UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-250-OLA-2
	)	50-251-OLA-2
FLORIDA POWER AND LIGHT COMPANY	)	
	)	
(Turkey Point Nuclear Generating Units 3 & 4)	)	(Spent Fuel Pool Expansion)

AFFIDAVIT OF REBECCA K. CARR
ON CONTENTION NO. 3

1. My name is Rebecca K. Carr. I am employed by Bechtel Power Corporation, Eastern Power Division, as an engineer in the Operating Services nuclear licensing group. As part of my previous duties as Group Leader within the Radiation Analysis Group of the Nuclear Engineering Staff, I supervised radiological evaluations performed in support of the expansion of the spent fuel storage capacity at Turkey Point Units 3 and 4. A summary of my professional qualifications and experience is attached as Exhibit A and is incorporated herein by reference.

2. The purpose of my affidavit is to address Contention 3. Contention 3 and the bases for the Contention are as follows:

Contention 3

That the calculation of radiological consequences resulting from a cask drop accident are not conservative, and the radiation releases in such an accident will not be ALARA, and will not meet with the 10 CFP [sic] Part 100 criteria.

Bases for Contention

The Florida Power and Light Company did not comply with the conservative assumption for a cask drop accident that are specified in the Standard Review Plan 15.7.5 (5) and Regulatory Guide 125 [sic] (5), in that they used a 1.0 radial peaking factor, rather than a 1.65 factor. Thus, the potential offsite dose using the more conservative calculations could cause FPL to exceed the 10 CFP [sic] Part 100 criterion.

(In the Memorandum and Order dated September 16, 1985, the Licensing Board stated that "reference to the ALARA principle is inappropriate because ALARA generally applies to routine operation, not accidents.") This affidavit demonstrates that the offsite doses for a postulated cask drop accident at Turkey Point were calculated conservatively using appropriate peaking factors and that the resultant radiation doses are well within the guidelines of 10 CFR Part 100.

3. The remainder of this affidavit is divided into three principal parts. The first part defines radial peaking factor and describes how the radial peaking factor relates to the physical phenomena occurring in the reactor core. The second part discusses the two cases involving cask drop accidents that were analyzed in the safety analysis report for the Turkey Point spent fuel pool expansion, and it demonstrates that the peaking

factors used in each case were appropriate. Finally, the third part shows that the radiological doses resulting from the cask drop accident would be acceptable even if a radial peaking factor of 1.65 was used in both cases.

