4/16/82

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-250-SP
	)	
FLORIDA POWER & LIGHT COMPANY	)	50-251-SP
	)	
(Turkey Point Nuclear	)	(Proposed Amendments to Facility
	)	
Generating Unit Nos. 3	)	Operating License to Permit
	)	
and 4	)	Steam Generator Repairs)

AFFIDAVIT OF FREDERICK G. FLUGGER AND

H. H. JABALI ON CONTENTION 4A

My name is Frederick G. Flugger. My business address is P. O. Box 529100, Miami, Florida, 33152. I am Manager of Plant Engineering Licensing, Power Plant Engineering Department, Florida Power & Light Company (FPL). A statement of my professional background and qualifications is attached to this affidavit and made a part thereof.

My name is Habib H. Jabali. My business address is P. O. Box 529100, Miami, Florida, 33152. I am Manager of Civil Engineering, Power Plant Engineering Department, Florida Power & Light Company. A statement of my professional background and qualifications is attached to this affidavit and made a part thereof.

This affidavit addresses Contention 4A which states:

> There are likely to occur radioactive releases from one or more stored assemblies to unrestricted areas which violate 10 CFR Part 20 or are not as low as is reasonably achievable within the meaning of 10 CFR Part 50, as a result of:
>
> (a) substantial immersion of the steam generators in sea water during a hurricane;

8104230589

(b) movement of steam generators while so immersed;

(c) impact of such moving steam generators upon the walls of the structure in which they are stored or upon another object or objects;

(d) corrosion resulting from moisture, sea water, or salt spray; or

(e) leakage through the floor beneath the stored steam generators.

We are familiar with FPL's plans for storage of the lower assemblies.

The "NRC Staff Second Motion for Summary Disposition" moved, among other things, for the summary disposition of Contention 4A. There was included in the motion a "Statement of Material Facts as to Which There Are No Genuine Issues to be Heard" (pp. 6-9; variously hereafter "Statement of Material Facts" or "Staff's material facts") which contains statements applicable to each subpart of Contention 4A. With the inconsequential corrections referred to below, we agree with the Statement of Material Facts.

Contention 4A (a), (b), and (c)

Statement of Material Facts Nos. 1 through 9 address subparts (a), (b) and (c) of Contention 4A. The aspect of the contention addressed by these subparts presupposes that the steam generator lower assemblies (SGLAs) will become immersed by the flood associated with a hurricane. The Staff's material facts correctly demonstrate that the SGLAs will not float and that the steam generator storage compound (SGSC) will be so situated that the flood stage associated with a probable maximum hurricane (PMH) will not immerse the SGLAs. The discussion infra (i) provides additional technical support of the Staff's material facts, and (ii) demonstrates that SGLAs subjected to hurricane conditions pose no undue radiological hazard to

public health and safety.

Each stored SGLA will weigh about 185 tons and its specific gravity will exceed 1.7 (it will be about 2). The SGLAs will not float.

If it were postulated that the SGLAs were fully immersed in sea water, based on data for comparable environments, the corrosion effect of immersion for one full year would be 0.008 inches. Since a hurricane induced immersion would be measured in hours, not years, an immersion associated with a hurricane will give rise to much less than 0.008 inches. Thus corrosion resulting from immersion in sea water does not pose a threat to SGLA integrity.

The area underlying the proposed site of the SGSC has been demucked, and crushed limestone structural fill material was placed directly on the limestone bedrock in layers. Each layer was compacted before the next layer was set in place. The result is a compacted limestone plateau with a surface elevation of approximately +17.5 feet mean low water (MLW). The SGSC's structural footings will be set about 2 feet below the +17.5 feet MLW surface elevation of the limestone plateau.

Compacted crushed limestone is a proven structural fill material. The favorable structural characteristics result, in part, from the chemical bonding which occurs when oolitic limestone, which contains calcium carbonates, is crushed, graded, wetted and compacted. The recementation (chemical bonding) results in a firm monolithic mass which resists hydraulic forces since energy must be expended by the hydraulic forces to break loose particles of material for eventual water-borne migration. By contrast, compacted sands do not have any binding qualities and hence any initial hydraulic energy is immediately available for particle

migration (erosion). Erosion of compacted limestone is appreciably less than erosion of sand, since erosion of sand is based upon a mass consisting of individual particles of a nominal size that are free to be transported. Accordingly, because of the recementation process, erosion of the limerock fill during a PMH will not be sufficient to undermine the SGSC.

The Staff's material facts No. 5 indicates that there is about 65 feet of compacted limestone between the SGSC and the 3/1 slope leading to the +7 foot MLW elevation. The 65 foot number was apparently based on a preliminary SGSC location, which has been altered somewhat by the final SGSC location which is slightly west and south of the location evaluated by the Staff. The result of SGSC relocation is that about 180 feet of compacted limestone fill will lie to the east of the SGSC (the area most vulnerable to wave attack).

Based on the cement-like nature of compacted limerock, it can be concluded that PMH-induced erosion effects would not undermine the SGSC foundation. The large quantities of this material that surround the SGSC ensure building foundation integrity.

