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

March 20, 2012

Mr. R.W. Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE FLORIDA POWER AND LIGHT TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4, LICENSE AMENDMENT REQUEST FOR AN EXTENDED POWER UPRATE

Dear Mr. Borchardt:

During the 592nd meeting of the Advisory Committee on Reactor Safeguards, March 8-10, 2012, we completed our review of the license amendment request (LAR) for the extended power uprate (EPU) of Turkey Point Nuclear Plant (PTN), Units 3 and 4, and the associated Safety Evaluation Report (SER). The SER was also reviewed during our 590th meeting, January 19-20, 2012. Our Subcommittee on Power Uprates reviewed this matter in meetings on December 14, 2011, and February 24, 2012. These meetings included a review of the assessments by the licensee, Florida Power and Light (FPL), and the NRC staff of the effects of the degradation of fuel thermal conductivity with burnup. During these reviews, we had the benefit of discussions with representatives of the staff, FPL, and their consultants. We also had the benefit of the documents referenced.

**RECOMMENDATION**

The license amendment request for an extended power uprate of Turkey Point Nuclear Plant, Units 3 and 4, should be approved.

**BACKGROUND**

PTN, Units 3 and 4, are three-loop Westinghouse-designed pressurized water reactors (PWRs), originally licensed to operate at 2200 MWt in 1972 and 1973, respectively. The units were subsequently approved for a 4.5% stretch uprate power increase to allow operation at the currently licensed thermal power (CLTP) of 2300 MWt. In the current amendment, FPL applied for an EPU of approximately 15% increase above the CLTP to 2644 MWt.

The plants are unique in terms of the number of shared systems and crossties that exist between the units. There are four high-pressure safety injection pumps, which all start on a safety injection signal from either unit. Two are required to mitigate a loss of coolant accident (LOCA) in one unit; only one is required for other design basis events in a unit. The pumps can be configured to supply either unit. There are three turbine driven auxiliary feed pumps that can be configured to supply either unit; any one pump can supply enough flow for the heat removal requirements of both units. In addition, there is a diesel-driven standby feedwater pump that provides enough flow for both units. There are four emergency diesel generators (EDGs), two for each unit, which all start on a safety injection signal from either unit. The EDGs can be remotely cross tied from the control room. Each EDG has sufficient capacity to maintain both units in a safe shutdown condition. The additional redundancy provided by this equipment and the crossties results in a core damage frequency (CDF) for PTN that is lower than for most current operating Westinghouse PWRs.

The reactor coolant system and reactor vessel internals will remain the same as pre-EPU. The uprated core will use a Westinghouse 15x15 upgraded fuel design. This fuel is currently approved for use in PTN. However, transition cores will also utilize another fuel design, the 15x15 seven-grid debris resistant fuel assembly (DRFA) design. The higher power level is achieved by increasing the average enrichment of fuel assemblies and the amount of new fuel in each reload. As a consequence, the temperature difference across the core and the operating reactor coolant average temperature both increase.

The licensee made numerous modifications to support the safety analysis at the higher power and undertook major plant modifications to support generation at the higher power level. A number of changes to the safety analysis methods were made to assess the effect of the EPU conditions on the safety analyses including the use of the ASTRUM statistical methodology for best-estimate analyses of large break LOCAs. The analyses in the original application for the power uprate did not address the effects of the decrease in fuel thermal conductivity with burnup identified in Information Notices 2009-23 and 2011-21. In response to Requests for Additional Information from the staff, FPL performed analyses to assess the impact of this decrease and proposed new requirements on operational limits and reload safety analyses to ensure all acceptance criteria can be met.

## DISCUSSION

### Safety Analysis Results

Core design parameters, such as the hot channel enthalpy rise factor and the heat flux hot channel factor, will be reduced to offset the effect of the EPU and maintain margins to fuel design limits. These reductions will be achieved using normal fuel management methods such as adjustment of enrichment, feed batch size, core loading pattern, and burnable absorber placement.

The licensee will also make a number of changes to setpoints and modify some components to restore margin. For example, the pressurizer backup heater actuation on a pressurizer high level deviation signal will be removed to improve the plant response to the loss of normal feedwater and loss of normal alternating current events. New fast acting feedwater isolation valves will be added just upstream of the current feedwater main and bypass flow control valves to perform the backup automatic feedwater isolation function. To ensure that each of the three turbine-driven auxiliary feedwater pumps will meet the increased minimum flow requirements at EPU conditions, they will be refurbished and modified.

