May 14, 2010

NRC 2010-0042
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 letter to NextEra Energy Point Beach, LLC, dated March 31, 2010, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Health Physics Branch RE: Extended Power Uprate (TAC Nos. ME1044 and ME1045) (ML100820459)

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 license 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 was 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 May 14, 2010.

Very truly yours,

NextEra Energy Point Beach, LLC

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 Health Physics 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.

IHPB HP RAI-1:

The first paragraph of item number 5. "Ensuring that Occupational and Public Radiation Exposures are [as low as reasonably achievable] ALARA," (page 2.10.1-4 of Attachment 5) implies that the PBNP ALARA design basis for members of the public is 0.24 rem/year whole body dose, at the site boundary, which "compares with the 10 CFR 20 limit of 0.5 R/y." Describe the radiation sources that result in 0.24 rem/year at the site boundary. Also, the reference to a public dose limit of 0.5 R/y implies that PBNP has a Nuclear Regulatory Commission (NRC) approval, per the provisions in 10 CFR 20.1301(d), to exceed the current public dose limit of 100 mrem in 10 CFR 20.1301(a). Provide a reference for this approval or provide a technical basis for why PBNP cannot meet the current annual dose limit of 100 mrem in 10 CFR 20.1301(a).

NextEra Response

Item 5 of LAR 261, Attachment 5, Section 2.10.1, Ensuring that Occupational and Public Radiation Exposures are ALARA, was intended to summarize the PBNP license basis as stated in Amendment 1 to "effect such changes in its equipment and/or operations as will reduce radiation exposures and releases of radioactive materials to unrestricted areas as far below the limits specified in 10 CFR Part 20 as the state of the art in the reduction of such emissions will allow..." Section 2.10.1 continues on to provide the results of the whole body dose analyses reported in Amendment 1 (0.24 R/yr) and compares it to the 0.5 R/yr dose limit of 10 CFR 20 that was effective at that time (note that at that time, 10 CFR 50, Appendix I was not finalized and the concept of "as low as reasonably achievable" (ALARA) being introduced into federal regulations for effluent releases was still in draft stage).

It was not NextEra's intent to imply that the above values represented dose limits applicable to the PBNP license basis as opposed to that required by the current 10 CFR 20.1301(a). As demonstrated in LAR 261, Attachment 5, Section 2.10, Table 2.10.1-2, PBNP meets the requirements of 10 CFR 20 and 10 CFR 50, Appendix I under EPU conditions. This section of the LAR also identifies the applicable final safety analysis report (FSAR) sections that discuss the implementation of the overall requirements of the 10 CFR 20 and 10 CFR 50, Appendix I. The text directs the reviewer to the PBNP Technical Specification (TS) requirements of the Radioactive Effluent Controls Program and the Offsite Dose Calculation Manual (ODCM), which cite compliance with 10 CFR 20 and 10 CFR 50, Appendix I.

IHPB HP RAI-2

The second paragraph of the "Results" section on page 2.10.1-10 of Attachment 5 concludes that the proposed 17.6 percent increase in reactor power (above the current licensed power) does not result in any change to the plant's radiation zoning. However, the reviewer did not find any plant radiation zoning in the updated final safety analysis report (UFSAR) submitted with the power uprate request. Provide updated FSAR figures indicating the current radiation zoning for each area of the plant.

NextEra Response

Section 11.6.2 of the FSAR, System Design and Operation, states that all plant areas capable of personnel occupancy are classified as one of five radiation zones of radiation levels listed in FSAR Table 11.6-1, Shield Design Zone Classification. The shielding design zone classifications defined in FSAR Table 11.6-1 are as follows:

- Zone 0 – Unlimited occupancy, maximum allowable dose rate of 0.1 mrem/hr;
- Zone I – Normal continuous occupancy, maximum allowable dose rate of 1.0 mrem/hr;
- Zone II – Periodic occupancy, maximum allowable dose rate of 2.5 mrem/hr;
- Zone III – Controlled occupancy, maximum allowable dose rate of 15 mrem/hr; and
- Zone IV – Controlled access, maximum allowable dose rate of >15 mrem/hr.

The FSAR does not include figures providing the current radiation zoning for each area of the plant. However, the following textual description of typical plant areas and assigned zones is provided in FSAR Section 11.6.2:

> "Typical Zone 0 areas are the turbine building and turbine plant service areas. Typical Zone I areas are the offices and control room. Zone II areas include the local control spaces in the auxiliary building, and the operating deck of the containment during reactor shutdown. Areas designated Zone III include the sample room, valve galleries, fuel handling areas, and intermittently occupied work areas. Typical Zone IV areas are the shielded equipment compartments in the Auxiliary Building, waste drum storage area, and the primary loop compartments after shutdown."

Pages 2.10-5 through 2.10-11 of LAR 261, Attachment 5 discuss the impact of EPU on normal operation radiation levels and shielding adequacy. The discussion is provided in two parts (the impact of the EPU on (1) normal operation radiation levels and (2) plant shielding adequacy).

