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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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ATOMIC SAFETY AND LICENSING BOARD

B. Paul Cotter, Jr., Chairman
Glenn O. Bright
Dr. Richard F. Cole

SERVED MAY 10 1989

In the Matter of:	Docket No. 50-335-OLA
FLORIDA POWER & LIGHT COMPANY	(ASLBP NO. 88-560-01-LA)
(St. Lucie Plant, Unit No. 1)	May 9, 1989

INITIAL DECISION
(Authorizing Spent Fuel Pool Reracking)

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I. INTRODUCTION

Licensee, Florida Power & Light Company, applied for and received[1] a license to rerack the spent fuel pool at its St. Lucie Unit 1 plant. Staff Exhibit 1. The reracking enabled licensee to increase the spent fuel storage capacity from 728 to 1706 fuel assemblies by reracking the spent fuel pool into two discrete regions using new, high density storage racks.

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[1] On March 11, 1988 pursuant to 10 C.F.R. § 50.91(a)(4) (1988), the Nuclear Regulatory Commission Staff made a finding of "no significant hazard consideration", approved the high density reracking, and issued Amendment 91 to License Number DPR-67 authorizing the modification to the spent fuel pool.

Campbell Rich, a nearby resident ("Mr. Rich" or "Intervenor"), challenged the reracking, contending that specific aspects of Licensee's plan would not adequately protect the public health and safety. Of Intervenor's seven contentions originally admitted, Florida Power & Light Co., 27 NRC 452 (LBP-88-10A, 1988), aff'd., 27 NRC 627 (ALAB-893, 1988), one was dismissed at the request of the Intervenor, and all of four and parts of two additional contentions were dismissed by this Board in a ruling on Licensee's motion for summary disposition. Florida Power & Light Co., 28 NRC 455 (LBP-88-27, 1988).

In the modified contentions remaining at issue, Mr. Rich asserts that the safety of the reracked spent fuel pool is not assured because of uncertainties in the effectiveness of Boraflex (a reactivity inhibitor), the risk of an accident resulting from the possible mishandling of fresh fuel rods, and the possibility of a criticality accident in the absence of a neutron moderator in the spent fuel pool. The foregoing issues were tried in a three-day hearing in the Martin County Courthouse, Stuart, Florida, beginning on January 24, 1989.[2]

In considering whether the license amendment granted by the NRC Staff may remain in effect, we must determine, for each of the factual issues remaining in dispute, whether the preponderance of the evidence supports the Licensee's position. See Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-763, 19 NRC 571, 577 (1984), review declined, CLI-84-14, 20 NRC 285 (1984). For the reasons set forth within, we find that Licensee is entitled to judgment on all these contentions subject to the condition we impose as to the use of Boraflex. Anything in the record not expressly addressed in this decision is rejected as unsupported by the record as a whole or as unnecessary to reaching our decision.

[2] The parties completed post hearing filings on March 27, 1989. Staff and Applicant filings suggested corrections to the transcript. Those accepted by the Board are attached hereto as Appendix A.

II. THE SPENT FUEL POOL CONFIGURATION AND OPERATION

For clarity it is worth reiterating some aspects of an earlier description of the configuration and operation of the spent fuel pool ("pool") at issue in this proceeding. Florida Power & Light Co., 28 NRC 455, 457-459 (LBP-88-27, 1988). The pool is adjacent to Unit 1 of the St. Lucie nuclear power plant which is owned and operated by Florida Power & Light Company on Hutchinson Island in St. Lucie County, Florida. The St. Lucie plant contains two units and is sited 12 miles south of Fort Pierce on the east coast of Florida.

A. General Configuration

The spent fuel pool is 37 feet long, 33 feet wide, and 40 feet, six inches deep. It is constructed of 6 feet thick reinforced concrete walls and a reinforced concrete floor and foundation mat 9 feet 6 inches thick. The floor and walls are lined with stainless steel, 1/4-inch thick on the floor and bottom of the walls and 3/16-inches thick on the remainder of the walls.

