UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	)	
FLORIDA POWER & LIGHT COMPANY	)	Docket No. 50-335 OLA
(St. Lucie Plant, Unit No. 1)	)	ASLBP No. 88-560-01-LA

AFFIDAVIT OF STEPHEN MARSCHKE
ON ADMITTED CONTENTION 1

I, Stephen F. Marschke, being duly sworn, say as follows:

1. I am currently employed as a Supervising Engineer in the Nuclear Licensing Department at Ebasco Services Incorporated. My business address is Ebasco Services Incorporated, 2 World Trade Center, 89th Floor - NW, New York, New York, 10048. Prior to transferring to the Nuclear Licensing Department, I was a Supervising Engineer within Ebasco's Radiological Consulting Department. While there I was responsible for determining the radiological consequences resulting from a cask drop accident that are presented in the Safety Analysis Report submitted in support of the expansion of the spent fuel storage capability at St. Lucie 1. A summary of my qualifications and experience is attached hereto as Exhibit A, which is incorporated herein by reference. I have personal knowledge of the matters stated herein, and believe them to be true and correct. This affidavit is offered in support of "Licensee's Motion for Summary Disposition of Intervenor's Contentions," regarding Admitted Contention 1.

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2. Admitted Contention 1 (Originally Amended Petition Contention 3) states:

> That the calculation of radiological consequences resulting from a cask drop accident are not conservative and that radiation releases in such an accident will not meet the 10 CFR Part 100 criteria.

3. The bases for Admitted Contention 1 were stated as follows:

> The study prepared by the Department of Nuclear Energy, Brookhaven National Laboratory entitled "Severe Accidents in Spent Fuel Pools in Support of Generic Safety," NUREG/CR-4982, BNL-NUREG-52093, indicates that ". . . the calculation of radiological consequences resulting from such an accident are, at this point in time, apparently impossible to determine." "There is substantial uncertainty in the fission product release estimates. These uncertainties are due to both uncertainty in the accident progression (fuel temperature after clad oxidation and fuel relocation occurs) and the uncertainty in the fission product decontamination." (S.6) In light of such uncertainty, no estimate can be determined to be conservative.

4. The purpose of my affidavit is to demonstrate that the off-site doses for a postulated cask drop accident at St. Lucie 1 were calculated conservatively, with the resulting radiation doses being well within the criteria of Title 10 of the Code of Federal Regulations, Part 100 (10 CFR Part 100). The Affidavit of Murray Weber, also offered in support of "Licensee's Motion for Summary Disposition of Intervenor's Contentions," regarding Admitted Contention 1, addresses how a postulated cask drop will not result in draining of the spent fuel pool water or uncovering of stored spent fuel.

5. The cask drop accident analysis addressed the off-site consequences of a radioactivity release from the spent fuel rod's gas gap. Criticality following a cask drop accident is not of concern due to 1) the fact that the crushed fuel will be in a less critical geometry then the intact fuel rods, and 2) the presence of neutron poison in the spent fuel pool water (see Affidavit of Dr. Stanley E. Turner on Admitted Contentions 6 and 7).

6. In 10 CFR Part 100 the Nuclear Regulatory Commission (NRC) presents guidance to be used in the determination of the exclusion area and low population zone surrounding a nuclear power reactor. This guidance is given in 10 CFR Section 100.11 (1988). In effect, it provides that: a) an individual located at any point on the exclusion area boundary for two hours immediately following the onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure; and b) an individual located at any point on the outer boundary of the low population zone who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

7. The major distinction between the above definitions is that for the exclusion area the exposures are limited to a two hour duration, while at the low population zone the exposures extend for the duration of the accident. For St. Lucie 1 the exclusion area boundary is at 5100 feet and the low population zone boundary is at one mile.

8. For accidents, such as the cask drop accident, where the release is short term, i.e., less then two hours, the dose at the exclusion area boundary is the controlling criteria. This is because the duration of exposure will be the same at both locations and, since the exclusion area boundary is closer to the reactor, the concentration of radioactivity within the plume will be less dispersed at the exclusion area boundary than at the low population zone boundary.