The LAR provided analyses of transients such as decrease (loss) in reactor coolant system (RCS) flow, reduced secondary cooling, and overcooling using approved codes. These analyses indicate that the safety criteria for departure from nucleate boiling ratio (DNBR), RCS pressure, fuel linear heat generation rate, and pressurizer levels are met. The analyses were based on conservative input conditions.

The LAR also considered reactivity addition events. The fuel enthalpy results were below the safety criterion of 200 cal/gm. As we have noted previously, this criterion should be reviewed, since the available data suggest it should be set at a lower value. The licensee's analysis methodology of reactivity addition events yields conservatively high calculated values. The staff finds such an approach acceptable since they consider that the conservatism in the analysis compensates for the non-conservatism in the acceptance criterion. Although we would prefer to have a more realistic analysis and acceptance criterion, we agree that this is an acceptable approach to provide reasonable assurance that reactivity addition events are adequately addressed.

The LAR provided analyses of large and small break LOCAs for EPU conditions. Large break LOCAs were analyzed using an approved best estimate approach with the WCOBRA/TRAC code, the ASTRUM statistical methodology, and the PAD 4.0 fuel rod performance code. The containment response was calculated using GOTHIC. The results in the LAR indicate adequate margins to the acceptance criteria for the large break LOCAs. Small break LOCAs were analyzed with the approved NOTRUMP code. The results in the LAR indicate that the margins continue to be adequate for EPU conditions.

FPL has an ongoing program to resolve issues related to sump screen blockage and downstream effects due to LOCA debris (GSI-191). NRC staff evaluations of GSI-191 are ongoing for many PWRs including PTN, Units 3 and 4. The operating fleet of PWRs has been allowed to continue operations based on the current state of resolution, while the NRC staff and the PWR licensees continue their work to resolve any remaining open items. All future evaluations related to GSI-191 at PTN, Units 3 and 4, will consider both the current and EPU conditions.

The PTN, Units 3 and 4, emergency core cooling system (ECCS) uses high head safety injection (HHSI) flow into the cold legs during the injection phase. Following drainage of the refueling water storage tank (RWST), the source for the HHSI flow is switched to the sump. To avoid boric acid precipitation, the HHSI must be realigned to hot leg injection to flush the core and reduce the boric acid concentration before the precipitation limit of boric acid is reached. Analyses must be utilized to determine the time available for this switch.

Boric acid precipitation during long-term cooling, especially for hot leg breaks, is difficult to model. FPL presented WCOBRA/TRAC calculations of the concentration of boric acid in the core. The staff conducted independent analyses that used a more conservative value than that used by FPL for the fraction of the core generated vapor that is condensed and returned to the sump to reduce the sump boric acid concentration. The staff analysis showed a reduction in the time available to switch to hot leg injection to flush out the boron. When the licensee redid the calculations using the staff's value for the condensation of core generated vapor, they obtained a value for the time available (6.7 hours) very close to that obtained by the staff (6.5 hours). The emergency operating procedures will require the switch to be performed at 5.5 hours to provide additional margin against boric acid precipitation.

During the realignment, ECCS injection into the RCS is terminated. With no injection into the RCS at 5.5 hours post-LOCA for the limiting large break LOCA using Appendix K assumptions, analysis shows that the core would begin to uncover in 3.8 minutes. However, using a more realistic decay heat, FPL showed that the core would begin to uncover after 7 minutes if injection is not reinstated, with the peak cladding temperature (PCT) reaching 2200 °F after an additional 20 minutes. Thus, it would take 28 minutes to reach 2200 °F after terminating injection. These calculations do not take into account the effect of fuel thermal conductivity degradation. Hence, they may be somewhat optimistic, but they are adequate to show that there is considerably more margin than shown by the conservative Appendix K calculation. Operators are required to perform the realignment in less than 3 minutes. During simulator training exercises, FPL demonstrated that operators could perform the switch in 2 minutes. FPL also confirmed that the operators would be tested to assure that the operator action time of less than 2 minutes for realignment of HHSI would be maintained and verified as part of the operator qualification and training program.