Part (1) notes that PBNP is currently operating at a core power level of 1540 megawatts thermal (MWt) and an 18-month fuel cycle, and discusses the impact of an EPU to an analyzed core power level of 1810.8 MWt on radiation levels in the plant. The EPU assessment concludes that the radiation levels will increase by approximately the percentage of the uprate, i.e., 17.6%. The assessment discusses areas near the reactor vessel, in-containment areas adjacent to the reactor coolant system (RCS), areas near irradiated fuels and other irradiated objects, and areas outside the containment where the radiation source is either primary coolant or downstream sources originating from primary coolant.

Part (2) notes that shielding is used to reduce radiation dose rates in various parts of the station to acceptable levels consistent with operational and maintenance requirements and that PBNP shielding was based on a core power level of 1520 MWt, a 1-year fuel cycle length and a design basis reactor coolant concentration based on 1% fuel defects. The EPU assessment documented in this section is based on an analyzed core power level of 1810.8 MWt, an 18-month fuel cycle and TS that limit the reactor coolant concentration levels. The LAR credits the conservatism in the original design basis source terms used to determine plant shielding (i.e., the gamma and neutron flux data which bound EPU operations, and the original reactor coolant concentrations based on 1% fuel defects, which bound the EPU coolant concentrations as allowed by the TS), and the conservative analytical techniques typically used to establish plant shielding design (such as ignoring the shadow shielding effect of the neighboring sources, rounding up the calculated shield thickness to a higher whole number, etc.), and demonstrates continued adequacy of the existing reactor primary shield, reactor secondary shields, fuel transfer shields and all other shielding outside containment at EPU conditions.

Therefore, the plant radiation zones discussed in FSAR Section 11.6.2, whose boundaries are marked by the shield walls determined to be adequate for EPU operations, are not impacted by the EPU.

IHPB HP RAI-3

Items 2, "Liquid Effluents," and 3, "Gaseous Effluents," (page 2.10.1-18 of Attachment 5) indicate that the 17.6 percent increase over the currently licensed power, would result in a 19.1 percent increase in the concentration of I-133 in the steam generator liquid phase, a 35.7 percent increase in the concentration of I-131 in the secondary coolant, and a 1330 percent increase (mainly from a 11.3 fold increase in moisture carryover) in particulates (cesium) in the secondary coolant. Provide the basis for each of these assumed factors.

NextEra Response

NUREG-0017, Revision 1 (Reference 3) provides primary and secondary coolant isotopic concentrations (μCi/cc) for a specific power level, and the methodology and adjustment factors by chemical group, to develop plant-specific primary and secondary coolant concentrations.

The NextEra EPU assessment discussed in LAR 261, Attachment 5, Section 2.10.1 utilizes the adjustment factors by chemical group and the associated definitions presented in Tables 2-6 and 2-7 of Reference (3), respectively, to estimate the percentage increase in primary and secondary coolant concentrations, by chemical class and radionuclide, due to the EPU. As discussed in Section 2.10.1, these adjustment factors take into consideration plant operating parameters such as reactor coolant mass, steam generator (SG) liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, SG blowdown flow rate, or SG moisture carryover for both current and EPU conditions. The maximum EPU scaling factor for the chemical class applicable to the coolant is then applied as deemed appropriate to the effluent releases. Thus, the 19.1% increase in the concentration of I-133 in the SG liquid phase, a 35.7% increase in the concentration of I-131 in the secondary steam, and a 1330% increase (mainly from an 11.3 fold increase in moisture carryover) in particulates (cesium) in the secondary steam are not assumed values, but conservatively calculated values using the above described methodology.

Using the above methodology, the RCS concentrations for I-131, I-133 and Cs-137 (elements in chemical Classes 2 and 3 of NUREG-0017) are calculated to increase 17.6% at EPU with radionuclides with shorter half-lives increasing slightly more with radioactive decay becoming the dominant removal mechanism.

The 19.1% increase in the concentration of I-133 in the SG liquid is primarily due to the 17.6% increase in reactor coolant concentrations and the approximately 7.5% decrease in SG liquid mass post-EPU.

The 35.7% increase in the concentration of I-131 in the secondary steam is primarily due to the 17.6% increase in reactor coolant concentrations, and the increase in the entrainment of iodine in steam for the EPU conditions (see the NextEra response to IHPB HP RAI-5, below).

The 1330% increase in particulates (cesium) is primarily due to the 17.6% increase in reactor coolant concentrations and the increase in the SG moisture carryover from the current estimated value of 0.015%, to a conservatively estimated EPU value of 0.17% (which represents a potential increase factor of approximately 1130% for entrainment).

