NEXTera ENERGY
POINT BEACH

September 28, 2010

NRC 2010-0143
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
Extended Power Uprate
Response to Request for Additional Information

References:
(1) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)
(2) NRC electronic mail to NextEra Energy Point Beach, LLC dated September 8, 2010, Request for Additional Information (SRXB) re: EPU License Amendment Request (TAC Nos. ME1044 and ME1045) (ML102580398)

NextEra Energy Point Beach, LLC (NextEra) submitted License Amendment Request (LAR) 261 (Reference 1) to the NRC pursuant to 10 CFR 50.90. The proposed amendment would increase each unit's licensed thermal power level from 1540 megawatts thermal (MWt) to 1800 MWt, and revise the Technical Specifications to support operation at the increased thermal power level.

Via Reference (2), the NRC staff determined that additional information is required to enable the staff's continued review of the request. Enclosure 1 provides the NextEra response to the NRC staff's request for additional information.

This letter contains no new Regulatory Commitments and no revisions to existing Regulatory Commitments.

The information contained in this letter does not alter the no significant hazards consideration contained in Reference (1) and continues to satisfy the criteria of 10 CFR 51.22 for categorical exclusion from the requirements of an environmental assessment.

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 September 28, 2010.

Very truly yours,

NextEra Energy Point Beach, LLC

for

Larry Meyer
Site Vice President

Enclosure

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
EXTENDED POWER UPRATE
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The NRC staff determined that additional information was required (Reference 1) to enable the Reactor Systems Branch to complete the review of License Amendment Request (LAR) 261, Extended Power Uprate (EPU) (Reference 2). The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's request.

SRXB-1

Please demonstrate that the margin-to-overfill consequences of a postulated steam generator tube rupture (SGTR) are no greater at EPU power level than at the currently licensed thermal power level.

NextEra Response

The most significant impact of an EPU on the plant response to a steam generator tube rupture (SGTR) is the decay heat that must be removed to cool the reactor coolant system and to provide subcooling margin prior to depressurization, as a necessary step to terminate break flow. The higher decay heat levels associated with the EPU will result in a longer cooldown duration and accumulation of additional break flow in the secondary side of the ruptured steam generator (SG). However, this is offset by the lower initial secondary mass associated with operation at the higher power level.

The analysis of the margin to overfill transient includes a series of sensitivity runs to determine the limiting initial condition assumptions related to the vessel average temperature and SG tube plugging. The analysis also includes additional sensitivity cases to investigate the competing effects on the margin to overfill of the assumed decay heat level, auxiliary feedwater temperature and safety injection temperature. While higher values for these analyses result in an extended cooldown period and additional break flow accumulation in the ruptured SG, they also result in higher steam release from the ruptured SG which reduces the accumulated mass.

Transient calculations were performed in a manner consistent with the analyses performed at EPU conditions (see LR Section 2.8.5.6.2.2.5 of Reference 2) with initial conditions consistent with current licensed thermal power level. The limiting case for current licensed thermal power level showed less margin to overfill than the limiting case for the EPU. Therefore, NextEra concludes that the margin to overfill results of a postulated SGTR are no greater at EPU power level than at the currently licensed thermal power level.

SRXB-2

The margin-to-overfill SGTR analysis does not appropriately characterize limiting plant initial conditions or uncertainties. In light of the narrow margin to overfill without acceptably characterized uncertainty, please provide the information referenced in Requirement (3) of the NRC staff SER approving WCAP-10698 regarding the main steam lines and associated supports under water-filled conditions.

NextEra Response

The main steam lines and associated supports have been analyzed to remain intact under water filled conditions without having to pin the piping spring supports.

SRXB-3

Two "better-estimate" thermal-hydraulic analyses were described; one evaluated margin to overfill with a minimum RCS temperature and 10-percent tube plugging, while the other supplemented the dose analysis with a maximum RCS temperature and 0-percent tube plugging.

Please provide a summary statement comparing the purposes of these two analyses, and explain how the different initial condition assumptions achieved each intended purpose.