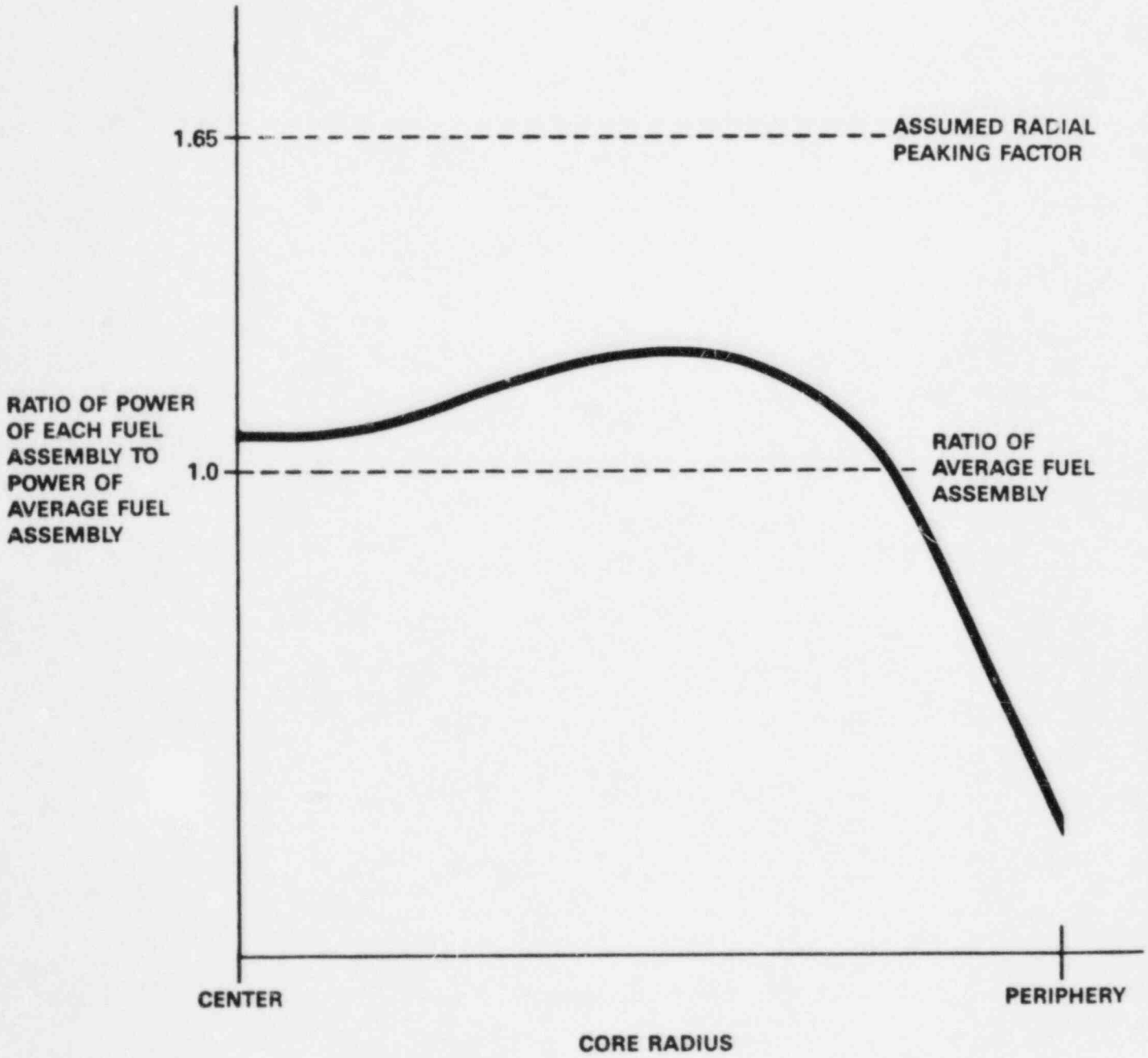
I. Discussion of Radial Peaking Factor

4. In performing an analysis of the radiological impacts of a postulated cask drop accident in a spent fuel pool, it is necessary to calculate the amount of fission products in the fuel assemblies in the spent fuel pool which could be available for release as a result of a cask dropping onto the spent fuel assemblies. The production of fission products in a fuel assembly and the power of the assembly throughout its operation in the core are both proportional to the number of fissions which have occurred in the assembly. Therefore, the production of fission products in an assembly is also proportional to the power of the assembly throughout its operation in the core. Since the amount of power produced by an assembly depends upon its location in the reactor core, the amount of fission products in a fuel assembly will vary from assembly to assembly based upon its power and location in the reactor core.

5. One method of quantifying the variation in the power of the fuel assemblies in a reactor core is to calculate the ratio of the power of each assembly to the power of the average assembly. Figure 1 shows how this ratio, in general, varies depending upon the radial location of a fuel assembly in the

Figure 1

REPRESENTATIVE CORE RADIAL POWER PROFILE



core. As is demonstrated in Figure 1, this ratio tends to vary from a value of less than 1.0 at the outer portions of the core (where the fuel assemblies are producing less power than the average assembly) to a value of greater than 1.0 toward the center of the core (where the fuel assemblies are producing more power than the average assembly). The maximum ratio is the radial peaking factor, which can be expressed as the ratio of the maximum assembly power to the average assembly power.

6. Several aspects of Figure 1 should be noted. First, the radial peaking factor applies only to the assemblies in the core which produce the maximum power; the other assemblies in the core have a lower ratio of power to average power than the assemblies with the radial peaking factor. Second, use of a radial peaking factor is inappropriate when calculating the amount of fission products in all of the assemblies in the core, since the fission products in the core as a whole are dependent upon the overall power of the core and not the power of any individual assembly. In other words, when dealing with all of the assemblies in the core, it is appropriate to use the characteristics of the average assembly which, by definition, has a ratio of power versus average power equal to 1.0.

