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FROM: DUE: 07/11/12 EDO CONTROL: G20120441
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FINAL REPLY:

J. Sam Armijo, ACRS

TO:

Chairman Jaczko

FOR SIGNATURE OF : ** GRN ** CRC NO: 12-0296

Chairman Jaczko

DESC:

Final Safety Evaluation Report Associated with the Florida Power and Light St. Lucie, Unit 1, License Amendment Request for an Extended Power Uprate (EDATS: SECY-2012-0324)

ROUTING:

Borchardt
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Mamish
OGC/GC
McCree,RII
Kotzalas,OEDO

DATE: 06/22/12

ASSIGNED TO: CONTACT:

NRR Leeds

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Subject: Final Safety Evaluation Report Associated with the Florida Power and Light St. Lucie, Unit 1, License Amendment Request for an Extended Power Uprate

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OEDO Concurrence: YES

OCM Concurrence: NO

OCA Concurrence: NO

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

June 20, 2012

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

SUBJECT: FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE FLORIDA POWER AND LIGHT ST. LUCIE, UNIT 1, LICENSE AMENDMENT REQUEST FOR AN EXTENDED POWER UPRATE

Dear Chairman Jaczko:

During the 595th meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2012, we completed our review of the license amendment request (LAR) for the extended power uprate (EPU) of St. Lucie, Unit 1, (St. Lucie 1) and the associated NRC staff's Safety Evaluation Report (SER). Our Subcommittee on Power Uprates reviewed this matter in a meeting on April 26, 2012. During these reviews, we had the benefit of discussions with the representatives of the staff, Florida Power and Light Company (FPL or the applicant), and their consultants. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. The license amendment request for an extended power uprate of St. Lucie, Unit 1, should be approved subject to the conditions imposed in the Safety Evaluation Report.

2. The staff's calculations to confirm the applicant's RODEX-3A fuel performance results should be discussed in the SER.

BACKGROUND

The two unit St. Lucie Nuclear Power Plant is located on Hutchinson Island, near Ft. Pierce, Florida in St. Lucie County. Unit 1 is a 2x4 loop Combustion Engineering (CE) designed pressurized water reactor (PWR), originally licensed in 1976 to operate at 2560 MWt. In 1981, the unit was approved for a 5.5% stretch uprate to the currently licensed thermal power (CLTP) of 2700 MWt. In the current amendment, FPL requests approval of an extended power uprate of 10% and a measurement uncertainty recapture uprate of 1.7% to allow a core power level of 3020 MWt.

The reactor coolant system and reactor vessel internals will remain the same. The fuel will be of the AREVA CE14 High Thermal Performance design utilizing both the standard and MONOBLOC™ guide tubes. Enrichment will increase to a maximum planar average of 4.6 weight percent ^{235}U. The total integrated radial peaking factor will be reduced from 1.70 to 1.65, and the linear heat rate will decrease from 15.0 kW/ft to 14.7 kW/ft. The core temperature difference will rise from 50.4°F to 55°F, and the coolant average temperature will increase from 574.2°F to 578.5°F.

St. Lucie 1 has two steam generators (SGs), the originals were replaced in 1998 with B&W International Series 67 SGs. Each SG has 8,523 thermally treated Alloy 690 tubes with outside diameters of 0.75 inch and a nominal wall thickness of 0.045 inch. The volumetric steam flow in the U-bend entrance of the SGs will increase from 657 ft^3/s to 722 ft^3/s at EPU conditions.

The reactor vessel closure head was replaced in 2005. The reactor vessel head temperature will increase from about 594°F to 603°F.

Safety-related changes made for the LAR include increases in safety injection tank design pressure and hot leg injection flow, upgrades to the main steam isolation valves, additions of neutron absorption materials to spent fuel pool storage racks, installation of leading edge flow measurement systems to decrease flow measurement uncertainty, modifications to nuclear steam supply system setpoints, upgrades to radiation shielding for electrical equipment, modifications to safety related piping supports, and a higher reactor protection SG low-level trip setpoint.