NextEra Response

Two supplemental analyses were performed. Both model the plant and operator responses to the postulated SGTR. The first analysis includes modeling to maximize the accumulation of water in the secondary side of the ruptured steam generator (SG) and is used to demonstrate margin to overfill. The second analysis includes modeling to maximize the release of steam from the ruptured SG and is used to demonstrate that the input to the SGTR dose analysis is conservative. The major differences in the analysis input are summarized in Table SRXB-3-1 below:

Table SRXB-3-1: Comparison of Margin to Overfill and Input to Dose Modeling

	Margin to Overfill	**Input to Dose**
SG Model	Unit 1 Model 44F	Unit 2 Model Δ47
RCS Tavg	558.0°F	577.0°F
Tube Plugging	10%	0%
Feedwater Temperature	390°F	458°F
AFW Flow Rate	400 gpm/SG	137.5 gpm/SG
AFW Initiation Delay	1 second	300 seconds
AFW Enthalpy	0.0274 Btu/lbm	70.9 Btu/lbm
Decay Heat Model	Nominal 1971 ANS decay heat	1971 ANS decay heat +20%

The Unit 1 Model 44F steam generators and the low feedwater temperature are selected for the margin to overfill analysis, since these result in a higher initial secondary side inventory. Low Tavg with maximum tube plugging was determined to result in lower margin to overfill for the EPU. Maximum auxiliary feedwater (AFW) flow with minimum initiation delay maximizes the accumulation of water in the secondary side of the ruptured SG. Minimum AFW enthalpy and nominal decay heat were determined to result in lower margin to overfill for the EPU as noted in the response to SRXB-1.

The Unit 2 Model Δ47 steam generators and the high feedwater temperature are selected for the input to dose analysis, since these result in a lower initial secondary side inventory. High Tavg, with minimum tube plugging, results in higher releases from the ruptured SG. Minimum AFW flow with maximum initiation delay and maximum enthalpy minimizes the absorption of energy by the AFW flow and maximizes the steam released. High decay heat maximizes the releases.

SRXB-4

In light of the extensive evaluations of post-30 minute steam generator tube rupture consequences, it is not clear that the 30 minute break flow termination time is reasonable.

Please provide additional information to confirm that this assumption is based on observed operator capability, or revise the mass release calculations to incorporate a more realistic break flow termination time. Provide the plant-specific information identified in Requirement (1) of the NRC staff SER approving WCAP-10698 regarding assurance that the necessary actions and times can be taken consistent with those assumed in the SGTR analyses.

NextEra Response

Consistent with plants of the same vintage, the current SGTR licensing basis consists of a simplified thermal-hydraulic analysis to determine the mass of primary-to-secondary break flow and the mass of steam released to atmosphere for input to a radiological consequences analysis. This simplified thermal-hydraulic analysis assumes that primary-to-secondary break flow continues for 30 minutes following the start of the event and includes conservative assumptions to provide appropriate radiological dose consequences. This methodology was used for the Alternative Source Term radiological consequences analysis for SGTR at EPU conditions.

A supplemental SGTR dose analysis was then performed using selective implementation of the modeling provided in WCAP-10698-P-A methodology to model operator responses leading to termination of break flow to the ruptured steam generator consistent with Emergency Operating Procedure EOP-3, Steam Generator Tube Rupture. The details of this supplemental analysis are documented in the NextEra Response to Question 3 of Reference 3. The operator action times used for the supplemental SGTR dose analysis were discussed in the NextEra Response to RAI IHPB HF-2 in Enclosure 1 of Reference 4.

The results of the supplemental dose analysis show that even with termination of break flow at 53 minutes following the start of the event, the radiological consequences from the supplemental SGTR dose analysis were much lower than those calculated using the SGTR licensing basis methodology, which assumes break flow termination in 30 minutes.

Plant operating personnel are periodically trained and tested on SGTR scenarios. A time critical action for verifying operator response to a SGTR requires that the steam releases from the ruptured SG to the environment must be terminated within the first 30 minutes of the event with no re-initiation of the release past the 30-minute point. Plant operating personnel have been evaluated to confirm they can meet this time requirement.

Therefore, based on the supplemental dose analysis and the operator simulator training requirements discussed above, the 30-minute break flow termination is reasonable for the SGTR radiological consequences analysis.

SRXB-5

For the SGTR analyses, provide a list of systems, components and instruments that are credited for accident mitigation in the plant specific SGTR EOPs. Specify whether each component is safety grade, consistent with Requirement (4) of the NRC staff SER approving WCAP-10698.

NextEra Response

EOP-3, Steam Generator Tube Rupture identifies the following systems, components, and instruments for accident mitigation. Note that the EOP identifies multiple means and equipment available to the operators to perform required mitigation functions. Therefore, not all of the equipment in Table SRXB-5-1 below is required for any postulated SGTR event.