The SGLAs will be housed in a substantial reinforced concrete structure, the Steam Generator Storage Compound (SGSC). This structure will consist of 2'-0" thick reinforced concrete exterior walls with a 2'-0" thick reinforced concrete interior wall that separates the Unit 3 and Unit 4 SGLA storage areas. These walls will be founded on continuous strip footings located about 2 feet below finished grade. The SGSC roof will be watertight, consisting of precast concrete panels, with a minimum 4 inch reinforced concrete topping, resulting in a one-foot thick roof at a minimum. The floor will be reinforced concrete approximately 6" thick. Steel saddles will support each SGLA during storage.

The design of the SGSC complies with applicable design requirements of the Code of Metropolitan Dade County, Florida and the South Florida Building Code. The Code of Metropolitan Dade County, Florida requires that the elevation of buildings be above the 100 year hurricane flood level of 12 foot MSL. The South Florida Building Code specifies that buildings shall be designed for wind speeds of not less than 120 mph. The exterior wall thickness was established based on the radiological shielding requirements and not on the hurricane induced wind loadings which would have required less thickness. Thus, the structure has a higher resistance to wind loadings than would result from a design where the wind loading governed.

Hence, although not specifically designed for the PMH wind loadings, the SGSC has considerable capability to accommodate wind loadings that exceed those associated with the 120 mph design value. The PMH wind loadings will not cause stresses in structural members to exceed their respective yield stress values, and thus, a catastrophic collapse of the building is precluded.

In addition, prior to removal from the containment, SGLA exterior surfaces will be decontaminated as necessary to ensure that removable contamination will not be significant as defined by 49 CFR § 173.397 regarding surface contamination on radioactive shipments. Thus, the potential for radioactive effluent releases from external surfaces is effectively eliminated.

Also, the steam generators will be completely sealed to preclude the release of radioactive materials from the internal surfaces of the SGLAs. The SGLAs will have a minimum wall thickness of 2.63 inches, and the various openings in the steam generators will be closed by either steel plates or steel bar stock as follows:

(a) The large channel head and transition cone openings will have steel plates, three inches thick, placed over them and will be welded closed by a 1-1/2 inch fillet weld.

(b) The small nozzle connections (i.e., instrument taps, drains, and blowdown nozzles, all 2 inches and smaller) will have bar stock inserted into the openings and will be welded closed by a 1-1/2 inch fillet weld.

(c) The handhold openings will have steel inserts welded with a 1-1/2 inch fillet weld to the inside of the opening. The handhold cover plates will also be put back into place over the handholes.

Furthermore, the welds on the SGLAs will be coated with a protective coating to provide added protection against corrosion and additional assurance that oxygen will not enter the steam generators via the welds.

In conclusion, the SGLAs will be completely sealed to preclude the release of radioactive materials contained on internal SGLA surfaces; external surfaces will be decontaminated; and the PMH will not jeopardize the integrity of the stored SGLAs. Thus, there is no credible mechanism for a release of radioactivity from within the lower assemblies during storage as a result of the PMH, and there is no NRC regulation or any real-world radiological concern that would necessitate establishing a PMH design criterion for the SGSC. The discussion *infra* demonstrates that even if it is arbitrarily assumed that the radioactivity contained in the SGLAs were released to the hurricane storm surge, the potential consequences would be acceptable. It must be noted that the assumed release is an analytical assumption and is not expected to occur during the period of onsite SGLA storage.

Before discussing the postulated release, it is necessary to define the nature of the radioactive source contained in the stored SGLAs. The earliest time that a SGLA could be placed in the SGSC is 80 days after shutdown. Since the repair activities for Units 3 and 4 and placement of their respective SGLAs in storage are scheduled to commence about one year apart, the maximum curie content in the SGSC would occur about 445 days after commencement of repair activities on the first unit. At this time, the activity within three lower assemblies will have experienced 80 days of decay since shutdown and the three other lower assemblies will have experienced 445 days of decay since shutdown.

In assessing the potential radiological hazard associated with the stored SGLAs, the properties of the radioactivity must be defined and evaluated. Physical characteristics relevant to the evaluation are:

- The radioactive properties of the radionuclides present.
- The physical nature of the radioactive material.
- The solubility of the radionuclides present in water.
- The volatility of the radionuclides present.

The activity contained on the interior surfaces of the steam generators prior to removal is distributed as follows:

• Tubes	88.0%
• Tube Sheet	1.4%
• Tube Ends	5.3%
• Divider Plates	2.6%
• Channel Head	2.7%

The radioactivity in a SGLA consists almost entirely of activated corrosion products in the form of metal oxides, most of which are insoluble. These metallic oxides form a thin, tightly adherent layer on the primary side surfaces of the steam generator. Sea water has negligible ability to dissolve these corrosion products. During normal operation, the primary coolant flows through the steam generator's tubes at 24 feet/sec and 2235 psig and 546-602°F - - - the erosive effects of this flow do not remove this radioactive layer from the tubes. In fact, if one wants to remove the film, <u>i.e.</u>, decontaminate the SGLA interior surfaces, extreme measures must be used. These include blasting with oxide grit or soaking in harsh chemicals. Hence, it is difficult to envision any realistic mechanism during storage which could release more than a negligible amount of this oxide layer.