A separate but adjacent fuel cask storage area is located in the northeast corner of the spent fuel pool. It is 10 feet long and 12 feet wide. Its floor is a depression in the base mat which is 3 feet 6 inches deep, lined with 1-inch thick stainless steel plate. The walls are lined with 1/2 inch stainless steel plate. The cask storage area is separated from the fuel storage area by steel plate walls 6-7/8 inches thick, 14 feet 9 inches high, and lined with 1/4 inch stainless steel. This requires that the fuel cask must be raised above the top of the stored fuel before the cask can be moved laterally. The spent fuel cask weight is limited to 25 tons.

The fuel assembly structures containing the spent fuel to be stored in the pool are made of stainless steel and inconel. The fuel rod cladding is Zircaloy. These materials were selected because of their resistance to harmful changes in their properties resulting from: (1) high radiation fields in nuclear reactors; and (2) their exceptional resistance to corrosion in high temperature water and steam. The assemblies were designed and constructed to withstand the high temperatures experienced in nuclear reactor vessels (500° to 640° Fahrenheit ("F") at the coolant outlet). Vessel or core temperatures are far more severe than those normally encountered in spent fuel pools which are well below the boiling temperature of water, 212°F at atmospheric pressure. The fuel assemblies are stored in storage racks resting under water on the bottom of the spent fuel pool.

B. The Reracked Spent Fuel Pool

As noted, the amendment authorized Licensee to increase the spent pool capacity from 728 to 1706 fuel assemblies. The old storage racks were removed. The pool, as reracked with new, high density racks, is divided into two discrete regions, identified as Regions 1 and 2, each with its own specially designed racks. Region 1 contains 4 rack modules with capacity for 342 fuel assemblies. It is designed to receive and store new assemblies up to 4.5 weight percent U-235 or spent fuel that has not achieved adequate "burnup" (i.e., U-235 depletion) for storage in Region 2. Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to the Reracking of the Spent Fuel Pool at the St. Lucie Plant, Unit No. 1 as Related to Amendment No. 91 to Unit 1 Facility Operating License No. DPR-67, Florida Power and Light Company, Docket No. 50-335, at p. 2 (hereinafter "SER-Amendment 91". The foregoing document is attached to License Amendment 91 for the St. Lucie Plant.

The essential difference between Region 1 and Region 2 storage rack modules is that the Region 1 racks are provided with additional neutron absorbing material in the form of Boraflex so as to control the higher potential reactivity that would result with fresh nuclear fuel. The Region 1 racks consist of stainless steel, square cross-section tubes equipped with a sheet of Boraflex and cover plate on each of its four sides. The spacing between assemblies in Region 1 is 10.12 inches. SER-Amendment 91, pp. 2 and Appendix A, pp. 39, 40.

Region 2 contains 13 rack modules with capacity for 1364 fuel assemblies. The spacing between assemblies is 8.86 inches and Boraflex panels are sandwiched between channels. The Region 2 channels do not have cover plates, and the Boraflex panels are held in place by the mating of adjacent channels. Id., Appendix A, pp. 41, 42. Region 2 racks with their slightly closer spacing and about 50 per cent of the Boraflex neutron shielding material contained in Region 1 racks, are designed to receive and store spent fuel which meets fuel burnup requirements. The burnup requirements depend upon initial U-235 concentration and are graphically displayed in Figure 5.6-1 of Amendment 91 to License DPR-67, pp. 5-6b. The racks, as installed, are designed to provide storage up to the year 2008, assuming full core offload capability is maintained. SER-Amendment 91, p. 2.

The basic source of heat energy in the spent fuel pool is the decay heat emanating from the spent fuel. "Decay heat" is the term used to describe the heat generated by the continuing radioactive decay of fission products within spent fuel assemblies stored in the spent fuel pool after the fuel assembly contents have burned up to a certain extent in the nuclear reactor. The decay heat generated from such assemblies in the spent fuel pool diminishes very rapidly, but it is significant for an appreciable length of time. Decay heat is transferred to the pool water and hence to materials in contact with the water. Secondary heat sources are the gamma rays and neutrons emitted by the stored spent fuel rods.