IHPB HP RAI-4

The text on page 2.10.1-19 of Attachment 5 concludes that the offsite doses resulting from the 35.7 percent increase in I-131, and 1330 percent increase in radioactive particulates (Cs-137) released in gaseous effluents are "insignificant compared to the dose contribution from tritium." Provide the calculated offsite doses contribution for each of these (Cs, I, and tritium) in liquid and gaseous effluents under current licensed power, and extended power uprate (EPU) conditions.

NextEra Response

Isotope specific dose contributions are not reported in the annual radioactive release reports. The analyses performed to determine the impact of EPU utilized a bounding scaling factor approach. Therefore, doses from individual radionuclides were not calculated for pre-EPU or EPU for gaseous or liquid releases.

The evaluation utilized offsite effluent releases and doses based on an average 5-year set of releases, organ and whole body doses derived from the annual effluent reports for the years 2002 through 2006 including the associated annual average core power level. The reported doses were extrapolated to 100% capacity factor for both units. Releases occurring during periods of a unit shutdown were conservatively lumped with operational releases and were included in the releases scaled for 100% availability and core power. The 5-year annual average of pre-EPU radionuclide releases and doses calculated formed a base case at a 100% capacity factor from which maximum expected EPU doses were projected using the scaling factors derived using NUREG-0017 (Reference 3) methodology and adjustment factors for each chemical class.

Isotope specific importance factors, as shown below, was derived for the 5-year weighted average airborne "Iodine & Particulate" releases reported for the base case because of the high scaling factors estimated for the particulate category. The corresponding 5-year weighted average organ dose (thyroid) was 3.12E-02 mrem.

Nuclide	5-year Weighted Average Releases (Ci/yr)
H-3	7.55E+01
I-131	4.96E-05
I-133	8.35E-07
F-18	1.41E-01
Co-60	6.20E-07
Cs-137	1.09E-07

Using the methodology and pathway data found in the ODCM, an importance factor relative to dose contribution from each radionuclide found in the 5-year average was calculated for each organ and age group. At pre-EPU conditions, the thyroid was determined to be the dominant organ. For the thyroid, tritium had an importance factor in the mid-0.90s with I-131 contributing to the remainder of the dose. The importance factor of Cs-137 and Co-60 was approximately four orders of magnitude less than tritium for all major organs.

Multiplying the radionuclide importance factor by its EPU scaling factor and summing for each organ and age group provides dose multiplier factors that are used in projecting EPU organ doses based on pre-EPU operations. At PBNP, the thyroid remained the critical organ with an approximately 18% increase at EPU projected over the 5-year weighted average pre-EPU organ dose.

IHPB HP RAI-5

Provide an analysis of the impact of the EPU on Main Steam moisture carry over in terms of the potential for higher in-plant dose rates and increased radioactive wastes resulting from the increase in the release of radioactive materials from the steam generators to the secondary side of the plant. Include the impact of dose rates associated with the condensate polishing system.

NextEra Response

The EPU is expected to increase the SG moisture carryover from the current estimated value of 0.015%, to a conservatively estimated EPU value of 0.17%, which represents a potential increase factor of 11.3. The increase in the moisture carryover value will increase the iodine and non-gaseous fission product/corrosion product activity concentrations in the main steam, condensate and feedwater systems.

The iodine concentration in the main steam depends on the SG liquid concentration, iodine partition coefficient in the SG and the moisture carryover fraction.

- The EPU will increase the activity concentration in the reactor primary coolant and the SG liquid by approximately the percentage of uprate, or 17.6%.
- The iodine partition coefficient depends on the iodine composition relative to the chemical species (i.e., organic, elemental, particulate), secondary side water chemistry and the temperature/pressure in the SGs. In accordance with NUREG-0017, a nominal iodine partition coefficient value of 0.01 was used in the EPU evaluation for both the pre-EPU and EPU conditions.
- The SG moisture carryover is estimated to increase from the current value of 0.015% to an estimated EPU value of 0.17%.

Thus, the EPU increase factor for iodine concentration in the main steam is estimated to be approximately 1.176 x [(0.01 + 0.0017) / (0.01 + 0.00015)] = 1.36.

The non-gaseous fission product/corrosion product concentration in the main steam depends on the SG liquid concentration and the moisture carryover. The EPU increase factor for the other fission product/corrosion product concentration in the main steam is therefore estimated to be approximately 1.176 x 11.3 = 13.3.

With respect to radiation levels near components in the main steam system, the noble gases and N-16, which increase by the percentage of the EPU, are considered the dominant source.

With respect to the iodine versus non-gaseous fission product / corrosion product activity concentrations in the main steam, the radioiodines are considered the dominant source since both the activity concentration in the SG liquid, and the fraction of the activity in the boiling liquid that is partitioned or carried over to the steam, is higher for the iodines relative to the other fission products/ corrosion products. Therefore, the dose rate in the plant areas near the main steam, condensate and feedwater systems (where noble gases and N-16 are no longer prevalent) is determined mostly by the iodine concentration in the fluid. As discussed earlier, the EPU increase percentage is expected to be approximately 36%. PBNP does not have a condensate polishing system. Therefore, the major impact of this approximately 36% increase of activity increase will be on the feedwater system.