Table 1 provides a compilation of the isotopes present in the oxide layer. The activity at 80 and 445 days is indicated thereon for a single SGLA. The

point to note is that the total activity has fallen from about 220 curies to about 105 curies in a period of one year. This is due to the natural decay of the radionuclides contained in the oxide layer. The process is ongoing, thus, the potential radiological hazard diminishes with time. The table also shows the time for each radionuclide to reach 1/10, 1/100 and 1/1000 of its initial concentration. The overall decay of the SGLA radioactive source with time is depicted by Figure 1.

Examination of Table 1 and Figure 1 indicates that:

(1) The bulk of the nuclides essentially disappear in a year or two, and that only Mn-54, Ce-144, Cs-137 and Co-60 remain for appreciable periods of time.

(2) There is a rapid reduction in total activity in the SGLAs during the first year or so. Thereafter, the decay is characterized by the behavior of the longer lived isotopes, predominantly Co-60.

The only appreciable soluble isotopes remaining after 30 days are cerium, barium, ruthenium, and cesium. However, their contribution to the total number of curies is less than 1/2% at 445 days, and only Cs-137 is long-lived. There are no volatile radionuclides present.

When considering hurricane-related effects, the soluble isotopes are the isotopes of concern since they could be potentially dissolved and carried to the environment via the receding hurricane storm surge. The insoluble isotopes are not of concern because of the adherent nature of the activated corrosion products. For the sake of this evaluation, it will be assumed that 100% of the soluble isotopes in all six SGLAs would be released to the receding storm surge. We believe this assumption

to be conservative.

Before proceeding further, it should be noted that storms with surges greater than the 100-year hurricane flood level will result in more dilution of any postulated radiological release than that associated with the 100-year hurricane flood level. Therefore, from a radiological standpoint, the postulated release to the 100-year hurricane flood level is limiting - - - use of PMH flood conditions would yield more favorable results.

The 100-year flood results in a sufficient flood stage to transport the postulated release to Biscayne Bay, and since it is the limiting radiological hurricane flood level, it was selected as a basis for evaluation.

The result of this evaluation is that the release of 100% of the soluble isotopes contained in all six SGLAs at day 445 will rapidly be diluted by the turbulent waters created by the storm to inconsequential levels. Analysis shows that 2.5×10^8 ft^3 of water would be needed to dilute the release to within the levels of 10 CFR Part 20, Appendix B, Table II. Sufficient mixing of the release with the 3×10^{10} ft^3 of water in Biscayne Bay would occur within 17 hours to dilute the release to average concentrations within these levels.

This analytical evaluation can be extended further. If 100% of all (soluble and insoluble) isotopes contained in all six SGLAs at day 445 is assumed to be released, the conclusion is substantially the same. Within 30 hours, the available dilution volume of water in Biscayne Bay would be sufficient to dilute the entire radioactive inventory of the SGLAs to average concentrations within the levels of 10 CFR Part 20, Appendix B, Table II. Since hurricane conditions will prevail

during the dilution period, and since Biscayne Bay is not potable, and since it can be reasonably assumed that public use of Biscayne Bay during hurricane conditions is unlikely, any postulated release of SGLA radioactivity during a hurricane will not endanger the public health and safety.

Moreover, it must be reemphasized that the above-postulated release is entirely hypothetical. Since the steam generators will be completely sealed and housed in a substantial structure, and since there is no realistic way to release an appreciable fraction of their radioactive inventory, a hurricane poses no radiological hazard to public health and safety of consequence.

There is no factual basis to demonstrate that radioactive releases of any consequence can result from the landfall of a severe hurricane at, or in the vicinity of, the Turkey Point site. To the contrary, the facts demonstrate that the proposed design is proper and that it is in full compliance with NRC and local requirements.

<u>Contention 4A (d) and (e)</u>

Statement of Material Facts Nos. 10 through 22 address subparts (d) and (e) of Contention 4A. The aspect of the contention addressed by these subparts is premised on a corrosion-related mechanism releasing radioactivity to the environment. The Staff's material facts correctly demonstrate that there is no credible liquid pathway to the environment from the stored SGLAs. The discussion <u>infra</u> is provided to (i) further support and clarify the Staff's position, and (ii) demonstrate that the liquid or gaseous pathways pose no undue radiological hazard to public health and safety.

As dicussed supra, the SGLAs will be completely sealed and the external surfaces decontaminated prior to placing the SGLAs in storage. During the period of onsite storage, the SGLAs will not be unattended; a SGLA surveillance program will be in effect. The SGLA surveillance program will be comprised of quarterly visual inspections of the external surfaces of the lower assemblies, area radiation surveys, and random swipes of the welds sealing the covered openings in the lower assemblies.