The spent fuel pool cooling system is a closed loop consisting of two centrifugal pumps and a tube and shell heat exchanger with a maximum capacity of 34 Million British Thermal Units per hour (MBTU/Hr.). The normal maximum heat load condition was calculated to be 33.70 MBTU/Hr. SER-Amendment 91, pp. 7, 8.

III. DECISION

A. The Safety of Boraflex

We adopt Licensee's and Intervenor's agreed statement of the Boraflex issues[3], as follows:

> Contention 3. The possible materials degradation and failure that might occur in Boraflex panels due to heat and radioactivity generated in the spent fuel pool have not been adequately considered or analyzed.
>
> Contention 6. The proposed use of Boraflex in the high density spent fuel storage racks designed and fabricated by the Joseph Oat Corporation is essentially a new and unproven technology.

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[3] Contention 3, which originally pertained to all rack and spent fuel cell materials as well as the concrete and steel of the fuel pool structure was the subject of a summary disposition motion which was granted as to all materials except Boraflex. See Memorandum and Order dated October 14, 1988, 28 NRC 455 (LBP-88-27, 1988). The motion was denied as to Boraflex because Licensee had not adequately demonstrated that there were no outstanding safety problems regarding the performance of Boraflex. Id. at 467. Even though the motion was denied, the Board accepted some proposed findings submitted with Licensee's motion for summary disposition pertaining to the application of Boraflex at St. Lucie. The accepted Boraflex-related findings from the August 5, 1988 filing (Licensee's Statement of Material Facts as to Which There Is No Genuine Issue To Be Heard with Respect to Intervenor's Contentions) are Contention 3: Findings Nos. 1, 7, 9. 10, 12, 15-20, and 62-67. As to Contention 6, Findings Nos. 1, 7, 8, 12, 16, 20, 22, 27, and 29 were accepted by the Board. Id. at 467, 473. These previously accepted findings are considered together with the evidence received during the January 24-26, 1989 hearings in Stuart, Florida.

Licensee's Proposed Findings of Fact and Conclusions of Law, ¶ 6 at p. 4; Intervenor's Proposed Findings of Fact and Conclusions of Law, ¶ 6 at pp.2-3.

Licensee and NRC Staff argue that the effects of heat and radiation on Boraflex are known and predictable and that there are no outstanding safety problems related to the use of Boraflex in spent fuel pools. Licensee presented three witnesses on this issue. Dr. Krishna P. Singh testified on behalf of licensee. Dr. Singh is President of Holtec International, a consulting firm which handled the design, analyses and licensing of the St. Lucie 1 Spent fuel racks as a sub-contractor to the rack manufacturer, the Joseph Oat Corporation. He described the specific structural and mechanical design and fabrication of the St. Lucie 1 spent fuel racks so as to accommodate shrinkage of the Boraflex material in such a manner as to prevent loss of its effectiveness following irradiation in the spent fuel pool. Dr. Singh also testified on the results of the Boraflex acceptance testing program and subsequent testing programs. Testimony of Dr. Krishna P. Singh on Contentions 3 & 6 (Singh on 3 & 6), following Tr. 139. Dr. Stanley E. Turner, Chief Scientist for Holtec International testified as to the design of the spent fuel racks authorized by the spent fuel pool expansion amendment (Amendment No. 91 to DPR-67), issued March 11, 1988 (see fn. 1, supra); NRC criteria and guidance; and industry standards for spent fuel pool criticality analysis and their application to the analyses performed for St. Lucie 1. Dr. Turner also addressed the calculational methods used in the criticality analysis and results obtained for the St. Lucie 1 spent fuel pool and the effectiveness of the Boraflex testing program with respect to its ability to identify Boraflex property changes which might affect the performance of the material as a neutron absorber. Testimony of Dr. Stanley E. Turner on Contentions 3 and 6 (Turner on 3 & 6), following Tr. 139. Edward J. Weinkam, III, a Principal Engineer with the Florida Power & Light Co., testified as to the surveillance activities prescribed by the FPL program for testing and in-service surveillance of the Boraflex neutron absorbing material contained in the St. Lucie 1 spent fuel storage racks. Testimony of Edward J. Weinkam, III, on Contentions 3 and 6 (Weinkam on 3 & 6), following Tr. 139.

