UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

March 23, 2011

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, EXTENDED POWER UPRATE APPLICATION

Dear Chairman Jaczko:

During the 581st meeting of the Advisory Committee on Reactor Safeguards, March 10-12, 2011, we reviewed the Extended Power Uprate (EPU) application for Point Beach Nuclear Plant (PBNP), Units 1 and 2, and the associated NRC staff's Safety Evaluation Report (SER). Our Subcommittee on Power Uprates also reviewed this matter on February 24-25, 2011. During these reviews, we had the benefit of discussions with the representatives of the NRC staff, NextEra Energy (the licensee) and their consultants. We also had the benefit of the documents referenced.

RECOMMENDATION

The application for an extended power uprate of PBNP, Units 1 and 2, should be approved.

BACKGROUND

PBNP, Units 1 and 2, are two-loop Westinghouse-designed pressurized water reactors (PWRs), originally licensed to operate at 1518.5 MWt in 1970 and 1973, respectively. The units were subsequently approved for a 1.4% measurement uncertainty recapture power increase to allow operation at the currently licensed thermal power (CLTP) of 1540 MWt. In the current amendment, NextEra applied for an extended power uprate (EPU) of approximately 17% increase above the CLTP to 1800 MWt.

The licensee undertook major plant modifications to support operation at the higher power level. The reactor coolant system and reactor vessel internals will remain the same as pre-EPU, as will the primary coolant flow rate and the Westinghouse-designed 14x14, 422 Vantage+ fuel design. However, the core design, operating and control strategies will change. The higher power level is achieved by increasing the average enrichment of fuel assemblies, the amount of new fuel in each reload, the temperature rise across the core, and the operating reactor coolant average temperature.

On the secondary side, the modifications and upgrades to accommodate the higher steam and feedwater flows needed to produce the augmented power will be extensive. The main steam isolation valve internals will be upgraded to address flow induced vibration (FIV) and closure loads. The check valves and piping supports will be modified as well to cope with the larger FIV loads. The condensate and the main feedwater pumps will be replaced with pumps that can provide higher flow rates and pressure heads. The feedwater heaters, heater drains, and extraction steam systems will be replaced to meet the higher operating pressures and flow rates. New feedwater isolation valves will be installed and a new higher capacity motor-driven auxiliary feedwater (AFW) pump to serve both steam generators (SGs) will be installed in each unit, eliminating the need for cross unit sharing. The current motor-driven AFW system pumps will be maintained as non-safety related standby pumps. Additional modifications to improve plant safety will be made to the compressed gas systems such as installation of a self-cooled instrument air compressor that is independent of service water cooling and aligned for automatic operation.

The PBNP steam generators, replaced in 1983 for Unit 1 (with Westinghouse Model 44F) and in 1996 for Unit 2 (with Westinghouse Model Delta-47), will remain the same as they are capable of handling the higher power levels with the increased core-outlet coolant temperature giving a higher temperature difference across the tubes. The secondary side flow rates will increase, with the steam flow rate increasing by approximately 20%.

Life extension of the PBNP units was approved in 2005. Therefore, the EPU review evaluates the impact on the license renewal and the associated commitments.

DISCUSSION

Safety Analysis Results

The EPU will result in lower core-inlet and higher core-outlet temperatures in the coolant and higher volumetric power density in the core, which reduce safety margins but still maintain adequate values.

The licensee provided analyses of transients such as decrease (loss) in reactor coolant system (RCS) flow, reduced secondary cooling, and overcooling. These analyses indicate that the safety criteria for departure from nucleate boiling ratio, RCS pressure, fuel linear heat generation rate, and pressurizer fill levels are met. The analyses were based on conservative bounding conditions, and we concur with the staff's acceptance of the results. Since some of the analytical methodologies used were approved several decades ago, consideration should be given to reviewing the level of conservatism of these analytical methods and codes in light of more recent information.

Similarly, the fuel enthalpy safety criterion of less than 200 cal/gm should be reviewed, which we recommended in the past, as the available data suggest it should be set at a lower value. Nonetheless, the licensee's analysis methodology of reactivity addition events yields conservatively high calculated values. We find these results acceptable.

Large and small break loss of coolant accidents (LOCAs) were analyzed for EPU conditions. The results indicate adequate margins to the acceptance criteria for the large break LOCAs when the best estimate (plus uncertainty) approach in the approved ASTRUM methodology is used. The small break LOCA margins continue to be large for EPU conditions, the results being calculated with the approved NOTRUMP code. The staff performed calculations for similar conditions with RELAP5 and confirmed that large margins to the acceptance criteria exist, in part, due to the large capacity of the high pressure injection system.