Standard swipe techniques as described in 49 CFR § 173.397 will be used to assess removable contamination. 49 CFR § 173.389 (e) defines removable radioactive contamination as that radioactive contamination which can be readily removed in measurable quantities by wiping the contaminated surface with an absorbent material. The measurable quantity is not considered significant if it does not exceed the limits specified in 49 CFR § 173.397. If the contamination levels on the swipe of the SGLA exceed the appropriate limit, the external surfaces will be decontaminated until the measured activity on the swipe is within the limit, and the source of any leakage will be investigated.*

The SGLA surveillance program will periodically reaffirm the in situ integrity of the stored SGLAs during the period of onsite storage.

The only mechanism that could potentially breach the in situ integrity of the SGLAs during normal storage conditions would be corrosion related. But for the reasons cited infra, through-wall corrosion is not likely. Thus, there is no mechanism that could be expected to lead to a release of radioactive materials from the interior of the stored SGLAs.

---

*Statement of Material Facts No. 10 states that "the outside of the assemblies will be periodically decontaminated..." As stated above, they will be decontaminated only when swipes indicate it to be necessary.

Corrosion of the exterior of the SGLAs was evaluated assuming (1) normal atmosphere, (2) continuous condensation atmosphere, and (3) sea water spray conditions. Corrosion which may be expected on the interior of the SGLAs was evaluated assuming (1) dry atmosphere, and (2) the presence of residual reactor coolant water contained in plugged tubes of the SGLA.

The results of this evaluation indicate that (1) under normal atmospheric conditions, a total exterior surface corrosion of 0.03 inches is estimated for a 30-year storage period; (2) under conditions of continuous atmospheric condensation, a total exterior surface corrosion of 0.06 inches is estimated for a 30-year storage period; (3) sea water spray conditions are not expected inside the storage compound because the SGSC will be located at approximately elevation +17.5' MLW, because the SGSC will be an enclosed structure, and because of the distance of the SGSC from the shoreline; (4) there is insufficient oxygen available within the sealed SGLA to support oxidation (corrosion) of the steel to the extent that through-wall penetration could result from interior corrosion; (5) under dry conditions within the sealed steam generators, the corrosion is negligible; and (6) under conditions of residual reactor coolant water within the SGLA, the total uniform corrosion of the immersed portions of the inside surfaces of the steam generators is 0.025 inches, and of the unimmersed portions of the inside surfaces will be less than 0.005 inches.

The phenomenon of pitting corrosion was also evaluated. Pitting corrosion is a very localized corrosive attack that can manifest itself under stagnant conditions. The results indicate that (1) localized pit corrosion can be reasonably estimated to be 5 times the uniform corrosion rate; (2) the maximum depth of internal corrosion pits is estimated to be 0.125 inches for immersed surfaces and 0.025

inches for unimmersed surfaces, and (3) the rate of external corrosion pitting is estimated to be 0.005 inches per year for humid conditions and 0.01 inches per year for conditions of a continuous atmospheric condensation.

The corrosion evaluation reaffirms the Staff's material fact No. 18 relating to through-wall corrosion, viz, "the probability of a through-wall corrosion crack is so small as to be insignificant."

In order to assess analytically the potential consequences of a radioactive release from the stored SGLAs, a radioactive release was postulated. The evaluation that follows demonstrates that the stored SGLAs pose no radiological hazard to public health and safety. It should be emphasized that the release discussed _infra_ is an analytical assumption and is not expected to occur during the period of onsite SGLA storage.

The potential for airborne releases associated with SGLA storage does not exist since:

(1) For the reasons described _supra_, the level of contamination on the exterior surfaces of the SGLA will not be significant.

(2) The SGLAs will be completely sealed, thereby precluding an airborne release. However, even if it is assumed that an SGLA, as a result of corrosion, develops an airborne release pathway, the activity that could be released would not be in a volatile or aerosol form, _i.e._, in a form that could be dispersed as a gaseous effluent. The oxide layer in the SGLAs does not contain any radionuclides that will vaporize. Particulates that conceivably could

be postulated to leave the oxide layer would be too heavy to disperse to the environment as an airborne release.

The occurrence of a radioactive release to the environment via the liquid release pathway is very unlikely since;

(1) For the reasons described supra, a mechanism that could cause a breach in the integrity of the sealed SGLA is not present.

(2) Only a small fraction of the SGLA inventory is soluble in water and thus, potentially available for transport via the liquid pathway.

(3) Liquids released to the SGSC would likely be retained by the reinforced concrete floor of the SGSC.

If an SGLA leak is postulated, and the SGSC reinforced concrete floor is assumed to be cracked, and it is assumed that the leakage passes through approximately 18 feet of compacted limerock fill and limestone bedrock, then the leakage would enter the aquifer that underlies the site, which at that location is saline and non-potable.