DISCUSSION

Safety Analysis Results

The St. Lucie 1 EPU will result in higher core inventories of radionuclides, higher core inlet and outlet temperatures, larger temperature differences across the core, and higher core stored energy.

The applicant provided analyses of non loss-of-coolant-accident (LOCA) transients using approved codes and correlations. The transients considered include decreased reactor coolant system (RCS) flow, reduced secondary cooling, and RCS overcooling. The results met the acceptance criteria associated with departure from nucleate boiling ratio (DNBR), RCS pressure, fuel linear heat generation rate (LHGR), and pressurizer fill levels.

For non-LOCA reactivity addition events, the acceptance criteria were also met but the computational methods and the modeling used by the applicant, for example, for the control element withdrawal at power event, while acceptable, are not consistent with the state of the art. The movement of a control element in a reactor will introduce a significant space-time variation in the neutron flux and therefore substantial spatial changes in the power shape during such

postulated events. Current best practice is to calculate this space-time flux shape by simulating the event with a spatial kinetics code and then use the resulting dynamic power shape in the core and the corresponding temperature/fluid conditions as inputs to a subchannel code in order to estimate the critical heat flux and the minimum DNBR. The analysis performed for the St. Lucie 1 EPU separates the spatial and time variation of the flux into a time varying power amplitude function and a time invariant power shape function. It is difficult to establish that such approximations, though usually conservative, guarantee conservatism for all reactivity events. Nonetheless, the variable high power reactor trip function incorporated in the design would be effective before departure from nucleate boiling could occur. The generally conservative analysis and the variable high power trip are sufficient to establish that St. Lucie 1 will meet the safety acceptance criteria. Amongst these, the fuel enthalpy criterion of less than 230 cal/gm should be revised, something we have recommended in the past, as the available data suggest that it should be set at a lower value. As the applicant's analyses of reactivity addition events is likely to result in conservatively high calculated values for the acceptance criteria, we concur with the staff and find the results acceptable.

Large break LOCAs were analyzed for the EPU in accordance with the methods in EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors." These methods were approved with some conditions and limitations. The results indicate substantial margins to the acceptance criteria for large break LOCAs. The calculations used a thermal model for the fuel, RODEX-3A, that does not account for fuel thermal conductivity degradation (TCD) at high burnup. This fuel TCD is a generic issue recently identified in NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation." To correct for this non-conservatism in the large break LOCA calculations, the applicant applied an empirical correction that biased the calculated fuel centerline temperature to fit the temperature measurements from high burnup experiments done in the Halden reactor. Calculations using the NRC fuel code, FRAPCON, which have been assessed against the Halden experiments, were also done to confirm that the applicant's RODEX-3A analyses were satisfactory. These confirmatory calculations and comparisons with the applicant's RODEX-3A results are not, but should be, discussed in the SER.

The staff is currently reviewing a revised version of the EMF-2103 report. Among other improvements, it uses COPERNIC to generate fuel initial conditions. COPERNIC, a more modern fuel code, has been assessed against the Halden data. If the staff's review of this revision of the EMF-2103 report identifies significant issues with the currently approved version, including its use of the empirically corrected results from RODEX-3A, licensees will be notified. Licensees will then be required to estimate the effects of errors in accordance with 10 CFR 50.46(a)(3). Because 10 CFR 50.46 already requires estimation of the effects of errors, the staff did not impose a license condition on the approval of the St. Lucie 1 LAR concerning upgrading from RODEX-3A (which is used for the large break LOCA calculations). The staff will brief the ACRS on their review of the revised EMF-2103 report, including any issues that might arise from the version of RODEX-3A used in this LAR.

In view of the large margins to the acceptance criteria for large break LOCAs indicated by the applicant's analyses, the account taken of the TCD effect on the analyses and the staff's confirmatory calculations using FRAPCON, and the continuing staff review of the large break LOCA analyses methods which will require correction of any errors identified, we concur with the staff and find the applicant's large break LOCA results acceptable.