With regard to long-term cooling, the licensee has put in place a program to resolve issues related to sump screen blockage and downstream effects due to LOCA debris (GSI-191) under EPU conditions. While final resolution of this issue lies in the future, the licensee did submit analyses on long-term cooling in the absence of debris effects. These indicate that adequate net positive suction head margin exists to prevent cavitation of the recirculation pumps without requiring credit for containment accident pressure.

Boron precipitation during long-term cooling, especially for hot leg breaks, is important and was evaluated in some detail. In PBNP, the low pressure, high capacity emergency cooling system injects into the upper plenum. It is difficult to estimate the rate of increase in core boron concentration in this situation, inasmuch as the experimental data available are more applicable to cold leg injection. The licensee presented WCOBRA/TRAC calculations to support assumptions regarding mixing of the lower plenum volume which significantly affects the rate at which boron concentrates in the core. The staff conducted independent analyses which took into account the time needed to mix part of the lower plenum fluid to dilute the boron. The staff analysis showed a reduction in the time available to switch from containment spray to high pressure injection to flush out the boron. The emergency operating procedures have been modified accordingly. Both staff and licensee calculations credit lower plenum mixing, which might not be conservative, but they also neglect boron carry out of the break, which is very conservative. It is difficult to estimate the levels of conservatism introduced by the various assumptions. Experiments that are more applicable to upper plenum injection are, therefore, desirable. For this particular EPU application, we concur with the staff that the measures now being taken are adequate.

EPU analysis of transients such as loss of load for PBNP, Units 1 and 2, in some instances, invoke the original plant licensing basis to meet overpressure acceptance criteria. This requires an exception from the staff guidance (Standard Review Plan Section 5.2.2, Subsection 3.B.), which states that the analysis should assume the “second safety-grade signal from the reactor protection system initiates the reactor scram.” If the first safety-grade signal is allowed to lead to scram, then the licensee is able to meet the overpressure acceptance criteria using bounding conditions. The staff accepts this approach based on the original plant licensing basis, and we concur with their conclusion for this specific plant.

Materials Effects

The power uprate will result in increased fast neutron flux, temperature, and flow velocity within the reactor vessel as well as higher temperature and flow velocity in portions of the primary and secondary system. These changes can increase the oxidation potential of the reactor coolant and the rate of irradiation hardening of core materials, and accelerate materials degradation rates. The licensee has evaluated relevant materials degradation mechanisms including stress corrosion cracking (SCC), irradiation assisted stress corrosion cracking (IASCC), fatigue, radiation embrittlement, stress relaxation, flow-assisted corrosion (FAC), and flow-induced vibration.

The EPU will increase the susceptibility of reactor internal components to IASCC. However, as part of its license renewal commitments, the licensee will implement an aging management program to address this issue.

The licensee has demonstrated that the vessel materials will have acceptable upper shelf energies through the end of the operating license. The licensee's Pressure–Temperature Limit Report process can adequately address the impact of the power uprate on the pressure–temperature limits. The nil ductility reference temperature for all vessel materials in Unit 1 remains below the 10 CFR 50.61 pressurized thermal shock (PTS) screening criteria through the expiration of its operating license. One of the Unit 2 vessel welds will exceed the 10 CFR 50.61 screening criteria before the expiration of its operating license. This would occur even without the power uprate and was addressed by the licensee during license renewal of Unit 2. The licensee chose to use the 10 CFR 54.21(c)(iii) option to manage the PTS issue. In accordance with its commitments under license renewal, Unit 2 will continue to operate with hafnium absorber assemblies to reduce flux to the vessel until the PTS issue can be resolved via an alternative analysis methodology.

High nickel alloys are used for control rod drive mechanism nozzles, dissimilar metal welds, and steam generator tubing. Susceptibility to SCC of these materials tends to increase with increasing temperature. However, except for the steam generator tubing in Unit 1 and some bottom-mounted instrumentation nozzles, all the nickel alloys in these units exposed to reactor coolant are Alloy 690/152 materials, which are highly resistant to SCC. The bottom-mounted instrumentation nozzles will actually operate at slightly lower temperatures after the power uprate. The Alloy 600 TT tubes in the Unit 1 steam generators are not as resistant as Alloy 690, but have performed well at Point Beach and at temperatures similar to those after the power uprate at other units. Inspections under the NEI 97-06 steam generator program provide additional assurance that any potential increase in susceptibility can be adequately managed.

The increased temperatures in the hot leg could lead to increased thermal aging embrittlement of cast austenitic stainless steels. The analysis of this issue was evaluated during license renewal and remains valid under the EPU, since it was performed using lower-bound values for the toughness rather than values based on a time-temperature history.

The power uprate will result in higher velocities and temperature changes which could affect FAC. Analyses using the CHECWORKS code show that both increases and decreases in FAC rates are expected. The licensee has committed to adjusting inspection scope to account for the predicted increases in wear rates.