While any of the four Turkey Point units are in operation, the movement of ground water beneath the site of the SGSC is controlled by the Turkey Point Cooling Canal System. Ground water movement is controlled by the differential operating head maintained between the intake approach channel and the discharge basin of the cooling canal system. Operation of the cooling system maintains the water level in the intake channel at a level slightly below the water table, so that it behaves as a discharge sink for the adjacent ground water. At the same time, water in the discharge channel is maintained slightly above the water table, so that it

behaves as a source of recharge to the adjacent ground water. The result is that ground water beneath the SGSC site moves to the intake channel.

In the highly unlikely event that none of the four Turkey Point units are operational, ground water movement beneath the site of the SGSC would be controlled by the natural ground water gradient, which is generally away from land. Any leakage under this condition would tend to move to the intake channel, as described above.

It should be noted that the cooling system-dominated ground water regime would result in movement of the postulated leakage to the cooling canal system more rapidly than the transit time of several years cited in the Staff's material fact No. 15. This is due to the hydraulic gradient imposed on the natural ground-water gradient by operation of the cooling system. As shown _infra_, the transient time is inconsequential since the postulated leakage will be diluted in the cooling canal system to inconsequential levels.

Should it be conservatively assumed that 100% of the solubles were simultaneously released from all six of the SGLAs at day 445 after shutdown of the first unit to be repaired, a volume of 2.5 X 10^8 ft^3 of water would be required to dilute this hypothetical release to within the NRC limits contained in Appendix B, Table II of 10 CFR Part 20 (commonly known as "drinking water" limits). The water normally available in the cooling canals is 7 X 10^8 ft^3. Thus, there is more than ample water available to dilute a postulated release to inconsequential levels; levels that are within 10 CFR Part 20, Appendix B, Table II.

To place the values of 10 CFR Part 20 into perspective, compare the drinking water guideline value of 0.02 pCi/ml for the only long-lived soluble isotope contained in the SGLAs, Cs-137, to the naturally occurring radioactivity in a few common substances. The naturally occurring radioactive content of tea is 0.4 pCi/ml, of cereal is 0.6 pCi/ml, and of human blood is 0.9 pCi/ml. Since any assumed leakage will enter the cooling canals, and since the cooling canals are saline and unfit for human consumption, and since the analytically postulated leak results in concentrations in the cooling canals that are within the commonly called "drinking water" standard of 10 CFR Part 20, and since the resulting concentrations are comparable to the natural radioactivity found in common substances, it is clear that a postulated radioactive leak from the SGSC poses no undue radiological hazard to public health and safety.

If one addresses the real-world effects of SGLA storage, the material facts indicate:

(1) There is no potential for airborne radioactive release from the SGSC.

(2) There is no reasonable mechanism for a liquid radioactive release from the SGSC.

Storage of the SGLAs in the proposed SGSC poses no radiological hazard of consequence to public health and safety.

FURTHER AFFIANT SAYETH NOT

Date: April 16, 1981

Frederick G. Flugger

STATE OF FLORIDA)
) SS.
COUNTY OF DADE)

SWORN to and subscribed before me this sixteenth day of April, 1981.

Notary Public

NOTARY PUBLIC STATE OF FLORIDA AT LARGE
MY COMMISSION EXPIRES DEC. 26 1981

FURTHER AFFIANT SAYETH NOT

Date: April 16, 1981

H. H. Jabali

STATE OF FLORIDA)
) SS.
COUNTY OF DADE)

SWORN to and subscribed before me this sixteenth day of April, 1981.