Table SRXB-5–1: Systems, Components, and Instruments Available for SGTR Mitigation

System, Component, or Instrument ID Number	System, Component, or Instrument Description	Safety Related
1&2MS-2017 & 2018	Main Steam Isolation Valves (MSIVs)	Yes
1&2MS-234 & 236	MSIV Bypass Valves	Yes
1&2MS-2015 & 2016	SG Atmospheric Steam Dump Valves	Yes
1&2CS-466 & 476	Main Feedwater Regulating Valves	Yes
1&2CS480 & 481	Feedwater Regulating Valve Bypass Valves	Yes
1&2P-28A & B	Main Feedwater Pumps and Discharge MOVs	No
P-38A & B	Motor-Driven AFW Pumps	Yes
1&2MS-2019 & 2020	Steam Supply to Turbine-Driven AFW Pump	Yes
AF-4021 & 4023 and 1&2AF-4000 & 4001	AFW Supply Line Isolation Valves	Yes
1&2FT-4036 & 4037	AFW Flow to Steam Generators A and B	Yes
1&2MS-2042, 2045, 5958 & 5959	SG Blowdown Isolation Valves	Yes
1&2MS-2050 to 2057	Condenser Steam Dump Valves	No
K-2A & B	Instrument Air Compressors	No
1&2IA-3047 & 3048	Instrument Air Containment Isolation Valves	Yes
1&2C-111, 113 & 116	Safety Injection and Containment Isolation Bistables & Associated Actuation and Reset Circuitry	Yes
1&2P-15A & B	Safety Injection Pumps	Yes
1&2LT-426, 427 & 428	Pressurizer Water Level	Yes
1&2PT-420A, B, & C	Reactor Coolant System Wide-Range Pressure	Yes
1&2PT-468, 469, & 482, 478, 479 & 483	SG Pressure Transmitters	Yes
1&2LT-461, 462, & 463, 471, 472 & 473	SG A and B Narrow Range Level Transmitters	Yes
1 &TE-450A-D & 451A-D	Reactor Coolant Hot and Cold Leg Temperature	Yes*
1&2TM-970 & 971	Reactor Coolant System Subcooling Monitors	Yes
1&2TE-1 to 39	Core Exit Thermocouples	Yes
1&2RC-430 & 431C	Pressurizer Power-Operated Relief Valves	Yes
1&2RC-431A & B	Pressurizer Spray Valves	Yes
1&2CV-296	Pressurizer Auxiliary Spray Valves	Yes
1&2P-2A, B & C	CVCS Charging Pumps	Yes

System, Component, or Instrument ID Number	System, Component, or Instrument Description	Safety Related
1&2RE-215 and RE-225	Condenser Air Ejector Radiation Monitors	No**
1&2RE-219 & 222	SG Blowdown Radiation Monitors	No**
1&2RE-231 & 232	Main Steam Line Radiation Monitors	No**

* Temperature Elements (TEs) A, B, and C are Augmented Quality; TEs D are safety-related.
** Radiation Monitors are Augmented Quality.

The following additional information is provided regarding plant-specific Requirement (4) of the NRC staff SER approving WCAP-10698:

1. For the pressurizer PORVs, the motive power to open these safety-related, air-operated valves to reduce RCS pressure is Instrument Air (IA), which is reliable but not safety-related. Redundant IA compressors and backup service air compressors are powered from diesel-backed safeguards electrical buses. Providing IA to containment where the PORVs are located requires resetting safety injection (SI) and containment isolation (CI) signals and opening the air-operated IA containment isolation valves, which are safety-related components. Backup means of reducing RCS pressure include: Use of safety-related normal pressurizer spray valves, which have nitrogen tanks to back up the normal IA motive power, but requires at least one reactor coolant pump (RCP) to be operating, or Use of safety-related pressurizer auxiliary spray valves, that can be opened by differential pressure from the charging pumps. The charging pumps are reliable and are powered from diesel-backed safeguards electrical buses.

2. For the safety-related atmospheric steam dump valves (ADVs), the motive power to close for isolation of the ruptured SG is safety-related control power to isolate and vent IA to the valve operator. The backup means to close or isolate these valves on the ruptured SG is local manual operator action to close a manual isolation valve. For opening the ADV on the intact SG for cooldown of the RCS, the motive power is safety-related control power to control IA to the air operator. The backup means to control the ADVs is local operator action with manual handwheels (see response to SRXB-6 for evaluation of a single failure during a SGTR event).