Flow-Induced Vibration

The increase in secondary system flow rates can lead to increases in vibration amplitudes of steam generator tubing especially in the U-bend region. Increased vibration levels could lead to increased susceptibility to failure by fatigue or wear. The licensee's analyses showed that vibration levels met the acceptance criteria developed by Westinghouse. For Unit 1, the fluid kinetic energy associated with the uprate conditions appears to be relatively high compared with that associated with comparable steam generators in other reactors. However, Westinghouse stated that the conditions are consistent with the range over which the methodology has been validated. Under the current operating conditions the generators have shown good performance with respect to wear and vibration. In addition, other plants with Westinghouse steam generators have shown good performance with respect to wear and vibration after power uprates. Again, inspections under the NEI 97-06 steam generator program provide additional assurance that any potential increase in susceptibility can be adequately managed.

The licensee will perform testing to assess vibration levels of piping systems during plant operation in accordance with Part 3 of the ASME OM Code, "Vibration Testing of Piping Systems." The Code establishes test methods, test intervals, parameters to be measured and evaluated, acceptance criteria, corrective actions, and records requirements. Compliance with the Code requirements provides adequate assurance that excessive vibration will not lead to degradation and failure of the piping.

Risk Evaluations

Although the licensing application is not a risk-informed one, the licensee performed a quantitative assessment of the change in risk associated with EPU for internal events and a qualitative evaluation of the change in risk for external events. The analysis looks for impacts on initiating event frequency, event tree sequence models, systems models, failure/ maintenance data, and human response. It considers both what was previously modeled and the need for new models. Both the human reliability analysis (HRA) and the probabilistic risk assessment (PRA) systems model were considered. The greatest impact on risk was associated with human performance in light of reduced times for response because of the EPU.

The engineering evaluations to support the PRA and HRA provide a reasonable basis for the assessment. Unfortunately, details of the quantification of PRA and HRA events are not fully convincing and appear to ignore uncertainty. There is no specific basis or identified evidence to support the "expert judgment based" changes. The EPU application indicates that the "CBDTM/THERP method" in the EPRI HRA Calculator was used. However, these are two different methods, and it is not clear how either method yields such precise results based on reduction in time available.

In spite of possible problems in the quantification, the licensee decided to develop compensating design changes that improve reliability and will automate many current human actions. These changes actually reduce the risk below the pre-EPU risk. The fact that these improvements do not directly address issues associated with the uprate is not a concern; finding alternative changes to compensate for identified weaknesses is often the most effective approach, sometimes the only approach.

Electrical Systems Impacts

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The staff's review focused on whether the loss of the largest operating unit on the grid, the loss of the nuclear unit, or the loss of the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU.

An Interconnection System Impact Study report was performed for the Midwest Independent System Operator (MISO) by the American Transmission Company (ATC) to evaluate the impact of increased electrical output of PBNP on the reliability of the local 345 kV transmission system and MISO bulk power systems. The licensee provided a summary of the ATC grid stability study for the proposed EPU at PBNP. The summary of the ATC grid stability study demonstrates that the PBNP electrical output can be increased up to each unit's uprated maximum generating capacity of 641.6 mega-watts electric (MWe) gross per unit without compromising the offsite power grid stability or its capability to supply in-plant loads.

The proposed increase in the generator output requires the rewinding of the stator and rotor of the existing main unit generators, the installation of new generator output breakers, and replacement of each unit's main generator output transformer to accommodate the new maximum main unit generator gross electrical output.

Based on their review of the grid impact study, the staff found that with a combination of system upgrades along with operating restrictions, the thermal, voltage, and stability performance of the 345 kV offsite power system will not be degraded by implementation of the EPU. The staff also found that the proposed EPU should not adversely affect the stability of the electric power grid since the proposed increase is within the limit identified in the ATC load study.

Power Ascension Testing and Large Transient Testing

The licensee has proposed a systematic power ascension test program. It includes tests to validate the performance of components and control systems, both at an individual system and integrated response level. Transient tests include a turbine overspeed trip and the resulting control system response for steam generator and feedwater heater levels. The planned duration of the power ascension is 21 days, and includes monitoring for vibrations, plant calorimetric tests, and verification of ultrasonic flowmeter calibrations. The procedure requires holds to gather and evaluate plant data after each 3% increase in power above the current licensed power level. If unexpected results are obtained at any level, the power will be reduced to the previous (acceptable) power level until the problems are resolved.

No large-scale transient testing is planned even though substantial modifications will be made on the secondary side, particularly to the feedwater systems. LOFTRAN calculations of transients, validated with the data available from Ginna and other similar plants would be used in place of large transient testing. The licensee stated that the information from the individual components and system level response tests, the data from plant transients, which have already occurred after some of the modifications were in place, and the results of the LOFTRAN calculations are sufficient to demonstrate adequate large scale transient performance. The staff has accepted this position and we concur.