Notary Public

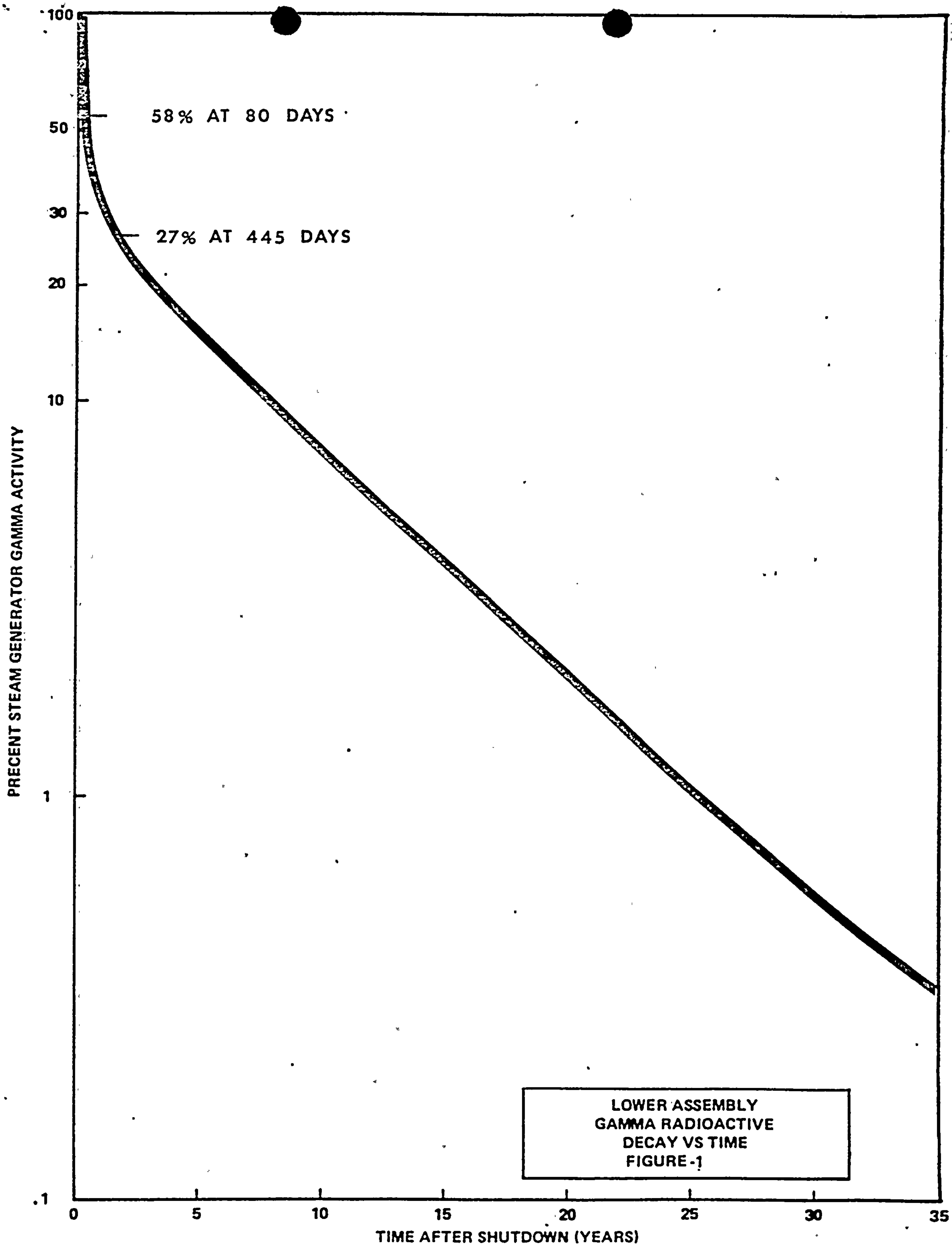
NOTARY PUBLIC STATE OF FLORIDA AT LARGE
MY COMMISSION EXPIRES DEC. 26 1981

TABLE 1

SGLA CONTENTS

ISOTOPE	ACTIVITY (Ci)		HALF-LIFE	TIME TO REACH		
	80 day decay	445 day decay		$\frac{1}{10}$	$\frac{1}{100}$	$\frac{1}{1000}$
Cr-51	.966	*	27.7 days	92d	184d	276d
Mn-54	8.99	4.01	312.5 days	1038d	2076d	3114d
Fe-59	.709	.002	44.6 days	148d	296d	444d
Co-58	62.98	1.771	70.8 days	235d	470d	705d
Co-60	92.99	81.54	5.27 years	17.5y	35y	52.5y
Zr-95	5.01	.097	64.0 days	212d	425d	638d
Nb-95	9.063	.210	35.0 days	116d	232d	349d
Nb-95m	.048	.001	87.0 hours	116d	232d	349d
Mo-99	*	0	66.0 hours	219h	438h	658h
Tc-99m	*	0	6.02 hours	219h	438h	658h
Rv-103	1.951	.0032	39.4 days	131d	262d	392d
Xe-131m	*	0	11.74 days	38.9d	77.9d	116.9d
I-131	.002	0	8.06 days	26.0d	53d	80d
I-132	*	0	2.28 hours	7h	15h	22h
Te-132	*	0	78 hours	259h	518h	777h
Cs-137	.358	.350	30.2 years	100.3y	200.7y	301y
Ba-137m	.339	.332	2.55 minutes	100.3y	200.7y	301y
Ba-140	.031	0	12.8 days	42d	85d	127d
La-140	.036	0	40.3 hours	42d	85d	127d
Ce-141	.722	*	32.5 days	108d	216d	324d
Ce-144	16.2	6.667	284.8 days	945d	1889d	2834d
Pr-144	16.2	6.667	17.3 minutes	945d	1889d	2834d
Np-239	*	0	2.35 days	7.8d	15.6d	23.4d

*Activity less than 10^{-3}



LOWER ASSEMBLY
GAMMA RADIOACTIVE
DECAY VS TIME
FIGURE-1

STATEMENT OF PROFESSIONAL QUALIFICATIONS

FREDERIC G. FLUGGER

EXPERIENCE

1973 to present — Florida Power & Light Company:

Manager, Plant Engineering Licensing, Power Plant Engineering Department for Florida Power & Light Company. Responsible for nuclear licensing, fossil licensing, quality assurance and administrative procedures. Direct engineering activities of FPL, the architect engineer and other consultants required to secure permits and licenses, to respond to NRC and other government agencies, and to select sites for the Company's generating facilities. Provide expert testimony at public hearings and to regulatory bodies; engineering evaluations required by regulatory bodies; safety evaluations for modifications to operating nuclear units; independent design review; QA training of engineering personnel; and develop and maintain licensing related computer codes.