Both staff and licensee calculations credit lower plenum mixing in the calculations of boric acid concentration, which might not be conservative, but they also neglect boron carry out of the break, which is very conservative. The analyses also ignore the effect of containment thermal-hydraulic conditions on RCS pressure above atmospheric pressure and use the boric acid solubility limit at 218°F (boiling point of saturated boric acid solution at atmospheric conditions) to determine the onset of precipitation, which is also conservative. While it is difficult to estimate the levels of conservatism introduced by the various assumptions, for this particular EPU application, based on the conservatism in the analyses and the margins for the times involved in the switch to hot leg injection, we agree with the staff that the issue of boric acid precipitation has been adequately addressed.

The safety analyses in the LAR did not address the impact of the thermal conductivity degradation (TCD) in fuel with burnup. The currently approved Westinghouse PAD 4.0 fuel rod performance code, which was used for these analyses does not account for TCD with burnup. Westinghouse is currently working on a formal revision to the PAD code. As an interim solution, FPL reanalyzed a number of accidents with a modified version of PAD 4.0, PAD4TCD, that accounts for TCD to compute fuel temperatures for use in fuel thermal-mechanical design analyses and downstream safety analyses. FPL has also committed to implement the formal revision to PAD when it becomes available.

Fuel temperature has a significant impact on fission gas release. However, FPL states that the approved PAD 4.0 fission gas release model already implicitly includes the effects of TCD because the model was calibrated to best fit measured data over an extensive empirical database including high burnup fuel rods.

PAD4TCD fuel centerline predictions were compared against Halden test reactor high burnup fuel temperature measurements including Instrumented Fuel Assembly (IFA)-432, IFA-562, IFA-515, IFA-533, and IFA-597.2. The PAD4TCD predictions are in much better agreement with the data than the PAD 4.0 predictions. The burnup and temperature dependence of the PAD4TCD thermal conductivity model were compared by the staff to the model used in the NRC-developed FRAPCON 3.4 code. There is good agreement between the two models.

In addition to comparing the thermal conductivity models in PAD4TCD and FRAPCON-3.4, the staff compared design calculations for fuel rod internal pressure, overpower fuel centerline melt, and overpower cladding strain. The benchmark calculations show reasonable agreement between the predictions of the two codes for end of life rod internal pressure. The PAD4TCD results are slightly higher. As expected, the thermal conductivity degradation results in higher predicted internal pressure, but all design criteria were still met. There is also reasonable agreement between the predicted power-to-melt limits. Overpower cladding strains resulting from anticipated operating occurrences (AOOs) were recalculated with PAD4TCD. The predicted cladding strain increased for all cases, but still met the cladding strain limit. The staff performed an AOO overpower cladding strain design calculation using FRAPCON-3.4. The FRAPCON-3.4 calculation started with a different rod power history which resulted in differences in initial conditions (e.g., gap size). In addition, although the change in local power was the same in the two calculations, the peak local power was higher in the FRAPCON calculation. The staff then ran a series of cases, sampling among the distribution of fuel manufacturing tolerances. A 95/95 upper tolerance limit was calculated from the distribution of resulting cladding strains. The FRAPCON predicted strain is higher than that predicted by PAD4TCD, but still met the strain limit.

A generic resolution of the incorporation of the effect of TCD for plants using Westinghouse safety analysis methodologies is needed. However, based on the comparison of the PAD4TCD predictions against Halden fuel temperature measurements, the comparison of PAD4TCD and FRAPCON-3.4 thermal conductivity models, and the comparison of PAD4TCD design calculations against benchmark FRAPCON-3.4 design calculations, the staff finds the PAD4TCD fuel performance code model acceptable for this application. We concur with the staff’s conclusion.

FPL reviewed the design calculations and safety analyses that were performed to assess the effects of the changes in the PAD code. Many of these analyses could be screened out because the analyses included assumptions that remained bounding even when the changes in thermal conductivity were considered. Additional analyses were performed for those cases that could not be screened out.