3. A list of radiation monitors is provided in Table SXRB-5-1 above. The radiation monitoring system is augmented quality. Although SG sampling may be performed to confirm the identification of the ruptured SG during a SGTR event, the installed augmented quality radiation monitors and the safety-related SG level transmitters are used in the EOPs to identify the ruptured SG. Therefore, the time duration for sampling and analysis of SG secondary water would not delay the response to this event.

SRXB-6

LR Section 2.8.5.6.2.2.6 evaluates a more realistic SGTR event to confirm that the 30-minute, licensing basis mass release results provide conservative (i.e., acceptably high) inputs to the dose calculations. It appears that this more realistic event shows significant margin to the licensing basis results. The more realistic analysis, however, did not consider (1) uncertainties, (2) a single failure assumption, and (3) conservatism on secondary mass.

Please characterize the impact that these analytic features would have on the predicted results and confirm that, in consideration of these conservatisms, the supplemental thermal-hydraulic analysis would still indicate that the 30-minute licensing basis analysis is still bounding.

NextEra Response

The purpose of the mass release calculations is to provide input to the radiological consequences analysis. The flashed break flow has the greatest impact on the SGTR radiological consequences analysis since it is modeled as a direct release from the reactor coolant system to the environment with no holdup, dilution, or partitioning in the secondary side of the ruptured steam generator (see Licensing Report (LR) Section 2.8.5.6.2.2.6 of Reference 2).

A conservatism included in the analysis presented in LR Section 2.8.5.6.2.2.6 is that the break flow flashing fraction shown in LR Figure 2.8.5.6.2-7 was determined using the hot leg temperature. Since the tube rupture flow calculated with the LOFTTR2 code consists of flow from the hot leg and cold leg sides of the SG, the actual break flow temperature and the flashing fraction is much lower. With the break modeled at the top of the tube sheet, approximately 75% of the break flow comes from the tube sheet side of the break, while the flow from the other side of the broken tube accounts for 25% of the total due to the modeling of the losses associated with the length of the tube. Shortly after a reactor trip, the enthalpy of the break flow from the cold leg side of the break would be below the saturation enthalpy at the ruptured SG pressure and that flow would not flash. Despite this conservative modeling, the radiological results show that the 30-minute licensing basis analysis is bounding, with considerable dose margin between the two calculations as presented in Table 3 of Enclosure 1 of Reference 3.

Consideration of uncertainties would have a much smaller impact on the flashed break flow and resulting doses. Following reactor trip and the assumed loss of offsite power the reactor coolant system temperatures trend towards the no-load temperature, independent of the initial conditions assumed. The initial secondary mass mainly impacts the steam releases. Steam releases from the ruptured SG are much lower in the realistic analysis than the 30-minute licensing basis analysis. The 30-minute licensing basis analysis assumes the ruptured SG participates equally in removing the decay heat in the period from reactor trip until break flow termination. The realistic analysis utilizes the intact SG for the cooldown, consistent with the plant emergency operating procedures. Adding conservatism to the initial SG mass would not change the conclusion that the 30-minute analysis is bounding. As noted in the response to SRXB-3 above, the decay heat model used in the calculation includes uncertainties to maximize the releases and the time required to cool the reactor coolant system.

As noted in LR Section 2.8.5.6.2.2.6 of Reference 2, exclusion of a single failure assumption for the SGTR event is consistent with the licensing basis for PBNP. Considering a single failure of the power-operated relief valve on the ruptured SG in the full open position similar to that considered in the Ginna SGTR analysis would result in (1) increased steam release from the ruptured SG, (2) increased break flow due to the lower ruptured SG pressure and (3) an increased flashing fraction due to the lower ruptured SG pressure (partially offset by the lower primary side temperature). Sensitivity transient runs and dose analysis calculations modeling this failure resulted in an increase in calculated doses of approximately a factor of 2 compared to the supplemental analysis results presented in Table 3 of Enclosure 1 of Reference 3 for LAR 241 (prior to rounding). Despite this increase, the sensitivity calculations indicate that the 30-minute licensing basis analysis is still bounding.

References

(1) NRC electronic mail to NextEra Energy Point Beach, LLC dated September 8, 2010, Request for Additional Information (SRXB) re: EPU License Amendment Request (TAC Nos. ME1044 and ME1045) (ML102580398)

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

(3) NextEra Energy Point Beach letter to NRC, dated June 1, 2009, Response to Request for Additional Information, License Amendment Request 241, Alternative Source Term (ML091560413)

(4) NextEra Energy Point Beach letter to NRC, dated April 29, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information (ML101190456)