Project Engineer, St. Lucie Units 1 and 2, for Florida Power & Light Company. Directed internal and external engineering efforts for two 810 Mw nuclear units (PWR). Numerous presentations to NRC and ACRS, and testified at the St. Lucie Unit #2 public hearing. Provided technical direction to Architect Engineer, Nuclear Steam System Supplier and technical consultants.

1972-1973 — NUS/Southern Nuclear Engineering:

Developed and managed the NUS Licensing Information Service. Responsible for the development of a 500 Mw nuclear units' (PWR) hot functional test program.

1970-1972 — Long Island Lighting Company:

Assistant to the Manager of Nuclear Projects for Long Island Lighting Company. Responsible for the licensing, safety, nuclear related design and fuel management aspects of the Shoreham Nuclear Project (819 Mw BWR). A member of the Company's Engineering Assurance Review Committee. Extensive involvement in the Shoreham public hearing.

1964-1970 Consolidated Edison Company of New York, Inc.:

Manager of Nuclear Bureau of Consolidated Edison Company of New York. Directed the Company's Nuclear and Reactor Fuels Divisions. Responsible for AEC licensing, shielding design, accident analysis, reactor plant safety, nuclear computer code development and fuel cycle engineering. Member Nuclear Facilities Safety Committee.

Engineer, Nuclear Division for Consolidated Edison Company of New York. Directed development of safety analysis reports and technical specifications; engineering associated with engineered safety features design, containment leak rate testing, urban nuclear plant siting studies and shielding design; and offsite radiological dose evaluations. Instructor for the Company's Power Reactor Technology course.

Engineer for Consolidated Edison Company of New York. An engineer in the Control and Instrument Division and at the Company's Ravenswood generating station.

1963-1964 U. S. Merchant Marine:

Licensed Engineering Officer aboard U.S. Merchant vessels. In charge of maintenance and operation of marine steam-turbine power plants and indirect mechanical refrigeration systems.

1962-1963 University of Maryland:

Teaching Assistantship in the Mechanical Engineering Department, University of Maryland. Taught undergraduate courses in Thermodynamics, Strength of Materials and Engineering Graphics.

EDUCATION State University of New York Maritime College; Bachelor of Marine Engineering, with Honors, 1960.

University of California at Berkeley; Master of Science in Engineering Science, Atomic Energy Commission Science and Engineering Feollow, 1961.

Specialized graduate work in Advanced Mathematics, Heat Transfer, Reactor Theory, Reactor Kinetics, and Nuclear Fuel Management (42 credits), 1962-1968.

MIT - Nuclear Reactor Safety, 1966.

NUS - Advanced Nuclear Fuel Management, 1970.

Florida Power & Light - Management Development Courses, 1975 - Present.

MEMBERSHIP/ LICENSES

American Nuclear Society.

American Society of Mechanical Engineers (ASME).

ASME Committee on Nuclear Quality Assurance, 1977 - Present; Vice Chairman, Design Control Work Group, 1980 - Present.

Third Assistant Marine Engineer, Steam and Diesel.

Atomic Industrial Forum, Probablistic Risk Assessment Subcommittee, 1981 - Present.

PUBLICATIONS

1. F. G. Flugger, et. al., "Turkey Point Units 3 & 4 Steam Generator Repair-Licensing Evaluation", American Nuclear Society Winter Meeting, November 1978.
2. F. G. Flugger, J. N. Burford, "A Fire Hazard Evaluation for Operating Nuclear Plants", American Nuclear Society Winter Meeting, December 1977.
3. Masters Thesis, "Associated Particle Method for Measurement of the Absolute Neutron Flux Produced by the T (d,n) α Reaction", University of California at Berkeley, 1961.

STATEMENT OF PROFESSIONAL QUALIFICATIONS

HABIB H. JABALI

<table>
<tr><td>EXPERIENCE</td><td>Florida Power & Light Company - Miami, Florida:</td></tr>
<tr><td>1976
to
present</td><td>Manager, Plant Civil Engineering, Power Plant Engineering Department. Responsible for overall plant Civil Engineering, both internal and external to FPL, for nuclear and fossil generating stations. Also responsible for monitoring plant construction to trouble shoot and resolve technical problems as they are identified to ensure that building codes and civil standards are met. Plan, organize and manage the Civil section to ensure most effective use of available manpower. Interview prospective employees, select, develop and assign personnel and oversee their training to ensure technical competence to meet civil engineering objectives.

Supervisor, Plant Civil Engineering, Power Plant Engineering Department. Establish scope and design criteria of power plants in the civil, structural and architectural areas and direct the engineering activities of architect-engineers and other consultants to ensure implementation thereof. Responsible for supervising the Civil Engineering Section personnel in preparing design modifications and additions at operating power plants and in providing technical assistance in the civil engineering areas to other FPL departments.