9. The NRC Staff has developed criteria for the evaluation of applicant/licensee analyses of spent fuel cask drop accidents. These are found in its Standard Review Plan (SRP) (NUREG-0800), Section 15.7.5, "Spent Fuel Cask Drop Accidents," (Rev. 2 - July 1981), paragraph II. In particular, paragraph 15.7.5.II.1 provides that the potential radiological consequences of a postulated spent fuel cask drop accident should be "well within" the 10 CFR Part 100 exposure guidelines. It then goes on to define "well within" to mean "... 25 percent or less of the 10 CFR Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses."

10. St. Lucie 1 Technical Specification 3.9.14 requires that, following normal refueling, when up to one third of the core is placed into the spent fuel pool, 1180 hours must elapse prior to the movement of the spent fuel cask into the fuel cask area. This means that any fuel assemblies damaged in a cask drop accident would have experienced decay of at least 1180 hours. As shown in Table 1, attached and incorporated herein by reference, if a Case 1 cask drop accident (see paragraph 15) occurs 967, or more, hours after reactor shutdown, then the exclusion area exposures will be 10%, or less, of the 10 CFR Part 100 guidelines. St. Lucie 1 Technical Specification 3.9.14 requires that, following the removal of the entire core, 1490 hours must elapse prior to the movement of the spent fuel cask into the fuel cask area. This means that any fuel assemblies damaged in a cask drop accident would have experienced decay of at least 1490 hours. As shown in Table 1, if a Case 2 cask drop accident (see paragraph 17) occurs 1273, or more, hours after reactor shutdown, then the exclusion area exposures will be 10%, or less, of the 10 CFR Part 100 guidelines.

11. Section 15.7.5 of the SRP identifies Regulatory Guide 1.25 (R.G. 1.25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handing Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," as the source of appropriate conservative assumptions that should be utilized in performing the spent fuel cask drop radiological consequence analysis. Appropriate R.G. 1.25 assumptions can and were utilized in the St. Lucie 1 radiological analysis.

Specifically, the fraction of total rod radioactivity contained in the gas gap and the pool decontamination factors for the inorganic and organic species of iodine were used. Only noble gases and radioactive iodine were considered in the analysis, in accordance with R.G. 1.25.

12. The radiological impact of a postulated accident involving a cask's dropping onto stored spent fuel assemblies depends upon the amount of fission products of concern released from the fuel in storage. In evaluating the consequences of a cask drop accident for St. Lucie 1 the release fractions specified in R.G. 1.25 were used with the assumption that all fuel assemblies in the completely filled spent fuel pool were ruptured.

13. As was done in the St. Lucie 1 Updated Final Safety Analysis Report, the cask drop accident for the expanded spent fuel pool capacity was analyzed for two different sets of assumptions. The first case assumed that the accident occurred after a normal refueling, in which one third of the number of fuel assemblies in the core, i.e., 80 fuel assemblies, were removed to the spent fuel pool. The second case assumes that all of the fuel assemblies in the core, i.e., 217 fuel assemblies, were removed to the spent fuel pool. In both cases a non-mechanistic release of all of the radioactive noble gases and iodine contained in the gas gap of all of the fuel rods from all of the stored fuel in the pool, i.e., 1706 fuel assemblies, is assumed.

14. In summary, the assumptions that all of the fuel assemblies in the spent fuel pool were damaged and that all of the radioactive noble gases and iodine contained in the gas gap of all the fuel rods from all of the stored fuel in the pool are extremely conservative assumptions.

Case 1

15. For Case 1 it is assumed that one third of the fuel assemblies from the core are placed in the spent fuel pool each year during refueling for 23 years, until the pool is filled. All of the fuel assemblies in the full pool are assumed to be damaged as a result of the cask drop accident. This is a very conservative assumption, since the cask movement is limited to the spent fuel cask storage area in the northeast corner of the pool and is physically prevented from being directly over any stored fuel (see the Affidavit of Murray Weber on Admitted Contention 1).