The applicant's small break LOCA analyses for the EPU was conducted using the S-RELAP5 based evaluation model and methods in the approved report, EMF-2328(P)(A) Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." The staff requested additional analyses to evaluate the effects of break spectrum, loop seal clearing, safety injection line break, and delayed reactor coolant pump trip. In all cases the EPU analyses results for peak clad temperature, maximum local oxidation, and maximum total core-wide oxidation retained substantial margins to the safety criteria in 10 CFR 50.46.

Though the staff accepted the applicant's safety analyses procedures, a license condition was imposed to account for the evolving treatment of TCD. This license condition requires the applicant to demonstrate that the St. Lucie 1 safety analysis remains conservatively bounded when compared to those done with an approved version of a generic supplement to address TCD in the fuel code RODEX2. The supplement is currently under NRC review. The license condition also requires the applicant to provide a schedule for re-analysis using the NRC-approved generic supplement for any affected licensing basis analysis.

The Halden data on fuel thermal conductivity for high fuel centerline temperatures at high burnups are limited. The trends in the available data suggest that the effect of burnup on fuel thermal conductivity appears to decrease at higher temperatures. Nonetheless, the continuing staff evaluations of the impact of TCD on safety analyses should carefully estimate the uncertainties that arise if TCD values are extrapolated beyond the existing database.

The applicant's emergency core cooling system (ECCS) net positive suction head analysis does not credit containment accident pressure.

The applicant 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 St. Lucie 1. 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 St. Lucie 1 will consider both the current and EPU conditions.

The staff reviewed and accepted the applicant's boric acid precipitation analyses for long term cooling following LOCAs. There were several conservatisms in these analyses to determine whether maximum concentrations could be maintained below solubility limits. In particular, the applicant's analyses and the staff's confirmatory calculations indicate that sufficient time exists for operator actions to align the ECCS for hot leg injection to prevent precipitation. The LAR is in compliance with the requirements in 10 CFR 50.46 and Appendix K in this regard.

<u>Materials Effects</u>

The power uprate will result in increased fast neutron flux and temperature within the reactor vessel as well as higher temperature and flow velocity in portions of the secondary system. These changes can increase the oxidation potential of the reactor coolant and the rate of irradiation hardening of core materials, and accelerate material degradation rates. The applicant has evaluated relevant material 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.

The applicant evaluated the effect of the increased fluence on reactor vessel embrittlement. All the materials in the reactor pressure vessel will have acceptable upper shelf energies through the end of the operating license. The applicant also concluded and the NRC staff confirmed, with support from some independent calculations, that the reference temperature pressurized thermal shock (RT_{PTS}) values will continue to meet the screening criteria of 10 CFR 50.61 through the period of licensed operation.

In PWRs, the prevalent material degradation mechanism affecting nickel-base alloys is primary water stress corrosion cracking (PWSCC). At St. Lucie 1, the reactor vessel closure head and pressurizer have been replaced with new components with Alloy 690/52/152 penetrations and the SG tubing with Alloy 690, which are much more resistant to PWSCC than Alloy 600/82/182. Inspections under the NEI 97-06 steam generator program and NRC inspection requirements for the reactor vessel heads provide additional assurance that any potential increase in susceptibility due to the EPU can be adequately managed.

The increased temperatures could lead to increased thermal aging embrittlement of cast austenitic stainless steels. Most of the cast materials in St. Lucie 1 have acceptable lower bound thermal aging fracture toughness values. For some materials, data on ferrite level and composition are not available and as part of its license renewal application St. Lucie 1 has committed to develop an aging management program for these materials. This program will account for any higher temperatures associated with EPU conditions.

The applicant currently uses the CHECWORKS™ code to monitor FAC. Data were provided to show that measurements by ultrasonic or radiographic techniques confirm that the code predictions are generally conservative. In most cases, only small changes in wear rates are predicted for EPU conditions. In a few locations, flow velocities are predicted to exceed industry guidelines. In these cases, components will be replaced with more resistant materials, modified to reduce velocities, or evaluated further. We concur with the NRC staff's conclusion that the applicant has adequately addressed the impact of changes in plant operating conditions on the FAC analysis.

