrates that the PBN internal events PRA model (including internal flooding) meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2.

The PBN Internal Events and Internal Flood PRAs were peer reviewed in November 2010 (Full scope), August 2011 (IF Focused Scope), and October 2011 (IE Focused Scope) applying NEI 05-04, the ASME/ANS PRA Standard and RG 1.200, Revision 2. The purpose of these reviews was to provide a method for establishing the technical adequacy of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used.
An F&O closure review was conducted in 2017/2018 in accordance with the process documented in Appendix X to NEI 05-04/07-12/12-06 (Reference 11) and (Reference 12), as well as the requirements published in the ASME/ANS PRA Standard and RG 1.200, Revision 2. For each closed F&O, the F&O resolution was assessed to determine if the PRA met the Capability Category II requirements of the ASME PRA Standard's SRs that were referenced in the F&O. A specific evaluation was also provided for each closed F&O to document whether the review team considered the F&O resolution a "PRA update" or a "PRA upgrade".

Following the external closure review (Reference 12), six (6) internal event and two (2) internal flood findings remained open. A follow-on independent assessment for finding closure was conducted by the Pressurized Water Reactor Owner's Group (PWROG) in February/March 2022 (Reference 10)

Table E2-1 discusses the disposition of the one remaining open finding after this latest finding closure review. With the disposition of this one open peer review finding, the one-top MOR to be issued for implementation of the 4b program will meet the requirements for PRA technical adequacy for this application.

5.0 Scope and Technical Adequacy of the PBN Fire PRA Model

NEI 06-09-A requires that the PRA be reviewed to the guidance of RG 1.200, Revision 2 for a PRA which meets CC II for the supporting requirements of the internal fire at power ASME/ANS PRA Standard. It also requires that deviations from these CCs relative to the Risk Informed Completion Time (RICT) Program be justified and documented.

The information provided in this section demonstrates that the PBN internal fire PRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2.

The PBN Internal Fire PRA was peer reviewed in June 2011 (full scope), May 2013 (FSS Focused Scope), and June 2013 (FQ Focused Scope) applying NEI 07-12, the ASME/ANS PRA Standard and RG 1.200, Revision 2. The purpose of these reviews was to provide a method for establishing the technical adequacy of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used.

An F&O closure review was conducted in 2017/2018 in accordance with the process documented in Appendix X to NEI 05-04/07-12/12-06 (Reference 13) and (Reference 14), as well as the requirements published in the ASME/ANS PRA Standard and RG 1.200, Revision 2. For each closed F&O, the F&O resolution was assessed to determine if the PRA met the Capability Category II requirements of the ASME PRA Standard's SRs that were referenced in the F&O. A specific evaluation was also provided for each closed F&O to document whether the review team considered the F&O resolution a "PRA update" or a "PRA upgrade".

Following the external closure review (Reference 14), sixteen (16) internal fire findings remained open. A follow-on independent assessment for finding closure was conducted by the PWROG in February/March 2022 (Reference 10). All internal fire findings are assessed as closed. The one-top MOR to be issued for implementation of the 4b program will meet the requirements for PRA technical adequacy for this application.

6.0 References

Reference 1. "Letter from the Technical Specification Task Force (TSTF) to the NRC, "TSTF Comments on Draft Safety Evaluation for Traveler TSTF-505, 'Provide Risk-Informed Extended Completion Times' and Submittal of TSTF-505, Revision 2"." *ML18183A493.* July 2, 2018.

Reference 2. "NEE License Amendment Request 287, "Application to adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants'"." *ML17243A201.* August 31, 2017.

Reference 3. "NEI 06-06-A Risk-Informed Technical Specifications Initiative 4b." *ML12286A322.* October 2012.

Reference 4. "Letter from the NRC to NEI, "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines." *ML071200238.* May 17, 2007.

Reference 5. "NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." *ML100910006.* Revision 2, May 2011.

Reference 6. "NRC Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." *ML090410014.* Revision 2, March 2009.

Reference 7. "ASME Standard ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008, Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." February 2009.