The NRC Staff also provided three witnesses on this contention, NRC employees Drs. James Wing and Laurence I. Kopp and Mr. Edmond G. Tourigny. Dr. Wing addressed the effects of radiation and heat on Boraflex. Dr. Kopp addressed reactivity considerations attributable to potential or unforeseen Boraflex degradation. Mr.

Tourigny's testimony described and evaluated Licensee's in-service surveillance program which was set up to detect unforeseen Boraflex degradation. Testimony of James Wing, Edmond G. Tourigny and Laurence I. Kopp on Contentions 3, 6 and 7 at p. 1, 6 and 8 resp. following Tr. 110 (Wing, Tourigny and Kopp on 3, 6 & 7).

All of the witnesses had appropriate credentials to support their expert testimony. Intervenor Campbell Rich presented no witnesses.

As described in our October, 1988 Memorandum and Order, gaps in the neutron-absorbing sheets of Boraflex were found at the Quad Cities Plant, a commercial reactor with high-density storage racks similar in design to the St. Lucie 1 racks. 28 NRC 455, 466-467 (LBP-88-27, 1988). The Quad Cities and St. Lucie 1 racks were manufactured by the Joseph Oat Corporation. Licensee argues that the problems identified at Quad Cities have been resolved and will not occur at St. Lucie. Florida Power & Light Co., 28 NRC 455, 466 et seq. (LBP-88-27, 1988).

1. THE USE OF BORAFLEX IN HIGH-DENSITY FUEL STORAGE RACKS

Neutron attenuation in the St. Lucie 1 racks is accomplished through the combined action of borated water and a widely used neutron absorber material, Boraflex. Commonly referred to as a neutron "poison", Boraflex is an effective entrapper of neutrons. It is produced by uniformly dispersing Boron carbide particles in a polymeric silicone encapsulant, which performs as the matrix element. Singh on 3 & 6, at pp. 7, 8. The neutron absorbing element is Boron. Since 1980, 85 percent of all high-density racks ordered by U.S. utilities have used Boraflex as the preferred "poison" material for neutron absorption. This involved 23 separate U.S. commercial nuclear power plants. Id., pp. 7, 14 (Table B). The Joseph Oat Corporation was involved in the fabrication of almost half (11 plants) of the spent fuel storage racks using Boraflex. Id., p. 18, Table A.

2. PROBLEMS WITH BORAFLEX -- SHRINKAGE AND THE FORMATION OF GAPS IN BORAFLEX PANELS

Gaps or separations were found in the Boraflex absorber materials used in the high density spent fuel storage racks at the Quad Cities Plant, 28 NRC 455, at 466, 477 (1988). NRC Information Notice No. 87-43, "Gaps in Neutron Absorbing

Material in High-Density Spent Fuel Storage Racks" and "Board Notification regarding Anomalies in Boraflex Absorbing Material (BN-87-11)" alerted Licensees to potential problems with the use of Boraflex in the spent fuel pools at the Quad Cities and Point Beach facilities. Gaps in the Boraflex plates were found at Quad Cities, and anomalies involving the discoloration and water permeation of Boraflex samples were found at Point Beach. Singh, p. 10. The Point Beach anomalies were found to be of no safety significance. Id. The gaps found at Quad Cities (some up to 4 inches) were determined to be of potential safety significance. Id. More recently, gaps up to 1.4 inches were found in Boraflex panels at the Grand Gulf Station, Unit 1. Wing on 3, p. 3. Both Quad Cities and Grand Gulf are Boiling Water Reactors (BWRs) with high density spent fuel storage racks using Boraflex and fabricated by the Joseph Oat Corporation. Id., Tables A and B, pp. 17, 19.

