Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

March 18, 2008

L-2008-060
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed License Amendment
Request for Additional Information Response
Alternative Source Term Amendment – TAC Nos. MD6173 and MD6202

On July 16, 2007, Florida Power and Light Company (FPL) submitted the St. Lucie Unit 1 and 2 Alternative Source Term (AST) license amendment requests via FPL letters L-2007-085 and L-2007-087. As a result of the submittals, the NRC requested additional information. This correspondence provides the FPL response to the NRC Request for Additional Information (RAI) received by letter dated February 21, 2008.

The no significant hazard analyses submitted with FPL letters L-2007-085 and L-2007-087 remain bounding. In accordance with 10 CFR 50.91(b)(1), a copy of the proposed amendment was forwarded to the State Designee for the State of Florida.

Please contact Ken Frehafer at 772-467-7748 if there are any questions about this submittal.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 18th day of March 2008.

Very truly yours,

Gordon L. Johnston
Site Vice President
St. Lucie Plant

GLJ/KWF

Attachment

cc: Mr. William A. Passetti, Florida Department of Health

A001
NRR

NRC Question 1:

In St. Lucie Plant Unit 2 (SLP2) Table 1.8.1-1 of NAI-1101-043, Rev. 2 attachment to Florida Power & Light Company letter dated July 16, 2007, there is a difference in the north and south control room (CR) intake heights of 0.6m (2.09 ft). However, in the second paragraph of §1.6.3.1 within the same attachment, there is no clear distinction of different heights for the north and south CR intake, but the term "approximately" is used.

> *"Outside air makeup is supplied through either of two outside air intakes located in the northern and southern walls of the Reactor Auxiliary Building at approximately elevation 78 feet."*

Considering the north and south CR intake heights are 19 feet above sea level, the north and south CR intake elevations of 78 feet (as noted in the above statement) adjust to 59 feet (18.0m). Please confirm that the north and south CR receptor heights are accurately defined as 18.2m and 17.6m, respectively (as in Table 1.8.1-1), resulting in a midpoint CR receptor height of 17.9m. If it is determined that these values are incorrectly noted, please indicate what affect using the incorrect receptor heights may have on the CR dose estimates.

(Note: Take into account that these receptor heights are used in the onsite (i.e., CR) atmospheric dispersion analyses and the resulting χ/Q values serve as input to the dose consequence assessment.

FPL Response to Question 1:

Review of the applicable plant drawing, 2998-G-873, has confirmed that the north and south control room receptor heights are accurately defined as 18.2 m and 17.6 m. respectively, resulting in a midpoint control room receptor height of 17.9m.

Note that NRC question 1 refers to NAI-1101-043, Rev. 2 as being applicable to St. Lucie Unit 2; however, NAI-1101-043, Rev. 2 is applicable to St. Lucie Unit 1. The corresponding document that is applicable to St. Lucie Unit 2 is NAI-1101-044, Rev. 2. The NRC properly cites the values from Table 1.8.1-1 of NAI-1101-044, Rev. 2 in this Unit 2-related question.

NRC Question 2:

The following statement is found in §1.6.3.2, "Emergency Operation," of NAI-1101-044, Rev. 2 attachment to letter dated July 16, 2007, for each SLP1 and SLP2:

> *"Outside air intake dampers are adjusted to allow sufficient outside air makeup flow to maintain control room pressurization. By observing the radiation monitors located in the outside air intake ducts, the operator restores outside air makeup by selecting*

which set of isolation valves to open. After determining which outside air intake has the least, or zero, amount of radiation, the operator opens the isolation valves on that intake and adjusts the system dampers for proper flow."

Please confirm that plant procedures for both SLP1 and SLP2 identifies the need for operators to stay aware of altering meteorological conditions throughout the 30-day event and how these changes may affect selection of the more favorable CR intake during emergency mode of operation and when initiating CR makeup air.

FPL Response to Question 2:

Current St. Lucie procedures do not include guidance for monitoring meteorological conditions throughout the event for potential impact on the selection of the more favorable control room outside air intake. The applicable procedures will be revised, as needed, to provide the necessary guidance.

