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 DR. GERALD R. KILP
ON ADMITTED CONTENTION 3 (KILP AFFIDAVIT 3b)

I, Gerald R. Kilp, being duly sworn, say as follows:

1. I am an Advisory Engineer in the Engineering Department of the Commercial Nuclear Fuel Division of the Westinghouse Electric Corporation. My business address is Westinghouse Electric Corporation, Monroeville Mall Office Building, P. O. Box 3912, Pittsburgh, PA 15230. 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 3.

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2. Admitted Contention 3 (originally Amended Petition Contention 6) states:

> The Licensee and Staff have not adequately considered or analyzed materials deterioriation or failure in materials integrity resulting from the increased generation of heat and radioactivity as a result of increased capacity in the spent fuel pool during the storage period authorized by the license amendment.

The bases for Admitted Contention 3 were stated as follows:

> The spent fuel facility at the St. Lucie Plant, Unit No. 1, was originally designed to store a lesser amount of fuel for a short period of time. Some of the problems that have not been analyzed properly are:
>
> a. Deterioration of fuel cladding as a result of increased exposure and decay heat and radiation levels during extended periods of pool storage.
>
> b. Loss of materials integrity of storage rack and pool liner as a result of exposure to higher levels of radiation over longer periods.
>
> c. Deterioration of concrete pool structure as a result of exposure to increased heat over extended periods of time.

3. The purpose of my affidavit is to demonstrate that the impacts of radiation and heat on the materials used in the fuel cladding and assemblies have been adequately considered. The impacts of radiation and heat on (1) the spent fuel pool liner and storage racks (as to the non-Boraflex materials) and (2) the concrete pool structure are considered in the Affidavits of myself, (Kilp Affidavit 3a) and Murry Weber, respectively, offered in support of "Licensee's Motion for Summary Disposition of Intervenor's Contentions," regarding Admitted Contention 3. The impacts of radiation and heat on the integrity of Boraflex, used in the storage racks, are discussed in the Affidavit of Dr. K. P. Singh, offered in support of "Licensee's Motion for Summary Disposition of Intervenor's Contentions," regarding Admitted Contentions 3 and 6.

Materials Used in Fuel Cladding And Assemblies

4. The materials used for the fuel assembly structure and for the fuel cladding were specifically selected for use in reactors because of their exceptional resistance to corrosion in high temperature water and/or steam. They were also selected for their resistance to deleterious property changes in the high radiation fields characteristic of nuclear reactors. Zircaloy is used for the cladding and end plugs of the fuel rods (or pins) and as part of the fuel assembly structures. Type 304 stainless steel, Type 304L stainless steel, Inconel 718, Inconel X-750, and Type CF-3 (304 stainless steel) are also utilized in the remainder of the fuel assembly structure.

Radiation Levels

5. Four types of radiation (alpha, beta, gamma, and neutron) will be present in the spent fuel pool to varying degrees. For materials such as the above, the alpha and beta radiation are no concern because they cannot penetrate the cladding surface to any significant depth and hence will not have any effect on the cladding outer surface or the other assembly materials. Gamma radiation, while it is penetrating compared to alpha and beta radiation, does not have any material effects on metals and alloys such as those comprising the fuel assembly and the fuel rod cladding. Its primary effect would be minor heating, amounting to only a few degrees at most compared to the water temperature in the pool.

6. Virtually all of radiation induced changes to the properties of the fuel assembly and clad materials can be attributed to fast neutrons. Any incremental property changes from neutrons caused by placing the fuel assemblies closer together can best be gauged by comparisons of reactor neutron fluences and pool storage fluences. Upon discharge from the reactor, a typical fuel assembly would have

accumulated a total fluence (or exposure) of about 10^{22} fast neutrons/cm^2. At the very low fast neutron fluxes in the spent fuel pool, an assembly would incur an estimated maximum additional fluence of less than 5×10^{15} neutrons/cm^2 as discussed in the Affidavit of Dr. Stanley E. Turner. The difference between the reactor neutron fluence and the highest postulated spent fuel pool fluence is several orders of magnitude. Putting it another way, the added neutron exposure after 40 years in the spent fuel pool would be equivalent to less than two minutes in the reactor while the reactor is at full power.