Senior Civil Engineer, Power Plant Engineering Department. Responsible for the review and approval of design, procurement, bid reviews and resolution of construction problems for all civil/structural related activities for St. Lucie Unit 1 and 2 Nuclear Power Plants.</td></tr>
<tr><td>1980
to
present</td><td>Lecturer in civil engineering, University of Miami, Department of Civil Engineering.</td></tr>
<tr><td>1971 - 1976</td><td>Sargent & Lundy Engineers - Chicago Illinois

Senior Structural Engineer, Responsible for developing and coordinating structural standards pertaining to the design of power plants. Review and evaluate state of the art technical papers, studies and reports and determine their impact on on-going power plant design. Supervised the design of the Turbine Building Complex for the Bailly Nuclear Power Plant. Directed and supervised the design of major modifications and additions to several existing power plants. This included the design of structures supporting Electrostatic Precipitators, air and gas ducts, chimneys and ID and FD fan foundations. 1971 - 1973, Design Engineer. Worked on the design of various buildings and structures within power plant complex, such as: Auxiliary Building, Reactor Building, Fuel Handling Building, etc. This included steel as well as reinforced concrete designs for both Category I and non-Category I structures.</td></tr>
</table>

1970 - 1971 — Research Assistant, Northwestern University. This research dealt with the behavior of structures under dynamic loads and was sponsored by a grant from the U.S. Army. The results led to the development of an improved mathematical model for soil-structure interaction and the selection of dynamic stiffness of foundations.

1968 - 1969 — Research Assistant, University of Miami. Duties consisted mainly of analyzing and developing solutions to stress concentrations around cracks in composite members. This research was sponsored by a grant from the National Science Foundation.

EDUCATION

University of Miami, Coral Gables, Florida. Graduated in 1968 with a Bachelor of Science in Civil Engineering, Cum Laude and Departmental Honors.

Graduated in 1970 with a Master of Science in Civil Engineering.

Northwestern University, Evanston, Illinois. Graduated in 1971 with a Master of Science in Theoretical & Applied Mechanics.

University of Miami, Coral Gables, Florida. Currently preparing for Doctoral Dissertation in Structural Mechanics.

MEMBERSHIPS

Professional Engineer in the states of Florida and Illinois.

Member of American Concrete Institute.

Member of American Society of Mechanical Engineers.

Member of American Institute of Steel Construction.

Academic Honorary - Phi Eta Sigma
Pi Mu Epilson
Pi Delta Phi

PUBLICATIONS

1. "Torsional Oscillations of a Layer Bonded to an Elastic Half-Space", INTERNATIONAL JOURNAL OF SOLIDS & STRUCTURES. Vol. 10, January 1974, pp. 1-13.

2. "Torsional Stiffness of Layer Bonded to an Elastic Half-Space", INTERNATIONAL JOURNAL OF SOLIDS & STRUCTURES. Vol. 11, 1975, pp. 1219-1221.

3. "Stresses in Concrete Chimneys Weakened by Openings", JOURNAL OF THE AMERICAN CONCRETE INSTITUTE, Proceedings Vol. 73, No. 8, August 1976, PP. 465-468.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSON

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

Docket Nos. 50-250-SP
50-251-SP

IN THE MATTER OF)
FLORIDA POWER & LIGHT COMPANY)
(Turkey Point Nuclear Generating)
Units Nos. 3 and 4)

(Proposed Amendments to Facility Operating License to Permit Steam Generator Repairs)

CERTIFICATE OF SERVICE

I HEREBY CERTIFY that copies of Licensee's Answer Supporting NRC Staff Motion for Summary Disposition of Contention 4a and Affidavit of Frederick G. Flugger and H. H. Jabali on Contention 4a were served on the following by deposit in the United States mail, first class, properly stamped and addressed, on the date shown below:

*Marshall E. Miller, Esq., Administrative Judge
Chairman, Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Dr. Emmeth A. Luebke, Administrative Judge
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Dr. Oscar H. Paris, Administrative Judge
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Mark P. Oncavage
12200 S.W. 110th Avenue
Miami, Florida 33176

Harold F. Reis, Esq.
Steven P. Frantz, Esq.
Lowenstein, Newman, Reis & Axelrad
1025 Connecticut Avenue, N.W.
Washington, D.C. 20036

*Steven C. Goldberg, Esq.
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

*Burt Saunders, Esq.
Assistant Dade County Attorney
1626 Dade County Courthouse
Miami, Florida 33130

*Henry H. Harnage, Esq.
Peninsula Federal Building
10th Floor
200 S.E. First Street
Miami, Florida 33131

*Neil Chonin, Esq.
1400 AmeriFirst Building
One Southeast Third Avenue
Miami, Florida 33131

STEEL HECTOR & DAVIS
Co-counsel for Licensee
1400 Southeast First
National Bank Building
Miami, Florida 33131
Telephone (305) 577-2863

By Norman A. Coll
Norman A. Coll

April 17, 1981

*Additional Service By
Hand or Courier