NRC Question 3:

In NAI-1101-043, Rev. 2 attachment to letters dated July 16, 2007, for SLP1 and SLP2, Table 2.6-1 states that 0.5% of the fuel is assumed to experience fuel centerline melt. However, Assumption #6 of §2.6.3 of the same attachment states that "0.05%" of the fuel is assumed to experience fuel centerline melt. Please confirm which value was considered in the dose analysis of the control element assembly ejection accident at SLP1 and SLP2.

FPL Response to Question 3:

Table 2.6-1 which states that 0.5% of the fuel rods experience fuel centerline melt is correct. The text in assumption #6 of §2.6.3 which states that 0.05% of the fuel is assumed to experience fuel centerline melt is a typographical error. This error has been corrected in the NAI reports. The correct value of 0.5% was used in the AST analyses.

NRC Question 4:

RG 1.183, Regulatory Position 4.3, "Other Dose Consequences," states that:

"The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST [alternate source term]. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE [Total Effective Dose Equivalent]."

In evaluating the submittal, the staff could not determine if RG 1.183, Regulatory Position 4.3 had been assessed for SLP1/2. Please provide additional information describing how Regulatory Position 4.3 is assessed for SLP1/2.

FPL Response to Question 4:

Post-Accident Access Shielding (NUREG-0737, II.B.2)
St. Lucie performed plant shielding studies in accordance with NUREG-0737, Item II.B.2. For locations where the studies indicated that mitigation actions could not be reasonably performed within the applicable dose limits, plant modifications were completed to either eliminate the need to access these areas or to reduce local dose rates following an accident.

The source term used for the shielding analysis was based upon Regulatory Guide 1.4. The nuclide release fractions specified in the AST methodology outlined in Regulatory Guide 1.183 differ somewhat from those of Reg. Guide 1.4. The difference in the release fractions has the potential to affect the dose rates in vital areas where piping containing RCS fluid is located. However, as discussed in Generic Issue 187, a study conducted by Sandia National Labs, "Evaluation of Radiological Consequences of Design Basis Accidents at Operating Reactors Using Revised Source Term", dated September 28, 1998, found that the traditional source term methodology and the AST methodology produced similar integrated doses. This report showed that the integrated AST doses did not exceed those from TID-14844 until 42 days after the event for a PWR. As such, the differences in the release fractions associated with the AST methodology would have little impact on the local dose rates during the 30-day post-LOCA mission time. Since the local dose rates are not expected to be significantly impacted by AST during the first 30-days following a LOCA, the conclusions of the shielding study would not significantly change by expressing the mission doses in terms of TEDE.

The St. Lucie Unit 1 Control Room and Technical Support Center (TSC) share the same HVAC envelope. Since control room doses were calculated as part of the revised radiological analysis using AST, the shielding study consequences for these areas have been re-evaluated in terms of TEDE.

Post-Accident Sampling Capability (NUREG-0737, II.B.3)
Technical Specification Amendments 174 and 114 eliminated the requirements to have and maintain the Post-Accident Sampling System for Units 1 and 2, respectively.

Accident Monitoring Instrumentation (NUREG-0737, II.F.1)
Post-accident monitoring instrumentation is available to measure various plant parameters. This instrumentation is designed to detect and remain operable under postulated design basis conditions. Although the more realistic AST source term would potentially involve a larger dose to equipment exposed to sump water over long periods of time, the conclusions of Generic Issue 187, which specifically addresses this issue, stated that there would be no discernable risk reduction associated with modifying the design basis for equipment qualification to adopt the AST. Therefore, the ability of the accident monitoring

instrumentation to meet the NUREG-0737 requirements is not impacted by the revised radiological analyses.

Leakage Control (NUREG-0737, III.D.1.1)
Section 6.8.4.a of the both units' Technical Specifications establishes a program to reduce leakage from primary coolant sources outside of containment. Appropriate values for ESF system leakage were incorporated into the revised radiological dose analyses performed using the AST methodology.

Emergency Response Facilities (NUREG-0737, III.A.1.2)
Emergency response facilities are shared by the two units. Provisions of NUREG-0737 regarding emergency response facilities were addressed by ensuring ventilation system isolation, filtering, and monitoring of the TSC. As part of the Control Room envelope, the dose consequences in the TSC have been evaluated using the AST methodology. For the Operational Support Center (OSC), plans were established for the evacuation and relocation if conditions became untenable. These provisions are unaffected by the implementation of AST. The Emergency Operations Facility (EOF) is located outside of the 10 mile Emergency Planning Zone (EPZ).