An increase of feedwater iodine activity concentration by 36% is not a major concern for a pressurized water reactor. The gross iodine concentration in the main steam for a typical PWR with 75 lb/day primary-to-secondary leakage is approximately 2E-7 µCi/g (Reference 3). Without a condensate polishing system, the iodine concentration in the main steam condensate and feedwater will be the same. The dose rate at 1 foot from a 24-inch pipe carrying 2E-7 µCi/g mixed iodine source is approximately 1E-4 mrem/hr. A 36% increase in this dose rate is not significant and is well below the dose rate limit of 0.1 mrem/hr for PBNP shield design zone 0, which is applicable to turbine building areas. The contribution of feedwater to the radiation levels in the plant is insignificant compared to other radiation sources including natural background. Thus, the assumption of an increase in radiation levels in proportion to the EPU near the main steam condensate and feedwater is conservative.

The leakage of main steam, condensate and feedwater will contribute to the liquid radioactive wastes. Because the half-lifes of most other fission products/corrosion products are greater than those of iodines, the EPU increase factor of 13.3 for the other fission products and corrosion product due to increased moisture carryover will manifest more in liquid waste than in the main steam or the feedwater. However, the total activity inventory in the liquid waste that is attributable to main steam and feedwater is small compared to that from the sources originated from the primary coolant and the SG liquid. Therefore, the EPU impact on the liquid waste activity inventory and the dose rates near the liquid waste components is determined primarily by the activity concentration increase factor in the primary coolant and the SG liquid, which is approximately 17.6%.

IHPB HP RAI-6

The fourth paragraph on page 2.10.1-20 of Attachment 5 indicates that administrative and storage controls in the offsite dose calculation manual (ODCM) will ensure that the direct radiation shine from solid radioactive wastes generated and stored onsite will meet the dose limits in 40 CFR 190. Describe the specific provisions of the ODCM that provide these controls.

NextEra Response

The ODCM provides an umbrella for a number of different programs that monitor and provide a path for addressing radiation exposures that are projected to exceed regulatory dose limits. If a dose is projected to exceed either 10 CFR 50, Appendix I or 40 CFR 190, the appropriate regulatory agency must be notified and an action plan developed to identify the causes of the exceedance, and to define and initiate a program of corrective action to bring radiation exposures to the public into compliance with regulatory limits.

Section 4.4 of the ODCM, "EPA Regulations," states:

> "Compliance with the provisions of Appendix I to 10 CFR 50 is adequate demonstration of conformance to the standards set forth in 40 CFR 190 regarding the dose commitment to individuals from the uranium fuel cycle. For 40 CFR 190 compliance, quarterly dose calculations shall include exposures from effluent pathways and direct radiation contributions..."

It is noted that though not specifically called out in the ODCM, solid radioactive wastes generated and stored onsite, will contribute to direct radiation contributions.

In addition, the PBNP radiological environmental monitoring program (REMP) is identified in Section 6.0 "Radiological Environmental Monitoring Program" of the ODCM. This program establishes the appropriate surveillance and monitoring program, monitor and thermoluminescent dosimeter (TLD) locations and sampling frequency. The REMP, including procedures and responsibilities, is contained in the PBNP Environmental Manual (EM), which states "The REMP is conducted to demonstrate compliance with applicable standards, to assess the radiological environmental impact of PBNP operations, and to monitor the efficacy of in plant effluent controls." The REMP and EM are incorporated into the ODCM by reference.

IHPB HP RAI-7

Table 2.12-2 of Attachment 5, "PBNP Extended Power Uprate Ascension Test Plan," indicates that "plant surveys, including radiation shielding measurements, will be performed" at 85 percent and 100 percent of the EPU full power. Describe the scope of these radiation surveys. Verify that they include surveys of all plant areas potentially affected by operations at the EPU full power level.

NextEra Response

The scope of the radiation surveys will be in accordance with plant procedures for routine daily and weekly area radiation surveys. The areas covered by the daily and weekly surveys will ensure that those areas that may expect an increase in radiation following power ascension will be monitored.

IHPB HP RAI-8

As part of the power ascension test, Attachment 5 Table 2.12-2 indicates that survey maps will be "updated as necessary." Describe the anticipated conditions that may require the updating of survey maps.

NextEra Response

Following the performance of a radiation survey, the survey maps are updated for the status of the radiological conditions that have changed as a result of the power ascension, in accordance with plant procedures.