Reference 8. "NEI Topical Report NEI 05-04, Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard." *ML083430462.* November 2008.

Reference 9. "NEI Topical Report NEI 07-12, Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines." *ML102230070.* June 2010.

Reference 10. "PWROG Independent Review of PBN Internal Events, Internal Flood, and Internal Fire PRA Peer Review Finding Resolutions Point Beach Units 1 and 2." *PBN-BFJR-22-015.*

Reference 11. " Independent Review of PBN Internal Events and Internal Flood PRA Peer Review Finding Resolutions Point Beach Units 1 and 2." *PBN-BFJR-17-041.* August 2017 [historical].

Reference 12. "PB Units 1 and 2 IE PRA Peer Review Findings Closure Review Update." *PBN-BFJR-18-055, Rev. 1.* December 2021.

Reference 13. "Point Beach Units 1 & 2 Fire Probabilistic Risk Assessment Peer Review Findings Closure." *PBN-BFJR-17-054.* October 2017 [historical].

Reference 14. "PB Units 1&2 Fire PRA Peer Review Findings Closure Review Update." *PBN-BFJR-18-056.* September 2018.

Table E2-1 – Point Beach PRA Internal Events Peer Review Findings

SR	Category and Finding	Other Affected SRs	PEER REVIEW FINDINGS	RESOLUTION	IMPACT ON APPLICATION
LE-C9	**CC-I** **Finding LE-C9-01**	**LE-C9 (CC-I) LE-C10 (CC-I) LE-C11 (CC-I) LE-C12 (CC-I) LE-D3 (CC-I)**	**2010 Peer Review Suggestion:** Point Beach uses a unit-specific NUREG/CR-6595 CAFTA one-top model covering both CDF and LERF. The model should be expanded to address the dual unit impacts but no additional level of detail would be required. The NRC has indicated that for most applications, they are only interested in LERF. Furthermore, they have indicated that the use of a simplified NUREG/CR-6595 LERF model was acceptable as long as it addressed plant-specific differences from the model in NUREG/CR-6595. However, for any application that directly addresses containment performance or the actual source terms and timing, a more detailed analysis approximating a Level 2 PRA would likely be required	**2010 Peer Review Plant Response:** The Peer Review was of the Level 1 CDF and LERF Model. The issue raised suggests that A Level 2 PRA would likely be required to evaluate some issues of containment performance. This is outside the bounds of the areas to be evaluated in a Level 1 Peer Review. Additionally, it should be noted that Point Beach has a complete Level 2 PRA model, which is in the process of being updated, that is available to be used for any evaluation of containment performance for which the Level 1 LERF model is deemed insufficient. **ALL issues identified in the 2010 Peer Review Findings were resolved in the PRA Model.**	**NO IMPACT** **Finding Resolved for CCI- No open issues from the 2010 or 2011 Peer Review**
			2011 Peer Review Finding: Original suggestion is limited in scope and since CC-II is not obtained for SR-C9 through C12, this should have been a finding. Above response does not address CC-II requirements (e.g., need to explain that containment analysis went beyond NUREG and is CC-II and add it to the notebook if this is true). Also, the present LERF Notebook is not a complete Level 2 model. If it exists it is not available.	**2011 Peer Review Plant Response:** The LERF Analysis does meet CC-I requirements. Use of the PRA which meets CC-1 requirements is conservative. **ALL issues identified in the 2011 Peer Review Findings were resolved in the PRA Model.**	

ENCLOSURE 3

Point Beach Units 1 and 2
License Amendment Request to Revise Technical Specifications
to Adopt Risk Informed Completion Times TSTF-505, Revision 2,
“Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b”

Information Supporting Technical Adequacy of PRA Models Without PRA Standards Endorsed by Regulatory Guide 1.200, Revision 2

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1.0 <u>Purpose</u>

This enclosure is not applicable to the Point Beach Nuclear Plant submittal. NextEra Energy Point Beach, LLC is not proposing to use any PRA models in its Risk-Informed Completion Time Program for which a PRA standard, endorsed by the NRC in RG 1.200, Revision 2 does not exist.