7. Assumptions regarding radial peaking factors which the Nuclear Regulatory Commission (NRC) Staff has found acceptable for use in analysis of cask drop accidents are contained in NRC

Standard Review Plan (SRP) Section 15.7.5 1/ and NRC Regulatory Guide (RG) 1.25. 2/ SRP Section 15.7.5, paragraph II.3, states that the NRC Staff will accept a model for calculating the consequences of a cask drop accident "if it incorporates the appropriate conservative assumptions in NRC Regulatory Guide 1.25." Section B of Regulatory Guide 1.25 does not specify the number of fuel assemblies which should be assumed to be damaged as a result of a cask drop accident, but instead "is addressed to the determination of the radiological consequences of a handling accident once an assumption as to the number of assemblies or rods damaged has been made." In this regard, Section B of Regulatory Guide 1.25 states that a conservative approach "is to assume that the assembly with the peak [fission product] inventory is the one damaged," and Section C.1.e states that the fission product inventory should be calculated using "an appropriate radial peaking factor." This section also states that the minimum acceptable radial peaking factor for a pressurized water reactor (such as Turkey Point) is 1.65.

8. The analysis of cask drop accidents at Turkey Point consisted of two cases using different assumptions regarding the number of freshly discharged assemblies damaged and the radial

---

1/ NUREG-0800, U.S. Nuclear Regulatory Commission Standard Review Plan 15.7.5, "Spent Fuel Cask Drop Accidents," Rev. 2 (July 1981).

2/ Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (March 23, 1972).

peaking factor. As is discussed below, the assumptions regarding radial peaking factors were appropriate in each case given the number of assemblies assumed to be damaged.

II. Analyses of Postulated Cask Drop Accidents at Turkey Point

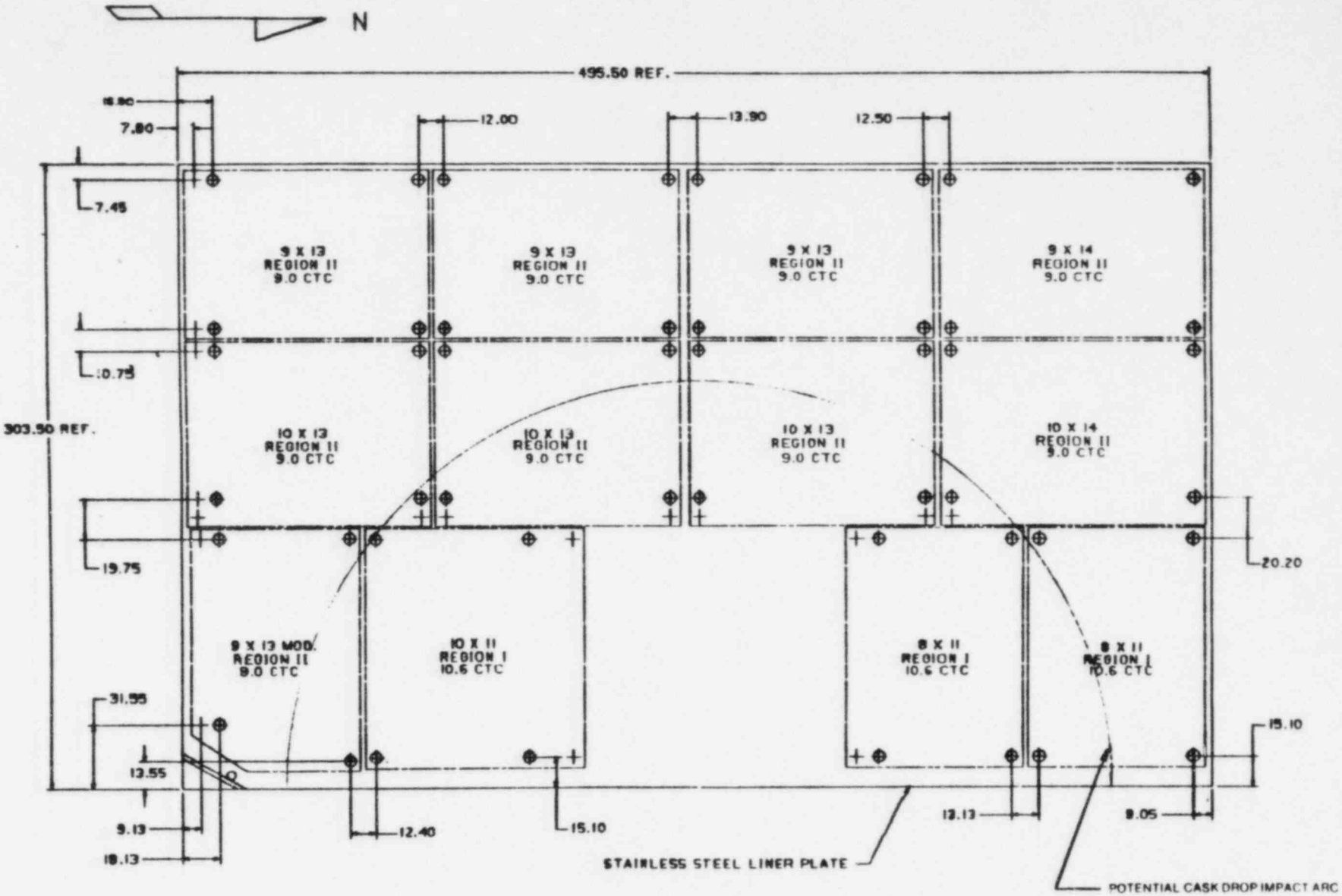
9. Two cases of cask drop accidents were evaluated in Section 5.3.1.2 of the Safety Analysis Report (SAR) for the Turkey Point spent fuel pool expansion. 3/ Both cases incorporate the assumptions of SRP Section 15.7.5 and RG 1.25, where appropriate, and both cases result in offsite doses that are well within 10 CFR Part 100 guidelines. The two cases and their respective assumptions are described in detail below.

