UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-250 OLA
FLORIDA POWER & LIGHT COMPANY	)	50-251 OLA
(Turkey Point Plant, Units 3 and 4)	)	

AFFIDAVIT OF SUMMER B. SUN
REGARDING CONTENTION (b)

I, Summer B. Sun, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Nuclear Engineer in the Core Performance Branch of the Division of Systems Integration, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.

2. The purpose of this affidavit is to address Intervenors' Contention (b), which states:

> Whether the entirely new computer model used by the utility, for calculating flood portions of accidents meets the Commission's ECCS Acceptance Criteria: specifically, whether a 2.2% reduction in re-flood rate is misleading because for a small decrease in re-flood rate, there results a large increase in fuel temperature. Reflood rates are critical if below 1 or 2 inches per minute.

3. I have read the "Licensee's Motion for Summary Disposition of the Intervenors' Contention (b)" and the "Licensee's Statement of Material Facts as to Which There is no Genuine Issue to be Heard With Respect to Intervenors' Contention (b)," dated August 10, 1984. The facts presented in relation to Contention (b) are correct and are supported

by the findings and conclusions of the NRC Staff's Safety Evaluation, dated December 21, 1983, in support of Amendment Nos. 99 and 93 to the facility operating licenses for Turkey Point Plant Unit Nos. 3 and 4, respectively. The material facts are also supported by the Staff Safety Evaluation of the BART A-1 Code (BART A-1 SE), dated December 21, 1983.

The following information and details expand on the factors and considerations provided in the above referenced Safety Evaluation related to Contention (b).

4. Contention (b) alleges that the BART A-1 code is not in compliance with the Appendix K requirements and expresses a concern about the applicability of the BART A-1 code for use in reflood calculations with low flooding rates (specifically, less than one inch/second). The BART A-1 code is developed based upon a mechanistic model to predict the fuel rod behavior during reflood. This code is incorporated into the modified version of the 1981 ECCS evaluation model and is used to replace the FLECHT correlation used in the Westinghouse ECCS evaluation model for reflood calculations. The BART A-1 code calculates a time and axial location dependent fuel rod clad surface heat transfer coefficient as input to the detailed, previously approved fuel rod heatup code, LOCTA. The BART A-1 calculation is based upon the hydraulic information calculated by the approved code, WREFLOOD, and beginning-of-reflood fuel rod initial conditions. Both the approved LOCTA and WREFLOOD Codes were included in the approved Westinghouse 1981 ECCS evaluation model for a large break LOCA analysis.

5. Staff has reviewed and approved the BART A-1 computer code. The Staff has evaluated the conformance of BART A-1 with the Appendix K,

¶ I.D.5 requirements for reflood calculation with reflood rates above and below one inch/second, and concluded that the BART A-1 code is acceptable for reflood calculations. The evaluation of the conformance of BART A-1 with the Appendix K requirements is documented in Section 2.8 of the BART A-1 SE, dated December 21, 1983.

6. Contention (b) is also concerned with the effect of the 2.2% reduction in the reflood rate on the calculated peak cladding temperature (PCT) and whether the conditions imposed by the NRC for the acceptance of the BART A-1 code are met (see Intervenors' Response to Interrogatories Propounded by Florida Power & Light Company, dated July 10, 1984 at ¶ b-7.) As discussed in Section 4.2 of our Safety Evaluation, dated December 23, 1983, the Licensee has applied the approved modified 1981 ECCS evaluation model using BART A-1 instead of the FLECHT correlation to calculate the PCT for a large break LOCA for Turkey Point Units 3 and 4. The resulting PCT is 1972°F for both the LOPAR and OFA homogeneous cores. For the transition mixed core, the calculated PCT is only 10°F higher than a homogeneous core. This slight increase of 10°F in PCT is due to the hydraulic resistance of the OFA fuel which is 4.5% higher than that of the LOPAR fuel and in turn, results in a reduction of reflood steam flow velocity past midplane of the OFA fuel by about 2.2%.

