June 9, 2006

Mr. Gene F. St. Pierre, Site Vice President
c/o James M. Peschel
Seabrook Station
PO Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - REVIEW OF REPORT RE: COMPLETION OF LICENSE CONDITION 2.K (TAC NO. MC8873)

Dear Mr. St. Pierre:

By letter dated November 7, 2005, as supplemented by letters dated January 11, 2006, and April 20, 2006, FPL Energy Seabrook, LLC (FPLE) submitted a revised analysis of the inadvertent actuation of the emergency core cooling system (ECCS) event for Nuclear Regulatory Commission (NRC) staff review. This revised analysis was submitted in response to NRC staff concerns captured in License Condition 2.K, which were identified during the review of FPLE's license amendment request 04-03 for Seabrook Station, Unit No. 1.

The NRC staff has completed its review of the submitted analysis and concludes that it adequately demonstrates that an inadvertent actuation of the ECCS would not fill the pressurizer prior to plant operators securing the event. Therefore, the NRC staff finds that the analysis submitted by FPLE meets the requirements described in License Condition 2.K. Details of the NRC staff's review are contained in the enclosed Safety Evaluation.

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosure:
Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REVIEW OF REPORT RE: COMPLETION OF LICENSE CONDITION 2.K

FACILITY OPERATING LICENSE NO. NPF-86

FPL ENERGY SEABROOK, LLC

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated November 7, 2005, as supplemented by letters dated January 11, 2006, and April 20, 2006, FPL Energy Seabrook, LLC (FPLE) submitted a revised analysis of the inadvertent actuation of the emergency core cooling system (ECCS) event for Nuclear Regulatory Commission (NRC) staff review. This revised analysis was submitted in response to NRC staff concerns (captured in License Condition 2.K), which were identified during the review of FPLE's license amendment request (LAR) 04-03 for Seabrook Station, Unit No. 1 (Seabrook).

Specifically, License Condition 2.K states the following:

> Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves to safety-grade status and confirm the safety-grade status and water-qualified capability of the pressurizer power operated relief valves, pressurizer power operated relief valve block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC-approved methodology, that concludes that the pressurizer does not become water-solid within the minimum allowable and verifiable time for operators to terminate the event.

In the interim, Seabrook has been operating under a justification for continued operation (JCO), according to the guidelines of Generic Letter (GL) 91-18, "Resolution of Degraded and Nonconforming Conditions." The JCO is supported by a revised analysis of the inadvertent actuation of the ECCS event, performed with best estimate assumptions, that predict the pressurizer will not become water-solid before the event can be terminated by Seabrook operators. This safety evaluation documents the NRC staff's review of said analysis.

2.0 REGULATORY EVALUATION

Inadvertent actuation of the ECCS is a Condition II event that is evaluated for the effects of adding water inventory to the reactor coolant system (RCS). This event could be caused by

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operator error or by a spurious, automatic actuating signal. Automatic actuation of the ECCS at Seabrook could be caused by a signal from the reactor protection system based on high containment pressure, lower pressurizer pressure, or low steamline pressure.

The NRC staff reviewed the sequence of events, the analytical model used for the analyses, the input parameters used in the analytical model, and the results of the transient analyses. The NRC staff's acceptance criteria are based on (1) Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A, General Design Criterion (GDC) 10, which requires that the RCS be designed with appropriate margin to ensure that design safety limits are not exceeded during normal operations, including anticipated operational occurrences (AOOs); (2) GDC 15, which requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCS pressure boundary are not exceeded during AOOs; and (3) GDC 26, which requires that a reactivity control system be provided and be capable of reliably controlling the rate of reactivity changes to ensure that design safety limits are not violated under normal conditions of operation including AOOs.

The acceptance criteria for a Condition II event, such as the inadvertent actuation of the ECCS event is a reactor trip with the plant being capable of returning to operation. A Condition II event may not propagate to cause a more serious fault, i.e., a Condition III or IV event. Additionally, it may not result in fuel rod failures or in overpresurization of the RCS or secondary system. Specific review criteria for the inadvertent actuation of ECCS event are contained in NUREG 0800, Standard Review Plan, Section 15.5.1-2.

3.0 TECHNICAL EVALUATION

When the ECCS is actuated, the reactor is tripped and borated water is pumped from the refueling water storage tank by the centrifugal charging pumps into the cold leg of each RCS loop. The safety injection pumps are also started; however, they cannot deliver flow until the shutoff head of these pumps is lower than the nominal RCS pressure. Since the initiating event also causes a reactor shut down, the inadvertent actuation of the ECCS is not expected to result in any fuel rod failures. Additionally, since the charging pumps have a shutoff head of the charging pumps of 2600 psia, they cannot pressurize the RCS to pressures exceeding 110% of RCS design pressure (2750 psia). The reactor trip also prevents the development of a significant power-load mismatch, such that the event will not result in an overpressurization of the secondary system.