Flow-Induced Vibration

The two St. Lucie 1 inverted U tube steam generators have performed well in the 14 years since they were installed. Of the 8,523 tubes in each SG, 14 tubes are plugged in one SG and only one tube is plugged in the other SG. Tube wear at supports in the U-bend region has been detected in only 27 tubes and wear due to tube-to-tube contact has not been observed.

The applicant analyzed the potential for increased flow-induced vibration under EPU conditions which would increase volumetric flow and kinetic energy in the SG U-bend region to values about 5% higher than those in comparable plants at Millstone 2 and Calvert Cliffs 1 and 2. These analyses considered initiation of fluid elastic instability which could result in accelerated wear and possible tube-to-tube contact. The results show only a slight increase in projected wear rates at tube supports, and a large margin between operating EPU conditions and those which might initiate fluid elastic instability.

Tube inspections will be performed at the first refueling outage which is less than a full cycle following implementation of the EPU. Therefore, if EPU conditions were to result in an unanticipated increase in tube wear rates, it would be detected less than a year into EPU operation. Based on these inspections, the good experience with the replacement SGs, and the projected margins for initiation of fluid elastic instability, we agree with the staff that EPU effects on SG tube integrity should be acceptable.

Risk Evaluations

Although the licensing application is not risk-informed, the applicant performed a quantitative assessment of the change in core damage frequency (CDF) and large early release frequency (LERF) associated with internal events and a qualitative evaluation for external events. The applicant set an EPU objective of maintaining or improving overall plant reliability and risk. Several physical plant and operational changes will be made to this end. For example, the SG low-level reactor trip setpoint will be raised from 20.5% to 35% to provide greater inventory for total loss of feedwater flow events, and plant procedures will also be changed to enhance mitigation. Comparing current plant configuration and procedures with the EPU plant configuration, the internal events CDF is predicted to decrease by 3.6E-07 per year and LERF is predicted to decrease by 8.2E-08 per year. For external events, the applicant found that no new fire, seismic, wind, or external flooding vulnerabilities will be introduced by the EPU. The time available for operator actions will decrease slightly, but the estimated impacts are small.

With regard to seismic events, representative analyses by the applicant indicate that the current and EPU estimates of CDF and LERF were both insignificant and remain essentially unchanged. The staff also developed an estimation of CDF using a seismic margins approach and the latest USGS seismic hazards information, and found the change in seismic risk for the EPU to be insignificant.

With regard to shutdown operations, the primary impact of the EPU is associated with the decrease in available times for operator response. Reductions in available time for operators to take compensatory or mitigating actions could vary from several to ten or more minutes, depending on the shutdown condition. The applicant concluded that the shorter available time under EPU conditions would not significantly impact safety.

In summary the applicant has evaluated the risk impact of the EPU. The applicant shows some reduction in the estimated CDF and LERF following the EPU. These changes are consistent with the guidance in Regulatory Guide 1.174.

Electrical Systems

Offsite power is transmitted to the plant switchyard by three physically independent 230 kilovolt (kV) transmission lines. The staff reviewed the applicant's assessment of the effects of the EPU on the offsite power system and concluded that following implementation of the modifications required to support the EPU, the offsite power system will continue to meet its current licensing basis. Adequate physical and electrical separation exists, and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The staff further concluded that the EPU would not degrade grid stability.

Several modifications will be made to the plant electrical systems to accommodate the EPU. Amongst these the main generator rating will be increased with the rotor and stator being rewound and the associated hydrogen cooling system will be modified. The increased rating of the main generator will be adequate to accommodate the EPU conditions. The existing generator step-up transformer design rating will be upgraded to support generator output at EPU conditions. Plant modifications will also be made to the safety-related portions of the 125 V DC System. Changes will move the power sources for isolated phase bus duct cooling fans and two vent fans from 480 V AC Motor Control Centers to 480 V AC load centers. These additional loads will increase requirements for 125 V DC control power to operate the 480 V AC load center circuit breakers. The applicant examined the battery loading and concluded that the 125 V DC System continues to have the capacity and capability to perform its function and remains within equipment ratings while maintaining adequate margin for battery capacity.