7. From the foregoing, it is evident that any radiation effects on fuel cladding and fuel assembly materials attributable to increasing the capacity of the spent fuel pool is negligible compared to prior reactor exposure. These materials are virtually immune to changes due to alpha, beta and gamma radiation, and the fast neutron levels in the spent fuel pool (even over 40 years or more) are so small as to be inconsequential. Hence, there is no threat to materials integrity caused by the reracking due to the minor increases in radiation compared to the original pool license.

Heat Load

8. The fuel assemblies and fuel cladding were designed and use materials to withstand the temperatures and heat loads present in the reactor. These are far more severe than those present in the spent fuel pool. Commercial reactors generally operate at coolant outlet temperatures in the 500 to 640 degrees Fahrenheit range which is to be compared to the short time maximum temperature associated with pool boiling which might occur after a loss of cooling event.

9. The Zircaloy alloy used for the fuel cladding and certain of the fuel assembly components has a very low corrosion rate, at the highest predicted spent fuel pool temperatures. At 500 degrees Fahrenheit, the corrosion rate of Zircaloy in water or steam is approximately 1/100,000 inches per year. At this rate, it would take over 100 years to corrode 1/1000 of an inch of the Zircaloy surface.

This amount of corrosion is well under ten percent of the Zircaloy thicknesses remaining after discharge from the reactor. At the much lower temperatures actually predicted under normal conditions for the spent fuel pool, the expected corrosion rate for Zircaloy would be at least an order of magnitude lower than at 500 degrees Fahrenheit.

10. High levels of hydriding can lead to embrittlement of Zircaloy, and is a direct function of corrosion in a water or steam environment. Since the corrosion rate is virtually nil as stated above, the hydriding rate in the pool will be nil also. The Zircaloy is also considered to be virtually immune to stress-corrosion cracking in spent pool water, so there is no known threat to the integrity of Zircaloy.

11. Similar assessments were made of the other fuel assembly materials (e.g., Type 304 stainless steel, Inconel X-750, and Inconel 718). All of these materials have been shown by test and experience to be virtually immune to corrosion at spent fuel pool temperatures and have at least as good high temperature corrosion rates as Zircaloy. Type 304 stainless steel would not be expected to corrode more than 6/10,000 inches after 100 years in a oxygen borated water environment like that of the spent fuel pool. Corrosion rates for the Inconels are at least as low as that for Type 304 stainless steels. Additionally, since stainless steel, Inconel and Zircaloy all form protective oxide films, no significant galvanic attack is expected among these materials.

12. The possibility of stress-corrosion cracking of sensitized Type 304 stainless steel adjacent to welds in the fuel assembly structure has been considered. Numerous studies of the performance of Type 304 stainless steel in fuel storage pools have been performed by organizations in the United States and throughout the world. From these studies, there is no evidence that stress corrosion cracking of stainless steel or the other assembly components is occurring to any significant degree. Even if localized stress-corrosion cracking were assumed to occur, it would not affect the fuel pins. The fuel assembly structure materials do not exhibit significant hydrogen pickup as they corrode.

Conclusions

13. Based on the preceding discussions, it was shown that the radiation fields resulting from the storage of reactor fuel will not significantly affect the fuel cladding or the fuel assembly materials. Further, there is no corrosion threat to cladding or fuel assembly materials. Hydriding and stress-corrosion cracking were also considered and judged not be of concern.