The most challenging case was the large break LOCA. FPL had to introduce new operating restrictions to ensure that the LOCA acceptance criteria were met, although the peak clad temperature and clad oxidation are still higher than in the original analyses. The new restrictions include a decrease in the allowable power peaking factors, a decrease in the HHSI delay time, an increase in the vessel minimum average temperature, and a decrease in the allowable steam generator tube plugging limit. It is now also necessary to introduce a burnup dependence for the allowable linear heat generation rate to preclude fuel centerline melt.

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-accelerated corrosion (FAC), and flow-induced vibration.

As part of its license renewal process, FPL addressed aging management of reactor vessel internals. The EPU conditions will increase the susceptibility of these internals to IASCC, but since the aging management will still be consistent with NUREG-1801, and PTN will continue to participate in the industry Materials Reliability Program/Issues Task Group efforts on reactor internals, it should still be adequate to manage aging.

Most of the materials in the PTN, Units 3 and 4, reactor pressure vessels will have acceptable upper shelf energies through the end of the operating license. For those welds that do not meet the 50 ft-lb limit, an equivalent-margins-against-ductile-fracture analysis was performed using a methodology acceptable to the NRC staff. FPL has accounted for the changes in neutron fluence due to EPU conditions on the Pressure-Temperature (P-T) Limits using an approved methodology for P-T Limit development. The staff performed confirmatory calculations and concluded that the licensee's P-T Limits meet the requirements of 10 CFR Part 50, Appendix G. FPL also concluded and the NRC staff confirmed that the reference temperature pressurized thermal shock (RTPTS) values will continue to meet the screening criteria of 10 CFR 50.61 through the period of licensed operation.

In PWRs, the prevalent materials degradation mechanism affecting nickel-base alloys is primary water stress-corrosion cracking (PWSCC). At PTN, the Unit 3 and Unit 4 reactor vessel closure heads were replaced with new heads with Alloy 690/52/152 penetrations in 2004 and 2005, respectively. Laboratory and field experience to date shows that Alloy 690 and its associated 52/152 welds are much more resistant to PWSCC than Alloy 600/82/182 material. In addition, NRC inspection requirements will help provide reasonable assurance of the structural integrity of the heads.

The Alloy 600 TT tubes in the PTN, Units 3 and 4, steam generators are not as resistant to PWSCC as Alloy 690 tubes, but have performed well at PTN and at other units at temperatures similar to those expected in PTN after the power uprate. Inspections under the NEI 97-06 steam generator program provide additional assurance that any potential increase in susceptibility can be adequately managed.

The only other components using Alloy 600/82/182 at PTN are the bottom-mounted instrument nozzles, the reactor vessel clevis inserts and lock keys, and the steam generator channel divider plate welds. These components will see small increases in temperature, which will increase susceptibility to PWSCC. However, these components are subject to inspection through ASME Code Case N-770, and these small temperature changes do not affect the inspection requirements of the Code Case.

The increased temperatures in the hot leg could lead to increased thermal aging embrittlement of cast austenitic stainless steels. This issue was analyzed during license renewal. The analysis remains valid under EPU conditions, because the temperatures in the hot leg will remain lower than that assumed in the evaluations performed for license renewal. FPL also validated the applicability of the environmental fatigue evaluation prepared for license renewal for the new EPU operating parameters.

The power uprate will result in higher velocities and temperature changes which could affect FAC. FPL analyzed piping systems under EPU conditions using the CHECWORKS code and stated that no changes to the existing FAC program will be required as a result of the EPU. FPL provided the staff a sample list of components for which wall thinning was predicted and measured by ultrasonic testing, or another approved method, to provide a comparison between actual wall thickness of a component and the predicted wall thickness by the CHECWORKS program. The staff found that the CHECWORKS program provides adequate conservatism between predicted remaining wall thickness and measured wall thickness, and thus there is reasonable assurance that the program will continue to be acceptable after the implementation of the EPU.

Flow-Induced Vibration

Under the current operating conditions, the steam generators have shown good performance with respect to wear and 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. The fluid kinetic energy in the U-bends associated with the EPU conditions is within the range associated with comparable steam generators in other reactors. Other plants with Westinghouse steam generators have shown good performance with respect to wear and vibration after power uprates. Inspections under the NEI 97-06 steam generator program provide additional assurance that any potential increase in susceptibility can be adequately managed.