A. Case 1

10. The following assumptions were made in Case 1 of the cask drop analysis:

a) It was postulated that 1404 fuel assemblies (which is the capacity of all of the new storage racks) would be damaged as a result of a cask falling into the pool. As shown on Figure 4-4 of the FPL-SAR (attached), a dropped cask could only impact stored fuel assemblies within a limited area of the spent fuel pool, and it would be impossible to damage every fuel assembly as a result of a dropped cask. Thus, the assumption that all fuel assemblies in the pool are damaged is conservative.

3/ Turkey Point Units 3 & 4, "Spent Fuel Storage Facility Modification Safety Analysis Report, " Docket Nos. 50-250 and 50-251, March 14, 1984 (hereafter FPL-SAR).



UNIT 3 — AS SHOWN
UNIT 4 — OPPOSITE HAND

ALL DIMENSIONS IN INCHES

Figure 4-4

SPENT FUEL POOL ARRANGEMENT

b) It was postulated that the fuel assemblies in the spent fuel pool at the time of the accident consist of 80 freshly discharged fuel assemblies (which is in excess of a normal refueling off-load) and 1324 fuel assemblies discharged during previous refuelings which have decayed for at least eighteen months. A typical refueling at either Turkey Point Unit 3 or 4 would consist of less than half of a full core (a full core contains 157 assemblies).

c) A radial peaking factor of 1.65 was conservatively applied to all 80 freshly discharged fuel assemblies. No radial peaking factor was applied to the other assemblies stored in the spent fuel pool (mathematically, this is equivalent to saying that the radial peaking factor equals 1.0).

d) Other assumptions made in Case 1 were consistent with the assumptions in RG 1.25.

11. Using these assumptions, conservative approximations of the thyroid and whole body doses were calculated in accordance with RG 1.25, Sections C.3.a and C.3.b, respectively. The results of the analysis indicate that, by conservatively assuming a minimum of 1475 hours of decay time prior to moving a spent fuel cask into the spent fuel pool, offsite doses at the closest site boundary from a postulated cask drop would be 27 rem to the thyroid and less than 1 rem to the whole body. These doses are well within the 10 CFR Part 100 guidelines of 300 rem thyroid

dose or 25 rem whole body dose. 4/ That is, the doses are less than 75 rem (25% of 300) for the thyroid and less than 6 rem (about 25% of 25) for the whole body doses. The Part 100 guideline doses are commonly used in the nuclear industry for evaluating the acceptability of accident conditions, and SRP Section 15.7.5, paragraph II, states that the doses calculated for cask drop accidents are acceptable if they are well within the Part 100 guidelines.