The staff has also reviewed the applicant's assessment of the effects of the EPU on the environmental qualification (EQ) of electrical equipment and found the applicant's treatment of EQ satisfactory.

The staff has concluded, and we agree, that the applicant has adequately addressed the effects of the EPU on electrical systems.

Station Blackout (SBO) Events

The applicant evaluated the impact of the EPU on the plant's ability to cope with and recover from a four-hour SBO. No AC power is assumed to be available for the first hour of the SBO event, and the alternate AC (AAC) source is assumed to be available for the subsequent three hours. The condensate inventory required for decay heat removal for the four-hour SBO at EPU conditions is bounded by 130,500 gallons, the volume required for plant cool down at a rate of 75 °F/hr. The minimum total contained condensate volume will be increased to 153,400 gallons to ensure sufficient cooling water for the EPU operating conditions. Additional systems and equipment were evaluated for SBO under EPU conditions including the 125 V DC Class 1E batteries, the compressed air system, and inside and outside of containment ventilation systems. The staff finds the EPU acceptable with respect to SBO.

Power Ascension Testing and Large Transient Testing

The applicant 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 once a steady state power of 10% to 15% of EPU rated power has been reached. With the majority of power being routed to the main condenser via the steam bypass control system (SBCS), the turbine will be accelerated until its speed causes an actuation of an overspeed trip. This test will verify the proper performance of the turbine overspeed trip function. It will verify proper operation of the turbine valves and verify expected plant performance subsequent to the turbine trip. Performance of plant control systems, such as steam bypass control valves, will be monitored in response to the transient.

No large transient testing is planned even though substantial modifications will be made on the secondary side, particularly to the feedwater systems. The main basis for not performing large transient tests relies on an analytical justification using the NRC approved computer code, CENTS, to evaluate plant responses to Condition I and II initiating events at EPU conditions. The CENTS code is acceptable for analyzing operational transients for CE designed PWRs and has been used on other CE designs. The staff has accepted this position and we concur.

SUMMARY

We agree with the staff's reasonable assurance determination that the health and safety of the public will not be endangered by the applicant's operation at the EPU power level and that such activities will be conducted in compliance with the Commission's regulations. Safety margins will be sufficient to ensure that the safety limits and acceptance criteria will not be challenged. The EPU license amendment request for St. Lucie 1 should be approved with the license conditions in the SER.

Sincerely,

/RA/

J. Sam Armijo
Chairman

REFERENCES

1. NRR SER Related to Extended Power Uprate Application by Florida Power and Light Company St. Lucie Unit 1, Docket Nos. 50-335, provided to the ACRS on April 30, 2012 (ML11326A109).

2. FPL Letter, Docket No. 50-335, Renewed License No. DPR-67, License Amendment Request for an Extended Power Uprate, FPL Letter No. L-2010-259. November 22, 2010 (ML103560429, ML103560415).

3. FPL Letter, Docket No. 50-335, Renewed License No. DPR-67, Information Regarding AREVA LOCA and Non-LOCA Methodologies Provided in Support of the St. Lucie Unit 1 License Amendment Request for Extended Power Uprate, FPL Letter No. L-2011-206. May 27, 2011 (ML11153A0482).

4. AREVA NP Inc. Report EMF-2328(P)(A), Supplement 1, Rev. 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2012 (ML12065A392).

5. AREVA NP Inc. Report EMF-2103(P)(A), Rev. 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003 (ML032691419).

6. NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," October 8, 2009 (ML091550527).

7. NRR Memorandum, "Audit Report – St. Lucie Nuclear Generating Plant, Unit 1 EPU Application Review," February 28, 2012 (ML12040A287).

8. NRR Memorandum, "Audit Report for AREVA Fuel Thermal-Mechanical Design Augmentation Factors," September 7, 2011 (ML11245A196).