Summary

The proposed extended power uprate at Point Beach Nuclear Plant, Units 1 and 2, will result in significantly changed operating conditions, but comparable to those in currently operating PWRs of similar design. Although safety margins will be decreased, the remaining margins will still be sufficient to ensure that the safety limits and acceptance criteria will not be challenged. The modifications made to PBNP to strengthen the case for the EPU have resulted in lower calculated core damage frequency (CDF) and large early release frequency (LERF) compared to pre-EPU values. The EPU application for PBNP, Units 1 and 2, should be approved.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

REFERENCES

1. Draft Safety Evaluation of the Office of Nuclear Reactor Regulation Related to Amendment Nos. 242 and 246 to Facility Operating Licenses Nos. DPR-24 and DPR-27, "Point Beach Nuclear Plant Units 1 and 2," (ML110450159). Revision Dated February 11, 2011.

2. Letter from Larry Meyer, FPL Energy Point Beach, LLC Site Vice President, "License Amendment Request 261, Extended Power Uprate," April 7, 2009, (Package ML091250562).

3. Letter from Terry A. Beltz, Senior Project Manager, NRC, to Mr. Larry Meyer Site Vice President, NextEra Energy, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 Issuance of License Amendments Regarding Use of Alternate Source Term, (TAC NOS. ME0219 and ME0220)," January 1, 2011, (Package ML110240054).

4. Letter from Larry Meyer, Site Vice President, FPL Energy Point Beach, LLC, "Submittal of License Amendment Request 241 Alternate Source Term," Point Beach Nuclear Plant Units 1 and 2, December 8, 2008, (ML083450683).

5. Letter from Terry A. Beltz, Senior Project Manager, NRC, to Mr. Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 Issuance of License Amendments RE: Auxiliary Feedwater System Modification, (TAC NOS. ME1081 and ME1082)," (ML110230016). Revision Dated February 11, 2011.

6. Letter from Terry A. Beltz, Senior Project Manager, NRC, to Mr. Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments RE: Revision of Protection Systems (RPS) and Engineered Safety Feature Actuation System (ESFAS) Instrumentation Setpoints (TAC NOS. ME1083 and ME1084)," (ML110320060). Revision Dated January 28, 2011.

7. Letter from Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC, "License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information: RE- NRC 2011-0027," March 2, 2011 (ML110620093).

8. Letter from Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC,"License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information: RE- NRC 2011-0028," March 4, 2011 (ML110660049).

5. Letter from Terry A. Beltz, Senior Project Manager, NRC, to Mr. Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 Issuance of License Amendments RE: Auxiliary Feedwater System Modification, (TAC NOS. ME1081 and ME1082)," (ML110230016). Revision Dated February 11, 2011.

6. Letter from Terry A. Beltz, Senior Project Manager, NRC, to Mr. Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC, "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments RE: Revision of Protection Systems (RPS) and Engineered Safety Feature Actuation System (ESFAS) Instrumentation Setpoints (TAC NOS. ME1083 and ME1084)," (ML110320060). Revision Dated January 28, 2011.

7. Letter from Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC, "License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information: RE- NRC 2011-0027," March 2, 2011 (ML110620093).

8. Letter from Larry Meyer, Site Vice President, NextEra Energy Point Beach, LLC,"License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information: RE- NRC 2011-0028," March 4, 2011 (ML110660049).

Accession No: **ML110730459** **Publicly Available** **Y** **Sensitive** **N**

Viewing Rights: ☒ NRC Users or ☐ ACRS Only or ☐ See Restricted distribution

OFFICE	ACRS	SUNSI Review	ACRS	ACRS	ACRS
NAME	ZAbdullahi	ZAbdullahi	CSantos	EHackett	EMH for SAK
DATE	03/23/11	03/23/11	03/23/11	03/25/11	03/25/11

OFFICIAL RECORD COPY

Letter to The Honorable Gregory B. Jaczko, Chairman, from Said Abdel-Khalik, ACRS Chairman, dated March 23, 2011

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, EXTENDED POWER UPRATE APPLICATION

ML# 110730459

Distribution:
ACRS Staff
ACRS Members
S. McKelvin
L. Mike
B. Champ
C. Jaegers
M. Orr
RidsSECYMailCenter
RidsEDOMailCenter
RidsNMSSOD
RidsNSIROD
RidsFSMEOD
RidsRESOD
RidsOIGMailCenter
RidsOGCMailCenter
RidsOCAAMailCenter
RidsOCAMailCenter
RidsNRRPMAAdamsResource
RidsNROOD
RidsOPAMail
RidsRGN1MailCenter
RidsRGN2MailCenter
RidsRGN3MailCenter
RidsRGN4MailCenter