B. Case 2

12. The following assumptions were made in Case 2 of the cask drop analysis:

a) Case 2 also postulated that 1404 fuel assemblies would be damaged by a dropped spent fuel cask. However, Case 2 considered 157 freshly discharged fuel assemblies (full-core offload) with the remaining fuel assemblies discharged during previous refuelings. These 1247 previously discharged assemblies were assumed to have decayed for at least 18 months. As discussed above, it is conservative to assume that every fuel assembly in the spent fuel pool would be damaged in a cask drop accident.

4/ SRP Section 15.7.5, paragraph II.1, defines "well within" as less than 25% of the doses in the 10 CFR Part 100 guidelines.

b) In the second case, a radial peaking factor was not applied to the fission product inventories. Mathematically, this is equivalent to saying that the radial peaking factor equals 1.0.

c) All other assumptions listed above for Case 1 were applied to Case 2 as well.

13. The results of the analysis indicate that, with a minimum decay time of 1525 hours prior to moving a spent fuel cask into the spent fuel pool (which is a condition in Turkey Point's technical specifications once the new high density storage racks are installed), the thyroid dose at the closest site boundary would be 27 rem and the whole body dose would be less than 1 rem should a dropped cask strike the stored fuel assemblies. These doses are well within the 10 CFR Part 100 guideline values of 300 rem thyroid and 25 rem whole body. That is, the doses are less than 75 rem (25% of 300) for the thyroid and less than 6 rem (about 25% of 25) for the whole body doses.

C. Propriety of the Assumptions Regarding the Radial Peaking Factor

14. In performing Case 1 of the accident analysis for the cask drop in the Turkey Point spent fuel pool, a radial peaking factor of 1.65 was applied to the 80 freshly discharged fuel assemblies. It was conservative to apply a radial peaking factor of 1.65 to all 80 fuel assemblies, because this implies that over half the assemblies in the core produced the peak power -- a condition which is not realistic. However, it might be possible,

although unlikely, for all of those 80 assemblies to have produced higher than the average assembly power. Therefore, an adjustment factor (in the form of a radial peaking factor) was appropriate in this case. Although the value of 1.65 overestimates the power and fission product inventories of the 80 assemblies, it was used for simplicity and conservatism rather than explicitly quantifying the ratio of actual power to average power for each of the 80 assemblies.

15. In Case 2 of the cask drop analysis, a freshly discharged full core off-load was assumed to be in the pool at the time of the postulated accident. For this case, no radial peaking factor was applied to the full core. As stated above, the radial peaking factor accounts for the assemblies with the peak fission product inventory. It would be incorrect to apply a radial peaking factor to the entire core fission product inventory. The full core inventory, which includes the peak power assembly as well as the lowest power assembly, accounts for all fission products and no additional adjustment is necessary or appropriate to account for assemblies with higher than average power. In other words, the overall average of the ratio of power to average power of all of the assemblies in a full core must, by definition, be equal to 1.0.

16. In both Case 1 and Case 2, no radial peaking factor was applied to the fuel assemblies assumed to be discharged during previous refuelings and stored for at least eighteen months. This assumption is appropriate because, for the large number of

assemblies assumed to be in the pool (approximately 8 to 9 cores), the stored assemblies on the average would be representative of the assemblies in an entire core. Thus, for the reasons discussed in the previous paragraph, no adjustment to the power levels of these assemblies is necessary. In any case, due to radioactive decay of the fission products, the assemblies discharged from previous refuelings contribute relatively little to the offsite doses in comparison to the freshly discharged assemblies, 5/ and application of a radial peaking factor of 1.65 to these assemblies would not significantly affect the results of the dose analyses.