3. RESULTS AND CONCLUSIONS OF BORAFLEX STUDY PROGRAMS

A considerable amount of information pertaining to Boraflex performance has been accumulated over the last decade. As part of a larger program to qualify Boraflex for use in spent fuel pools, a series of irradiation tests were conducted on small samples at the Ford Reactor at the

University of Michigan at Ann Arbor. Singh on 3 & 6, pp. 13-17. These earlier tests focused primarily on the neutron attenuation characteristics of Boraflex using small coupon samples. The size of the samples used did not permit ready identification of shrinkage characteristics. Singh on 3 & 6, p. 197. Following the discovery of gaps in Boraflex panels used in the Quad Cities spent fuel racks, additional testing was initiated to quantitatively determine radiation-induced shrinkage in Boraflex. Ex. No. 9; Turner on 3 & 6, p. 10, 13, 14. Also as a result of the identified Boraflex problems, the Electric Power Research Institute (EPRI) collected and analyzed data from utility surveillance programs, test reactor irradiations, and the open literature to assess the effect of service environment in spent fuel storage racks on Boraflex. Ex. No. 1. The evidence presented as to the effects of heat and radiation on long-term Boraflex performance is summarized below.

a. The Effect of Heat

Prior to accepting Boraflex as the neutron absorber material, the NRC required testing of this material under physical conditions which were more severe than the environment to which the material would be exposed in actual use. Heat aging tests at 350° F and long term (over 6000 hours) pressure bomb tests at 240° F in boric acid solution (3000 ppm) demonstrated Boraflex's stability under aggravated environmental conditions. Singh on 3 & 6, at p. 14; Exhibit No. 4, at pp. 7, 8. Measurement of the physical characteristics of the test specimens of Boraflex after 251 days indicated a dimensional change, i.e., shrinkage, of less than 1 percent (0.83%) and an average decrease in weight of the test sample of 0.03 percent. The rate of gas evolution was also measured and found to be less than 1.8 x 10^{-3} cubic inches per day per pound of Boraflex. Staff agreed that gas generation was not a problem. Wing on 3, pp. 2, 3, 6. See also Ex. 1 at pp. 4-5, 4-6. The spent fuel pool water at St. Lucie 1 hovers around 100° F, considerably below the test temperatures. Moreover, Boraflex is never exposed to temperatures in excess of 200° F anywhere in the St. Lucie spent fuel pool. Singh on 3 & 6, at 14.

Intervenor argues that the combined effect of heat and radioactivity were not considered in the study programs and therefore the data is meaningless. While it is true that the combined effect of temperature and radioactivity is not reported on as such, the results of in-reactor Boraflex irradiation studies would include the effects of reactor temperature along with radiation effects. Wing at Tr. 548,

549. Since the reactor temperatures are much higher in the reactor than in the spent fuel pool, synergistic effects of heat and radiation would be included in the reported in-reactor irradiation studies. Based on these studies and a review of the 240° F test data, the NRC Staff anticipates no significant heat-induced deterioration of the Boraflex material or its neutron-attenuation ability. Wing on 3, at 5, 6.