IHPB HP RAI-9

The third paragraph on page 2.10.1-13 indicates that there is only one vital area of the PBNP, (i.e., access to panel C-59) that requires personnel access in post accident conditions as defined in NUREG-0737 Item II.B.2. This statement is in contradiction to the discussion in section 3.2.4 of Attachment 1, "Post –LOCA [loss-of-coolant accident] Vital Area Access," that also includes access to the Unit 1 & 2 NaOH discharge line air-operated valves (26' elevation of the Aux. Building). In addition, the NRC Safety Evaluation referenced in the submittal (Reference 8 [6] to Section 2.10 of Attachment 5) indicates that motor control centers 1B32 and 2B32 are also vital areas. Also, NUREG-0737 specifies that the Control Room and the Tech Support Center (TSC) are vital areas. Provide an evaluation of the impact EPU will have on <u>all</u> vital areas at PBNP, include plant layout drawings indicating maximum dose rates and operator access/egress routs to each. Verify that these areas can be accessed within the dose criteria of GDC 19 as specified in NUREG-0737, Item II.B.2.

NextEra Response

NextEra has reviewed the actions required in response to a loss-of-coolant accident (LOCA) and has determined that there are no vital areas that require short-term access during the post-LOCA recirculation phase. The control room (CR) and technical support center (TSC), although not specifically identified as vital areas, are addressed below.

As discussed in LAR 261, Attachment 5, Section 2.9.2, Radiological Consequences Using Alternative Source Term, and Section 2.10.1, Occupational and Public Radiation Doses, and LAR 241, Alternative Source Term (Reference 4), the CR dose following a LOCA at EPU conditions does not exceed 10 CFR 50.67 limits, therefore, the CR is acceptable for operation at the uprated power.

In response to NUREG-0737 (Reference 5), the TSC is provided with air filtration (including prefilters and high efficiency particulate (HEPA) and charcoal filters) and shielding to maintain acceptable radiological habitability during an accident, appropriate radiation and airborne radioactivity monitoring instrumentation, and a permanent source of emergency power. Anti-contamination clothing, respiratory protection equipment and other protective gear is kept in the TSC for use in travel from the TSC to the CR, or during times that there are radiological conditions in the TSC requiring protective equipment. The Emergency Plan Implementing Procedures provide for transfer of TSC management personnel to the CR if the TSC becomes uninhabitable as prescribed by NUREG-0696 (Reference 6). The time between the CR and TSC is estimated to be 90 seconds and the only security barrier between the CR and TSC is the cardkey access door into the CR. The remaining TSC personnel are evacuated to a safe area. The access path to the safe area is identified by radiation protection personnel, there are no pre-determined maps or pathways. NextEra has determined that the TSC is acceptable for operation at EPU conditions.

LAR 261, Attachment 5, Section 2.10.1.2, Technical Evaluation – Post Accident Vital Area Accessibility, Page 2.10.1-13, identifies the C59 panel as the only vital area with access requirements. The discussion in LAR 261, Attachment 1, Section 3.2.4, Post-LOCA Vital Area Access, Page 1.0-34, lists the C59 panel and the Unit 1 and 2 sodium hydroxide (NaOH) discharge line air-operated valves (AOVs) as vital areas. What appears to be a contradiction between the two sections is explained by the fact that the C59 panel and the AOVs are located adjacent to each other and are both accessed to manually isolate spray additive tank discharge valves. IA calculation was performed that justifies leaving the AOVs open during an accident. This eliminates the need for operator action to manually close the valves. Therefore, references contained in Attachment 5 of LAR 261, Pages 2.10.1-12 through 2.10.1-16 to the C59 panel, spray additive tank discharge valves, and associated scaling techniques are deleted, and the vital areas listed on Page 1.0-34 of Attachment 1 of LAR 261 are also deleted.

LAR 261, Attachment 5, Section 2.10.1.1.3, Regulatory Evaluation – Post Accident Vital Area Accessibility, Page 2.10.1-3, lists the 1B32 motor control center (MCC) area, 2B32 MCC area, and the C59 panel area as needing additional shielding to meet the requirements of NUREG-0737, Item II.B.2.2. The NRC Safety Evaluation (Reference 7) provided NRC acceptance of the PBNP NUREG-0737, Item II.B.2.2, submittal for plant shielding. As stated above, NextEra has determined that access to the C59 panel is no longer required in the post-LOCA recirculation phase. Based on a review of the PBNP emergency operating procedures, NextEra has also determined that access to MCC 1B32 and to MCC 2B32 are steps that are not required. Therefore, there are no vital areas at PBNP that are required to be accessed during the post-LOCA recirculation phase.

IHPB HP RAI-10

The discussion of source term assumptions for the NUREG-0737, II.B.2 analysis on the bottom of page 2.10.1-13 of Attachment 5, references the previous PBNP license amendment (LAR No. 241, dated December 8, 2008) authorizing the use of the Alternate Source Term. Page 9 of LAR No. 241 states that the LOCA dose analysis assumes that the operator "throttle both the Containment Spray (CS) and Residual Heat Removal (RHR) systems." Verify that no vital area access is required for the operator to throttle the flow for these two systems.