16. Exclusion area boundary doses were determined as a function of the time after the most recent reactor shutdown that the accident is assumed to occur ("decay time"). The resulting whole body and thyroid doses are shown on Table 1. Given the limitations in Technical Specification 3.9.14, discussed above in paragraph 10, regarding minimum decay time, the whole body and thyroid doses are well within 10 CFR Part 100 guideline exposure values.

Case 2

17. For Case 2 it is assumed that one third of the fuel assemblies from the core are placed in the spent fuel pool each year during refueling for 20 years. Following the twenty first year the entire core is removed from the reactor and placed into the pool at once, filling the pool. All of the fuel assemblies in the full pool are assumed to be damaged as a result of the cask drop accident. This is a very conservative assumption, since the cask movement is limited to the spent fuel cask storage area in the northeast corner of the pool and is physically prevented from being directly over any stored fuel (see the Affidavit of Murray Weber on Admitted Contention 1).

17. Exclusion area boundary doses were determined as a function of the time after the most recent reactor shutdown that the accident is assumed to occur ("decay time"). The resulting whole body and thyroid doses are shown on Table 1. As with Case 1, given the limitations in Technical Specification 3.9.14, discussed above in paragraph 10, regarding minimum decay time, the whole body and thyroid doses are well within 10 CFR Part 100 guideline exposure values.

Conclusions

19. The analyses of the radiological consequences resulting from a cask drop accident are conservative and the radiation releases are such that a cask drop accident will meet the 10 CFR Part 100 criteria. The St. Lucie 1 Technical Specification on spent fuel cask movement will ensure that exclusion area exposures are within 10% of the 10 CFR Part 100 guidelines.

20. Expanding the capacity of the spent fuel pool will have negligible effect on the exposure at the exclusion area due to a postulated cask drop accident. The reason for this being that all of the isotopes of iodine of concern for this analysis are of sufficiently short half-life, i.e., eight days or less, that they will have decayed to negligible levels between refuelings. Therefore, the required decay times are dependent only on the amount of fuel freshly removed from the reactor, i.e., one third of the core for Case 1 and the entire core for Case 2.

FURTHER AFFIANT SAYETH NAUGHT.

Stephen F. Marschke

Stephen F. Marschke

Sworn and subscribed before me this 25th day of July, 1988.

Patricia M. Kavanagh

NOTARY PUBLIC

PATRICIA M. KAVANAGH
Notary Public, State of New York
No. 24-2051025
Qualified in Richmond County
Cert. Filed in New York County
Commission Expires March 30, 1989

Table 1

Fraction of 10 CFR Part 100	Decay Time (hours)	Dose (Rem) Whole Body	Dose (Rem) Thyroid
Case 1			
25% Thyroid	712.	0.1	75.
10% Thyroid	967.	0.1	30.
Case 2			
25% Thyroid	1018.	0.1	75.
10% Thyroid	1273.	0.1	30.

EXHIBIT A

STEPHEN F MARSCHKE
Supervising Engineer
Nuclear Licensing

SUMMARY OF EXPERIENCE (Since 1973)

Total Experience - Fourteen years in the areas of Nuclear Licensing, radiological impact assessment and nuclear engineering.

Professional Affiliations - American Nuclear Society
IEEE Computer Society
AIF Subcommittee on Radiation Protection

Education - B.S., State University of New York at Buffalo, 1973
- Nuclear Engineering

REPRESENTATIVE EBASCO PROJECT EXPERIENCE (SINCE 1977)

Ebasco Nuclear Licensing Supervisor on the TVA Browns Ferry restart project. Responsibilities include preparing/updating FSAR and Tech Spec sections. Assisting in the preparation of NRC technical presentations on various restart licensing issues, including cable ampacity and design calculation review. Responsible for approving 10CFR50.59 unreviewed safety question determination (USQD) of all Ebasco generated engineering change notices at Browns Ferry.