7. Contention (b) also states that a 2.2% reduction in reflood rate could result in a large increase in PCT and reflood rates are critical if below one or two inches per minute. As discussed in Section 2.8 of the BART A-1 SE, there is no difference in heat transfer mechanism for reflooding rates below or above one inch per second. The reflood rate, calculated with the NRC approved computer code WREFLOOD,

would be less than 2.2% and within the range of our approval in the BART A-1 Safety Evaluation. The Staff has reviewed the analytical results and found that (1) the BART A-1 code was used within the approved applicable regions, (2) an appropriate nodal scheme was used to represent the fuel rod and (3) the grid spacer model was not used for the ECCS analysis. Therefore, the Staff concluded that the conditions specified in Section 4.0 of the BART A-1 SE are fully satisfied. The Staff also concluded that the analytical results are acceptable since the ECCS analysis correctly uses approved BART A-1 code and demonstrated that the calculated PCT is less than the safety limit of 2200°F specified in 10 C.F.R. § 50.46. Safety Evaluation, dated December 23, 1983 at § 4.2. In addition, the Staff agrees with Licensee (see Parvin affidavit) that the 2.2% reduction is not in reflood rate but rather in reflood hot assembly steam flow velocity. The 2.2% reduction in steam flow velocity results in the slight temperature increase of 10°F.

8. For assessment of the effect of using BART A-1 and using the FLECHT correlation on the calculated PCT, the Staff requested the Licensee to perform a large break LOCA analysis using the previously approved non-modified 1981 ECCS evaluation model including the FLECHT correlation. The results show a PCT of 2130°F for a homogeneous core and the worst LOCA case. Adding 10°F for the transition mixed core still results in a PCT of less than the limit of 2200°F specified in 10 CFR 50.46. This calculational result demonstrates that even without using the BART A-1 code, and instead using the previously approved FLECHT correlation in the ECCS evaluation model, the result will be a PCT of less than the acceptable limit of 2200°F. Id.

9. In summary, based on the NRC Staff's Safety Evaluations related to the amendments and the BART A-1 code, (1) the BART A-1 code satisfies the applicable requirements of 10 CFR Part 50, Appendix K, and (2) the reduction in reflood hot assembly steam flow velocity associated with core reload results in a PCT which is less then the 2200° F limit imposed by 10 CFR § 50.46 when calculated either using the BART A-1 code or the previously approved ECCS model using the FLECHT correlation. Thus, the Staff concludes that the operation of Turkey Point Units 3 and 4 in accordance with the amendments imposes no undue risk to the health and safety to the public.

The foregoing and the attached statement of professional qualifications are true and correct.

Summer B. Sun

Summer B. Sun

Subscribed and sworn to before me
this 4th day of September, 1984

Malinda L. McDonald

Notary Public

My commission expires: 7/1/86

Summer B. K. Sun
Core Performance Branch
Division of Systems Integration
U. S. Nuclear Regulatory Commission

PROFESSIONAL QUALIFICATIONS

I am employed as a nuclear engineer in the Core Performance Branch of the Division of Systems Integration.

I graduated from National Taiwan University with a B.S. in Chemical Engineering in 1967. I received a Ph.D degree in Chemical Engineering from University of Missouri at Columbia in 1974. I am a registered Professional Engineer, Certificate Number 11309, in the State of Connecticut.

In my present work assignment, I have technical responsibility for the review of the reactor core thermal-hydraulic design submitted in reactor construction permit and operating license applications. In addition, I participate in the review of analytical models used in licensing evaluation of the core thermal-hydraulic behavior under operating, postulated accident and transient conditions.

Prior to joining the NRC staff in August 1980, I was employed by Combustion Engineering Company (CE). I was responsible for the safety analysis method development and application of methods for the transient analyses. My responsibility at CE includes safety and performance analyses in the area of thermal-hydraulic and system designs. My tenure at CE was from 1974 through 1980.

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(Turkey Point Plant,	)	
Unit Nos. 3 and 4)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO LICENSEE MOTIONS FOR SUMMARY DISPOSITION OF CONTENTIONS (b) AND (d)" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 4th day of September, 1984:

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Administrative Judge
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Mitzi A. Young

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