III. Application of a Peaking Factor of 1.65 to a Full Core Off-Load

17. While recognizing that such an approach is not necessary to establish compliance with the applicable regulatory bases in SRP Section 15.7.5 and RG 1.25, a cask drop damaging a full core off-load of fuel assemblies with 1525 hours decay and a radial peaking factor of 1.65 was also evaluated for the purposes of this affidavit in order to test the sensitivity of the assumptions used in Cases 1 and 2. The total number of assemblies assumed to be damaged was 1404; 157 freshly discharged assemblies plus 1247 previously discharged assemblies decayed for

---

5/ Assuming less than eighteen months decay for the fuel assemblies discharged during the previous refueling would not result in a net increase in the calculated doses, because in that event the freshly discharged fuel assemblies would have a correspondingly shorter period of operation at power, thereby reducing their fission product inventory.

18 months. Using these assumptions, the site boundary thyroid dose was 45 rem and the site boundary whole body dose was less than 1 rem. These doses are still well within the 10 CFR Part 100 guidelines of 300 rem thyroid and 25 rem whole body. Therefore, even if a radial peaking factor of 1.65 had been utilized in Case 2, the results would still be acceptable.

IV. Conclusion

18. In summary, Florida Power and Light Company did comply with the conservative assumptions for a cask drop accident that are specified in SRP Section 15.7.5 and RG 1.25. Specifically, a radial peaking factor of 1.65 was used in the analysis, where appropriate. The potential offsite doses resulting from a postulated cask drop accident are well within the guidelines of 10 CFR Part 100.

FURTHER AFFIANT SAYETH NOT

The foregoing is true and correct to the best of my knowledge, information and belief.

Rebecca K Carr
Rebecca K. Carr

STATE OF MARYLAND)
COUNTY OF MONTGOMERY)

Subscribed and sworn to before me this 22 day of January, 1986. My commission expires:

My Commission Expires July 1, 1986

Roberta B. Curreri

EXHIBIT A

STATEMENT OF PROFESSIONAL QUALIFICATIONS OF
REBECCA K. CARR

POSITION Project Licensing Engineer, Bechtel Power Corporation
EDUCATION BS, Nuclear Engineering, Pennsylvania State University, 1980

SUMMARY OF EXPERIENCE WITH BECHTEL:

Project licensing engineer, 1984-Present

Staff group leader, shielding, 1983-1984

Staff engineer, radiation analysis, 1980-1983

EXPERIENCE WITH BECHTEL

Ms. Carr is currently serving as licensing engineer with the Operating Services Group. This group provides engineering services to utilities with operating nuclear power plants, including the North Anna Power Station, Surry Power Station, and Millstone Nuclear Power Station. Her responsibilities include safety reviews of design changes to ensure compliance with NRC requirements, FSAR criteria, and plant technical specifications. Ms. Carr is also involved with licensing the steam generator replacement at Indian Point Station Unit 3, the independent spent fuel storage installation at Surry Power Station, and the spent fuel pool reracking at Turkey Point Plant.

Prior to this, Ms. Carr served as an engineer in the nuclear licensing group for the two-unit Grand Gulf Nuclear Station (Mark III). She was responsible for coordinating implementation of regulatory requirements, safety reviews and staff analyses for the operating Unit 1. While supporting the Unit 2 design effort, she was the lead engineer for the computer based licensing commitment tracking system.

Previously, Ms. Carr served as group leader - radiation analysis, on the Nuclear Engineering Department staff. In this capacity, she was responsible for shielding and dose analyses in support of both BWR and PWR projects in the construction and operating phases. Plants included the Grand Gulf Nuclear Station, Edwin I. Hatch Nuclear Plant, Wolf Creek Generating Station, Callaway Plant, Joseph M. Farley Nuclear Plant, Turkey Point Plant, and Calvert Cliffs

Nuclear Power Plant. Work included operating and accident doses, equipment qualification (radiation), spent fuel pool reracking, low level waste processing and storage, and steam generator replacement.

As a staff engineer, Ms. Carr was involved in the analysis of airborne radiation releases and doses within plants and in the environment resulting from normal operation and postulated accidents. This included control rooms and emergency facilities. She also performed shielding analyses, including neutron streaming, and fulfilled a licensing assignment at the Three Mile Island jobsite. In addition, Ms. Carr participated in several audits of design and analysis work done by projects.

PROFESSIONAL MEMBERSHIPS

American Nuclear Society and Society of Women Engineers