14. Support for the above conclusions is drawn from the extensive experience accumulated all over the world in safely storing nuclear fuel in pools similar to the St. Lucie 1 pool. Zircaloy clad fuel has been examined after nearly 21 years of pool storage with no detectable changes in corrosion film thickness, cladding hydrogen content, and cladding mechanical properties. Zircaloy clad fuel has actually run in a pressurized heavy-water reactor for almost 22 years and continued to operate satisfactorily. The performance of other fuel and fuel assembly material such as Type 304 stainless steel has also been evaluated after many years in storage pools with no discernible changes due to the heat and radiation present in the pools.

15. I, therefore, conclude that since there are no fuel assembly or fuel cladding material degradation concerns, these materials can be safely used in the St. Lucie 1 fuel storage pool from the present to the expiration of its operating license on March 1, 2016.

FURTHER AFFIANT SAYETH NAUGHT

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

Gerald R. Kilp

Gerald R. Kilp

STATE OF PENNSYLVANIA)

COUNTY OF ALLEGYENY)

Subscribed and sworn to before me this 22ND day of July, 1988. My commission expires 12-14-91.

Lorraine M. Piplica

Notary Public

LORRAINE M. PIPLICA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES DEC. 14, 1991
Member, Pennsylvania Association of Notaries

EXHIBIT A

STATEMENT OF PROFESSIONAL QUALIFICATION OF
GERALD R. KILP

My name is Gerald R. Kilp. My business address is Westinghouse Electric Corporation, P. O. 3912, Pittsburgh, Pennsylvania, 15230. I am an Advisory Engineer for the Fuel Performance Engineering Section of the Westinghouse Commercial Nuclear Fuel Division, Westinghouse Electric Corporation. I have served in this function since November, 1983. In this capacity, I am responsible for selected Materials Development programs and act as a general advisor on materials performance questions for the Westinghouse Nuclear Fuel Divison.

I graduated from Missouri Valley College, Marshall, Missouri, in 1952 with a Bachelor of Science degree in Chemistry. In 1957, I received a Doctorate of Physical Metallurgy from Iowa State College (since renamed to Iowa State University).

From 1952 to 1957, I was a Graduate Assistant at the Ames Laboratory for Atomic Research, an AEC supported laboratory at Iowa State College. During this period, I worked on binary phase diagrams and evaluated methods for protection of uranium metal from water corrosion.

From December, 1957 to May, 1962, I was a Senior Engineer and, later a Fellow Engineer, at the Westinghouse Atomic Power Department where I worked on thermoelectric and thermionic materials for application in nuclear reactors.

In May, 1962 and until September, 1968, I acted as supervisor and later Manager of Fuel Evaluation on the NERVA Reactor Project at the Westinghouse Astronuclear Laboratory at Large, Pennsylvania. In September, 1968, and until May, 1972, I served as the Engineering Manager of the Astronuclear Fuel Facility at Cheswick, Pennsylvania. In this capacity, I was responsible for process development for fabrication of NERVA reactor fuel as well as reactor fuel performance evaluation.

In May, 1972, I transferred to the Westinghouse Nuclear Fuel Division of Westinghouse Nuclear Energy Systems, in Monroeville, Pennsylvania. From then to May, 1980, I served as the Manager of Materials Design. This group had the basic responsibility for materials R&D, and approval of materials for use in Westinghouse Pressurized Water reactors. The duties further included determination of the necessary and sufficient requirements for reactor coolant and pool storage chemistries needed to assure satisfactory performance under all warranted conditions. All reactor and out-reactor corrosion testing evaluations were done under the cognizance of this group.

From May, 1980, and until November, 1983, I worked at the Westinghouse Advanced Energy Systems Division where I served as the Manager of Materials Interactions. These activities were primarily concerned with addressing materials selection and evaluation for application in long term storage of light water reactor fuel in underground and above-ground facilities.

Since 1979 I have also been a member of the American Society for Testing and Materials (ASTM) C26 Committee on the Nuclear Fuel Cycle. At the present time I am the Chairman of Sub-committee C26.02 (Fuel and Fertile Materials Specifications), serve on C26.03 (Neutron Absorber Materials Specifications) and am 2nd Vice-chairman of the C26 Main Committee.