NextEra Response

No vital area access is required for the operator to throttle the flow for containment spray (CS) and residual heat removal (RHR) systems. LAR 241, Enclosure 3, Section 6.1, Page 28 of 89, states:

> "New operator actions to align core injection and CS flow to preset throttled positions will be introduced as a result of the new LOCA radiological analysis. Once the RHR system is aligned to take pump suction from the containment sump, all of the manual operator actions to align the RHR and CS systems for recirculation spray will be accomplished from the CR with no local operator action required. All the necessary controls and instrumentation necessary to place the CS system in recirculation spray operation will be provided in the CR."

IHPB HP RAI-11

The list of regulatory commitments listed in LAR No. 241, indicates that the licensee will permanently install a radiation shielding on the control room doors and windows to ensure control room habitability. Describe the shielding analysis that supports the design of these permanent shields. Provide all input parameters and assumptions used in this analysis. Verify the analysis used a source term consistent with EPU power conditions. Include the dose rate contribution to the operators from direct radiation shine from plant systems, structures and components containing the LOCA source term.

NextEra Response

The CR shielding analysis includes the dose contribution due to direct shine from the external cloud and from contained sources following a LOCA. The external cloud contribution includes the contributions due to containment leakage and emergency core cooling system (ECCS) leakage. The contained sources include shine from the containment structure and the CR heating, ventilation, and air conditioning (HVAC) filter. The 30-day deep dose equivalent (DDE) to a CR operator due to the airborne source in containment, the passing cloud source and the CR filter source was calculated with a focus on the most vulnerable areas, specifically, near the vicinity of the south door opening, the north door opening, and the east window opening.

The CR direct shine dose estimates are based on proposed shielding modifications at the south door opening, the east window opening of the CR, and the floor above the CR. The 9 ft wide by 10 ft high south door opening is assumed to be covered by a 2-inch steel plate. The 12 ft wide by 9 ft high east window opening is assumed to be covered with a 3-inch steel plate. The north door opening does not require shielding modifications because it opens into the Operations office. The approximately 2.5 ft wide space on the floor of the mechanical equipment room (located above the CR) between the CR filter concrete pad and the neighboring heat exchanger concrete pad is assumed to be filled with 4-inch poured concrete. The above proposed modifications are being implemented with the exception of the 2-inch steel plate on the south door. Due to construction/installation constraints, and to facilitate CR access via the south door, a 7-inch concrete labyrinth with a roof has been designed as the effective replacement shield.

The radiation source term used to calculate the CR direct shine dose is the same as that used to calculate the CR inhalation/submersion dose, which is described in Enclosure 3 of LAR 241 (Reference 4). The inventory of the fission products in the reactor core is based on maximum full-power operation at an analyzed EPU core power level of 1811 MWt (to account for a 0.6% power uncertainty), and uprated values of fuel enrichment and burnup. A multiplier of 1.04 was applied to core inventory to account for cycle-to-cycle variations in enrichment, cycle burnup, and loading. The core activity inventory is provided in LAR 241, Enclosure 3, Table 5.

The CR inhalation/submersion dose and direct shine dose due to a LOCA are calculated based on AST and in accordance with NRC Regulatory Guide (RG) 1.183 (Reference 8). Activity from the core is released to the containment and then to the environment by containment leakage and leakage from the ECCS. The major input parameters and assumptions are listed in LAR 241, Enclosure 3, Table 18. The airborne source in the containment, the external cloud source due to containment leakage and ECCS leakage, and the CR HVAC filter source are considered in the CR shielding analysis.

<u>Containment Structure Source</u>

The containment shine source includes all airborne sources above the operating floor at El. 66 ft. The activity concentration in the containment atmosphere is calculated as follows:

The fission products released from the fuel are assumed to mix instantaneously and homogeneously throughout the free volume (sprayed and unsprayed) of the primary containment ($V = 1.00E+6\ ft^3$) as it is released from the core. Per RG 1.183, two fuel release phases are considered for the LOCA: a) the gap release, which begins 30 seconds after the LOCA and continues for 30 minutes; and b) the early In-Vessel release phase which begins 30 minutes into the accident and continues for 1.3 hours. The fission product release fractions and the timing/duration of releases are provided in Tables 16 and 17 of LAR 241, Enclosure 3.

Since the sump pH is controlled to values of 7 and greater, the chemical form of the radioiodine released from the fuel is 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. With the exception of noble gases, elemental and organic iodine, fission products are assumed to be in particulate form.

The activity depletion/transport model takes credit for aerosol/iodine removal via containment sprays. It considers mixing between the sprayed and unsprayed regions of the containment, reduction in airborne radioactivity in the containment by spray removal lambdas and isotopic in-growth due to decay. The injection spray starts at T = 0 and ends at T = 60 minutes. The recirculation spray starts at T = 80 minutes and ends at T = 4 hours 20 minutes. The spray coverage volume is 58.2% of the total containment free volume. The time dependent spray removal coefficients are listed in Table 18 of LAR 241, Enclosure 3. The mixing rate between the sprayed and the unsprayed volumes is 67,000 cfm. The maximum decontamination factor (DF) for elemental iodine is limited to 200. The particulate spray removal coefficient is reduced to 10% of the original value when particulate DF reaches 50. A sedimentation removal coefficient of 0.1 per hr for all particulate in the containment is modeled in all areas of containment except when an area is credited with spray removal.