Other experience includes performing the analyses required for Safety Analysis and Environmental Reports, Radiological Environmental Technical Specifications and Offsite Dose Calculational Manuals for various Ebasco client utilities, including LP&L, FP&L, CP&L, and CFE. Responding to questions from regulatory agencies concerning radiological safety. Provided testimony to the Atomic Safety and Licensing Board on the appropriateness of the radiological assessment models used to determine nuclear power plant normal operational impacts.

Manager of Ebasco's real time dose assessment computer system, CEPADAS. This system consists of over 45,000 lines of FORTRAN code, 15 separate tasks and 45 data files. Performed system design, FORTRAN debugging and overlay optimization. Wrote technical specifications, FORTRAN code, User's Manuals and acceptance test procedures. Handled all communications with clients. The system was developed on a Burroughs B7700, has been installed on a Gould 32/7780 and is presently being converted to a VAX 11/780.

STEPHEN F MARSCHKE (Cont'd)

Performed studies to determine the environmental and radiological consequences of decommissioning nuclear facilities. Principal author of the Office of Nuclear Waste Isolation report, "Decommissioning Requirements for Nuclear Waste Repository Licensing." Determined the off-site doses resulting from higher nuclear fuel burn-up for the AIF. Performed normal operational off-site ALARA evaluations and accident analyses to determine the required Engineering Safety Features for nuclear plants, e.g., Shearon Harris, St Lucie 2. Prepared the alternative waste disposal concepts, radiological impact sections of the high-level waste repository Environmental Impact Statement - DOE/EIS-0046F.

Developed the methodology and computer programs used to determine the time dependent buildup of radioactivity in alternative radwaste system designs. Performed ALARA analyses to determine the most advantageous mode of radwaste system design, calculating both in-plant and off-site impacts. Determined the effect on reservoir radionuclide concentration from the transfer of radionuclides to bottom sediments.

PRIOR EXPERIENCE (5 Years)

Ralph M Parsons Company, Nuclear Engineering (1 year)

Assigned to the project design team of the DOE's Fluorinel and Fuel Storage Facility (a nuclear fuels reprocessing plant) at the Idaho National Engineering Laboratory. Responsible for the determination of individual component and area gamma shielding requirements. Performed analyses to determine the proper design for shield wall piping, instrumentation and HVAC penetrations. Responsible for designing acceptable access labyrinths. Determined the exposure rate above the spent fuel pool from the spent fuel, the contaminated water and "skyshine."

United Engineers and Constructors, Inc, Nuclear Engineer (4 years)

Responsible for performing the radiological impact analyses of various postulated accidents in both PWR and HTGR systems. Performed analyses to determine the impact to control room personnel from both on-site and off-site radiological and toxic chemical accidents, including an accident of the Fast Flux Test Facility. Performed site radiological evaluation studies to determine which of a number of potential sites was preferred and for that site which was the preferred NSSS. Determined the radiological effects of corrosion in the PCRV steel cooling channels of an HTGR. Responsible for the

STEPHEN F MARSCHKE (Cont'd)

determination of fuel cycle costs for a number of fuel bid evaluations. Developed a computer program to determine the levelized nuclear fuel cycle costs, taking into account the costs of materials, processing, disposal and money. Wrote a computer program to determine the off-site normal operational radiological effects based on the models provided by the NRC in Regulatory Guide 1.109.

PUBLICATIONS

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Marschke, S F and J J Mauro, "Radionuclide Transport into Reservoir Bottom Sediments - A Licensing Approach," American Nuclear Society, June 1980.

Mauro, John J, et al, "The Environmental Consequences of Higher Fuel Burn-Up," National Environmental Studies Project, Atomic Industrial Forum, Inc AIF/NESP-032, June 1985.

Pon, W D and S F Marschke, "Conversion of Radionuclide Transport Codes to IBM Compatible Micro Computers," American Nuclear Society, November 1986.

Pon, W D and S F Marschke, "Conversion of Radionuclide Transport Codes from Mainframes to Personal Computers," Radiation Protection Management, Vol 4, No. 1, January/February, 1987.

Bhatia, R K, et al, "Preparation of License Application Discussions on Occupational Radiation Protection - Guidance Document," Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, September 1987.

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