The ECCS is, however, capable of filling the pressurizer and causing the discharge of water through the power-operated relief valves (PORVs) or the presurizer safety-relief valves (PSRVs). If these valves are not qualified for water relief, they are assumed, in accident analyses, to stick open and create a small-break loss-of-coolant-accident at the top of the pressurizer (a Condition III event). This would be a violation of the Condition II acceptance criterion, which prohibits a Condition II event from developing into a more serious (Condition III or IV) event.

As discussed previously, the possibility for the inadvertent actuation of the ECCS to cause this condition was identified during the NRC staff review of Seabrook LAR 04-03. During this review, the NRC staff identified that operator action to terminate the event was assumed to occur at approximately 10 minutes, when the pressurizer pressure was within about 22 psi of its opening setpoint. In this water-solid condition, there is no steam cushion available to control pressure and the RCS pressure is rising at a rapid rate. Given the assumed pressure measurement and PORV setpoint uncertainties, the licensee could not adequately assure that the PORVs would not open. Therefore, the NRC staff concluded that FPLE's analysis did not demonstrate that this particular Condition II event could not propagate to a more serious event.

Since the inadvertent actuation of the ECCS event is relatively insensitive to changes in rated power level, and thus insensitive to the stretch power uprate, the NRC staff agreed to consider an interim analysis, performed in accordance with the guidelines of GL 91-18, that provided a JCO until refueling outage 11.

In its revised analysis, FPLE indicated that the pressurizer would not fill before 14 minutes, which is the assumed length of time for the operators to take appropriate actions to terminate the injection of water. In the new analysis, FPLE credited the steam generator (SG) atmospheric relief valves (ARVs), assuming that they opened following the reactor/turbine trip to control RCS temperature. This is in contrast to the previous analysis that assumed steam relief through the SG safety valves (SVs), which have a higher relief setpoint than the ARVs. Because of the ample surface area available for heat transfer within the SG tubes and reduced heat input following a reactor trip, RCS hot leg temperature will trend towards the saturation temperature of the SG pressure. Thus, the revised analysis, which assumes the SG ARVs open at a lower pressure than the SG SVs, demonstrates a lower SG pressure, and in turn, a lower RCS temperature and a smaller volume of water surging into the pressurizer. Consequently, it would take longer to fill the pressurizer, allowing the operators time to secure the ECCS injection prior to going water-solid.

This revised analysis is based on the assumption that three of the four SG ARVs are available to relieve steam. The SG ARVs are listed as equipment required for safe shutdown in Table 7.4-1 of the Seabrook Updated Final Safety Analysis Report. As such, this equipment has redundant components, is safety grade, and is in compliance with applicable Institute of Electrical and Electronics Engineers Standards. Therefore, the NRC staff finds the assumption to be acceptable.

The SG ARVs can be operated automatically or manually, following a turbine trip, to maintain SG pressure at 1140 psia. This is approximately 45 to 85 psi lower than the pressure that would be maintained by the SG SVs. The SG ARVs, however, have a longer assumed stroke time than the SG SVs (up to approximately 70 seconds). The NRC staff identified this concern to FPLE and in its supplement dated April 20, 2006, FPLE demonstrated that the opening of the three SG ARVs could be delayed by as long as 8 minutes while still allowing the Seabrook operators adequate time to secure ECCS injection. Therefore, the NRC staff finds that the assumed stroke time for the SG ARVs is supportive of the revised analysis. Additionally, the NRC staff notes that this reanalysis is based on the assumption that all but one of the charging pumps are stopped by 9 minutes, with the last charging pump being stopped by 13 minutes. These response times were confirmed in Seabrook simulator time studies performed by FPLE in September and October of 2005.

4.0 CONCLUSION

The NRC staff has reviewed FPLE's revised analysis and finds that the reanalysis was performed using methods, assumptions, and operator time responses acceptable to the NRC staff. The results of this revised analysis indicate that the inadvertent ECCS actuation event would not fill the pressurizer before plant operators could terminate the event. Thus, the inadvertent ECCS actuation event is not expected to result in water relief through the pressurizer PORVs or PSRVs that could potentially lead to a more serious Condition III event. Therefore, the NRC staff finds that the reanalysis submitted by FPLE satisfies the requirements of License Condition 2.K.

Principal Contributor: S. Miranda

Date: June 9, 2006