The volume of containment atmosphere source that contributes to the CR direct shine dose includes the volume in the cylindrical section and the volume in the spherical-toroidal dome. The total volume is modeled as an equivalent cylindrical source. Only the major intervening shielding is credited (e.g., the containment liner/wall, CR walls and ceiling, some walls and floors in the primary auxiliary building, shielding provided by the Operations office, etc.).

External Cloud Source

The external cloud source includes contribution from containment leakage, ECCS leakage and refueling water storage tank (RWST) back-leakage.

The containment is assumed to leak at the proposed TS leak rate of 0.2 weight percent per day for the first 24 hours of the accident and then at half that rate (0.1 weight percent per day) for the remainder of the 30-day period following the accident.

The total ECCS recirculation leakage modeled in the analysis is 0.21 gpm (i.e., consistent with RG 1.183 guidance, the estimated 400 cc/min total ECCS leakage outside the containment is doubled to 800 cc/min). The 800 cc/min total ECCS recirculation leakage, 300 cc/min is assumed to leak into the PAB, and 500 cc/min is assumed to leak back to the RWST. ECCS leakage is conservatively assumed to start at T = 0 hr after the LOCA and continue for the duration of the accident. Ten percent of the halogens associated with ECCS leakage to PAB become airborne and are exhausted (without mixing and without holdup) to the environment.

The iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form. However, when the solution leaks into the RWST, the iodine will be in an acidic solution such that there is the possibility of conversion of iodine compounds to form elemental iodine. The iodine release rate versus time from the RWST vent utilized in the shielding analysis is consistent with that discussed in LAR 241.

The activity concentration of the external cloud at the CR peripheral (Ci/m^3) is determined by the release rate (Ci/sec) and the atmospheric dispersion factor (sec/m^3). The atmospheric dispersion factors from the containment surface (used for containment leakage), from the auxiliary building vent stack (used for ECCS leakage), and from the RWST (used for RWST back leakage) to the CR door and window areas are given in Table 18 of LAR 241, Enclosure 3. The external cloud is assumed to be a uniformly distributed source from El. 26 ft to approximately 1000 meters above the ground. The cloud source is divided into three parts to assess the dose impact at the south and north door locations of the CR as well as in the proximity of the east window. Similar to the containment direct shine contribution, only the major intervening shielding is credited (e.g., CR walls and ceiling, shielding modifications at the south door and the east window, Operations office walls and ceiling, turbine building floors, etc.).

Control Room HVAC Filter Source

The CR HVAC filter shine dose is calculated based on accumulation of a) the particulate fission products and elemental/organic/particulate iodines resulting from containment leakage; and b) elemental/organic iodines resulting from ECCS leakage and RWST back-leakage. The elemental and organic iodines are assumed to be accumulated on the charcoal filter. The particulate iodines and the other particulate fission products are assumed to be accumulated on the HEPA filter. All non-noble gas radioactive materials that enter the CR, either via the CR intake or CR in-leakage, are accumulated on the filter with 100% efficiency. This maximizes the loading on the filter for the duration of the accident. The CR filter is located at the northeast corner of the mechanical equipment room above the CR. The shielding modeling includes the 14-inch CR concrete ceiling and the 4-inch concrete pads under the filter and the neighboring heat exchanger. The approximately 2.5 ft gap between the two concrete pads is assumed to be filled with 4" poured concrete.

Computer code SW-QADCGGP is used to calculate the direct shine dose to an operator in the CR from the airborne source inside the containment, the external cloud source, and the CR charcoal/HEPA filter sources. SW-QADCGGP is a Shaw Stone and Webster version of the industry standard point-kernel radiation shielding computer code QAD-CGGP. The geometry utilized in the model does not have any significant un-accounted for contributions for scattering paths from the source to the receptor. Multiple receptors are placed inside the CR to ensure that the maximum dose is calculated. The source-shield-receptor geometry is such that the dose due to oblique angle scattering is not significant. The most conservative buildup factor is used if the gamma rays traverse through a multiplicity of materials and the last material constitutes less than 3 mean-free-paths.