Control Room Habitability (NUREG-0737, III.D.3.4)
Radiological analyses were performed to meet the requirements of item III.D.3.4 of NUREG-0737. These analyses ensure that control room operators are adequately protected against the effects of accidental release of radioactive gases and that the plant can be safely operated or shutdown under design basis accident conditions. As part of the revised radiological analyses using AST, the control room habitability consequences due to the release of radioactive gases were re-evaluated using TEDE dose conversion factors and acceptance criteria.

NRC Question 5:

In §2.3.4 of both NAI-1101-043 and NAI-1101-044, the following statement is made:

> *"Allowable levels of fuel failure for DNB [departure from nucleate boiling] and fuel centerline melt are determined for both the MSLB [main steam line break] outside of containment and the MSLB inside of containment. These allowable fractions are based on the dose limits specified in Table 6 of RG [Regulatory Guide] 1.183. The activity released from the fuel that is assumed to experience DNB is based on Regulatory Positions 3.1, 3.2, and Table 3 of RG 1.183. The activity released from the fuel that is assumed to experience fuel centerline melt is based on Regulatory Position 1 of Appendix H to RG 1.183."*

The staff could not determine the relationship between the allowable fractions discussed and the dose limits specified in Table 6 of RG 1.183. Please provide additional information describing the basis for the fractions of fuel damage assumed in the MSLB

analyses for SLP 1/2 and the relationship, if any, to the dose limits specified in Table 6 of RG 1.183.

FPL Response to Question 5:

Allowable fuel failures define the limits that the safety analyses must meet in terms of fuel failures. The analyzed fuel failure values used in the AST dose analyses do not represent values that are indicative of those that would be predicted by the core reload analyses.

The analyzed fuel failure values are used in the dose analyses to demonstrate compliance with the dose limits of Regulatory Guide 1.183. For example, for the St. Lucie Unit 1 Locked Rotor event, with DNB fuel failures of 13.7% and the unfiltered control room inleakage of 500 cfm, the AST dose for control room is calculated to be 2.53 rem TEDE, which meets the acceptance criteria (dose limit) of 5 rem TEDE. The limit that the cycle specific core design must meet for this event is, therefore, 13.7% DNB failures. As long as the amount of fuel failures are less that those used in the dose analyses (13.7% for this event), the acceptance criteria are confirmed to be met. 13.7% DNB fuel failure is thus the maximum allowable fuel failure for this event to stay within the analyzed dose consequences.

Typical cycle specific fuel failures are much less than the fuel failure limits established from dose considerations. For the outside containment steam line break event, current cycle specific core designs result in no fuel failures, whereas for the inside containment steam line break and the locked rotor events, cycle specific DNB fuel failures are less than 2.5%. No fuel centerline melt (FCM) fuel failures result for either St. Lucie Unit. These actual cycle specific fuel failures are thus significantly lower than the limiting allowable DNB fuel failures of 13.7% for the locked rotor event and 29% for the inside containment steam line break event.

NRC Question 6:

In Table 2.1-1 of both NAI-1101-043 and NAI-1101-044, under emergency core cooling system (ECCS) leakage the chemical form of the iodine in the sump water is defined as 0% aerosol, 97% elemental, and 3.0% organic. Assumption 22 of §2.1.2 of both NAI-1101-043 and NAI-1101-044, describes regulatory compliance with Regulatory Position 5.6, for ECCS leakage into the auxiliary building, and states that the form of the released iodine is 97% elemental and 3% organic. The NRR staff is unclear if the statement in Table 2.1-1 was intended to reflect Regulatory Position 5.6 describing the chemical form of the iodine released from ECCS leakage or the chemical form of the iodine in the sump water. Please provide additional information to clarify the cited statement in Table 2.1-1.

FPL Response to Question 6:

The statement in Table 2.1-1 of both NAI-1101-043 and NAI-1101-044 regarding the chemical form of the iodine in the sump water was intended to reflect Regulatory Position

5.6. The values cited in Table 2.1-1 are consistent with the values cited in assumption 22 of Section 2.1.2 of both NAI-1101-043 and NAI-1101-044 which specifically discusses compliance with Regulatory Position 5.6.