FPL will establish a Piping and Equipment Vibration Monitoring Program to ensure that any steady state flow-induced piping vibrations following EPU implementation are not detrimental to the plant, piping, pipe supports, or connected equipment. The methodology for these evaluations will be in accordance with Part 3 of the ASME OM Code, "Vibration Testing of Piping Systems." Compliance with the Code requirements provides adequate assurance that excessive vibration will not lead to degradation and failure of the piping.

Risk Evaluations

Although this is not a risk-informed LAR, 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 examined impacts on initiating event frequency, event sequence models, systems models, failure and maintenance data, and human response. Both the human reliability analysis (HRA) and the system models were considered.

The higher EPU decay heat levels result in an increased need for safety injection following small and medium break LOCAs. Although no equipment modifications were needed, EPU LOCA conditions require HHSI from two pumps. Current conditions require only a single injection pump.

The greatest impact on risk is associated with human performance in light of reduced available times for response because of the higher decay heat levels associated with the EPU. To offset the impact of EPU conditions on operator response, two changes were made to the Emergency Operating Procedures (EOPs): the steam generator level setpoint for initiation of feed and bleed cooling was changed from 22% wide range to 33% wide range; and the step to shut off residual heat removal pumps in the event of a LOCA where pressure remains high was moved to an earlier point in the EOP.

The calculated risk increase associated with the EPU is well below Regulatory Guide 1.174 thresholds. More important than the quantitative assessment is the examination of the changes in available operator response times and implementation of the procedural modifications described above.

Electrical Systems Impacts

The main generator electrical output of each unit will increase by approximately 130 MWe. Each main generator will be rerated from 894 megavolt amperes (MVA) to 1032 MVA with an allowable power factor of 0.85. To support unit operation at EPU conditions, the required modifications include rewinding the stator, a new replacement rotor, and new current transformers. In addition, more efficient hydrogen coolers and exciter air coolers will be installed.

A system stability study was performed to evaluate the impact of the increased power output to the FPL transmission system. The reactive capability analysis concluded that the units' increased turbine-generator output meets the system reactive power requirements, and the short circuit analysis concluded that the fault current levels did not exceed the rating of any circuit breakers. However, in view of uncertainties in the analysis, two new inductors will be installed in the switchyard to reduce the available fault current. Power system stabilizers will also be installed to improve oscillation damping.

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. The procedure requires holds to gather and evaluate plant data after each 3% increase in power above the current licensed power level.

Transient tests include a turbine overspeed trip; 10% ramp load changes; turbine stop valve, governor valve, and intercept valve tests; and steam generator level / feedwater flow dynamic testing. The piping and equipment within the scope of the vibration monitoring program will be observed at several different plant operating conditions.

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, will be used in place of large transient testing. The staff has accepted this position and we concur.

<u>Summary</u>

We agree with the staff's reasonable assurance determination that the health and safety of the public will not be endangered by the licensee's operation at the proposed EPU power level and that such activities will be conducted in compliance with the Commission's regulations. 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 EPU license amendment request for PTN, Units 3 and 4, should be approved.

Sincerely,

***/RA/***

J. Sam Armijo
Chairman

**REFERENCES**


1. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Extended Power Uprate Application by Florida Power and Light Company Turkey Point Plant, Unit Nos. 3 and 4, Docket Nos. 50-250 and 50-251, provided to the ACRS on February 21, 2012 (ML11293A366)

2. FPL Letter, "License Amendment Request Regarding Extended Power Uprate (LAR 205)," dated October 21, 2010 (ML103560169)

3. FPL Letter, "Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Thermal Conductivity Degradation," dated December 31, 2011 (ML12009A113)

4. FPL Letters, "Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Thermal Conductivity Degradation," dated January 16, 2012 (ML12020A063 and ML12018A394)

5. ACRS Letter, "Point Beach Nuclear Plant, Units 1 and 2, Extended Power Uprate Application," dated March 23, 2011 (ML11073459)

Summary

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Letter to Mr. R.W. Borchardt, EDO, from J. Sam Armijo, ACRS Chairman, dated March 15, 2012

SUBJECT: FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE FLORIDA POWER AND LIGHT TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4, LICENSE AMENDMENT REQUEST FOR AN EXTENDED POWER UPRATE

**ML#12072A200**

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