The calculated direct shine doses from the containment atmosphere source, external cloud source and CR HVAC filter source are as follows:

Location	30-Day CR Dose (Rem)
South Door Area	
Containment Shine	3.71E-05
External Cloud Shine	2.28E-01
CR Filter Shine	Negligible
Total	**2.28E-01**
North Door Area	
Containment Shine	2.32E-05
External Cloud Shine	2.02E-01
CR Filter Shine	3.82E-02
Total	**2.40E-01**
East Window Area	
Containment Shine	Negligible
External Cloud Shine	2.74E-01
CR Filter Shine	Negligible
Total	**2.74E-01**

IHPB HP RAI-12

A summary of the scaling factors use to adjust the pre-EPU dose associated with post accident vital area access to panel C-59 to post-EPU conditions, is provided on pages 2.10.1-15 and 16 of Attachment 5. The first bullet listed is an "EPU dose rate scaling factor of 1.4." Provide a detailed quantitative description of the factors and assumptions used to arrive at this factor of 1.4.

NextEra Response

This question is no longer applicable. Refer to the NextEra response to IHPB HP RAI-9.

IHPB HP RAI-13

The third bullet (top of page 2.10.1-16) is a factor of 0.5 to adjust for the "re-evaluated" time estimate required for operator actions at panel C-59. Provide a detailed analysis that demonstrates an operator residence time of 10 minute at C-59 under LOCA conditions (e.g., operator dressed in full protective clothing).

NextEra Response

This question is no longer applicable. Refer to the NextEra response to IHPB HP RAI-9.

IHPB HP RAI-14

Page 2.9.10.1-2 of Attachment 5 gives 0.5 rem whole body acceptance criterion for the postulated waste gas release events analyzed in Section 2.9.10.1.2 of Attachment 5. The basis for this acceptance criterion is given as the NRC Branch Technical Position (BTP) ETSB 11-5 (1981), as attached to Chapter 11.3 of NUREG 0800 Standard Review Plan (SRP). As discussed in the BTP, this 0.5 rem criterion was intended to insure that the postulated events "would not exceed the guidelines of 10 CFR 20 for a unique unplanned release." At the time (1981) the radiation release limits in 10 CFR 20 were based on an allowable dose of 0.5 rem/year to a member of the public. However, 10 CFR 20 was revised in 1991, lowering the allowable dose, for a member of the public, to 0.1 rem/year. Accordingly, Revision 3 to the SRP, and the related BTP 11-5, revised the acceptance criteria for this anticipated operational occurrence to "1 mSV (0.1 rem) at the exclusion area boundary." Provide a technical justification why the acceptance criteria in Rev. 3 of the BTP should not apply to the PBNP EPU, or provide an analysis demonstrating that the postulated releases of waste gas, under EPU conditions, will not exceed 0.1 rem.

NextEra Response

Consistent with Issue 11 of NRC Regulatory Issue Summary (RIS) 2006-04 (Reference 9), the events resulting in accidental waste gas releases were excluded from the full-scope implementation of the AST methodology (Reference 12), and these continue to use the current licensing basis analysis methodology and acceptance criteria of 500 mrem whole body. The 10 CFR 20 revision and Standard Review Plan update did not require that a plant's current licensing basis analyses be revised to meet the lower limit. Precedent for continued use of the 500 mrem limit for the non-design basis accident dose analyses in support of an uprate for a plant originally licensed and currently operating with that limit is provided by Reference (10) (Section 3.5.2.10), Reference (11) (Section 3.5), and Reference (12) (Section 2.9.5).

References

(1) NRC letter to NextEra Energy Point Beach, LLC, dated March 31, 2010, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Health Physics Branch RE: Extended Power Uprate (TAC Nos. ME1044 and ME1045) (ML100820459)

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

(3) NUREG-0017, Revision 1, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, April 1985

(4) FPL Energy Point Beach, LLC letter to NRC, dated December 8, 2008, Submittal of License Amendment Request 241, Alternative Source Term (ML083450683)

(5) U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980 (ML051400209)

(6) U.S. Nuclear Regulatory Commission, "Functional Criteria for Emergency Response Facilities," NUREG-0696, February 1981 (ML051390358)

(7) NRC letter to Wisconsin Electric Power Company, dated November 3, 1983, NUREG-0737 Item II.B.2.2, "Plant Shielding Modifications for Vital Area Access," Point Beach Nuclear Plant, Unit Nos. 1 and 2

(8) U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183 (ML003716792)

(9) U.S. Nuclear Regulatory Commission, "Experience with Implementation of Alternative Source Terms," Regulatory Issue Summary 2006-04 (ML053460347)

(10) NRC letter to Entergy Nuclear Operations, Inc, dated October 27, 2004, Indian Point Nuclear Generating Unit No. 2 - Issuance of Amendment Re: 3.26 Percent Power Uprate (TAC No. MC1865) (ML042960007)

(11) NRC letter to Entergy Nuclear Operations, Inc, dated March 24, 2005, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters (TAC No. MC3552) (ML050600380)

(12) NRC letter to FirstEnergy Nuclear Operating Company, dated July 19, 2006, Beaver Valley Power Station, Unit Nos. 1 And 2 (BVPS-1 and 2) - Issuance of Amendment Regarding the 8-Percent Extended Power Uprate (TAC Nos. MC4645 and MC4646), (ML061720274)