b. The Effect of Radiation

Upon irradiation, Boraflex undergoes shrinkage, becoming a hard, ceramic-like material, with increased compressive strength and reduced ductility. Turner on 3 & 6, ff. Tr. 139, at p. 10. Gamma radiation induces cross-linkage of the polymer in Boraflex which leads to shrinkage. As the accumulated radiation dosage increases, cross-linking becomes saturated and no further shrinkage will occur. The NRC Staff estimates that saturation of cross-linking in Boraflex occurs at the cumulative dose of 10^{10} rads, the dose at which Boraflex attains maximum shrinkage. Wing on 3 & 6, ff. Tr. 110, at p. 3. Radiation exposure tests of Boraflex at total equivalent doses of over 10^{12} rads (including 10^{11} rads gamma dose) were performed at the Ford nuclear reactor at the University of Michigan. The Michigan tests support the saturation of cross-linking theory in that the results showed no significant changes of Boraflex shrinkage at cumulative radiation doses from $5x10^9$ to 10^{10} rads. Id. The EPRI Study (Ex. 1) also concluded that shrinkage stops when cross-linking saturates at a gamma exposure of about 10^{10} rads with projected maximum shrinkage at 3-4%. Ex. 1, pp. 5-12, 6-2.

The EPRI Study concluded that an essential factor in Boraflex gap formation and growth appears to be the existence of a mechanism for restraint of the Boraflex sheet. Ex. 1, pp. 5-14 through 5-18. In Point Beach, the sheets were held in place between a pair of V-shaped grooves in the stainless steel sheathing. When removed for examination, the Boraflex sheets were intact. It was concluded that the frictional restraint provided by the V-grooves was not sufficient to result in local stresses to cause the material to tear as the radiation-induced shrinking of the Boraflex proceeded. Id., Figure 2-6, p. 2-9.

In those racks where gaps were observed, there was evidence of restraint through the use of adhesives or by mechanical means sufficient to cause the formation of tears or gaps. At Quad Cities, the Boraflex panels were held in place during manufacture with an adhesive, Dow Silicone Sealant No. 999. Additionally, the Quad Cities racks employ the so-called "cruciform" construction, wherein angles are welded together along the edges in a fixture to form a cruciform with the Boraflex panel contained between the faces of the angle. Cruciforms are attached to each other by welding along their junction. This welding must be done remotely and, as a result, the weld quality depends on the flatness and straightness of the cruciform surfaces. Singh at 10.

Licensee's witnesses concluded that it was the fabrication process that led to excessive restraint of the Boraflex panels, and their subsequent cracking and gap formation following shrinkage upon irradiation at Quad Cities. The "cruciform" construction method is used for rack modules for BWR plants. Id., p. 11. NRC Staff stated that, although it did not have sufficient information to determine conclusively what caused the gap formation, it postulated that because the Boraflex panels were physically restrained, gamma radiation induced shrinkage caused the breakup of the panels and led to separation. Wing on 3, at p. 4. No gaps were observed in Boraflex panels used in pressurized water reactors (PWRs). Turner at Tr. 367. Both Staff and Licensee witnesses concluded that gaps observed in

Boraflex panels were the result of the material being physically restrained while being irradiated (Wing at p. 10; Ex. 1 at 5-16) and further testified that if the Boraflex panels are free to shrink (absence of physical restraint) no gaps will be formed. Singh at Tr. 296; Wing on 3, p. 4; Kopp at Tr. 495; Wing at Tr. 544, 545.

4. ST. LUCIE 1 RACK DESIGN AND FABRICATION PROCESS WITH RESPECT TO AVOIDING EXCESSIVE MECHANICAL CONSTRAINT

The racks fabricated for St. Lucie 1 are not of the "cruciform" design which is unique to BWRs. St. Lucie is a PWR, and the apparently excessive restraint of Boraflex inherent in the BWR rack construction has never been found in the PWR rack design used by the Joseph Oat Corporation. No glue was used in the fabrication of the St. Lucie 1 racks. The racks as fabricated for Region 2 of the St. Lucie 1 spent fuel pool permit unconstrained shrinkage movement of the Boraflex panels within the stainless steel jacket. The panels are more than 6 inches longer than the active fuel length and, if not restrained, can accommodate panel shrinkage of at least 4 percent. The exterior cells in Region 1 are also more than 6 inches longer than the fuel length and are able to accommodate shrinkage movement without external stress. The interior cells in Region 1 are (as a result of construction requirements) of a design which upon shrinkage of the panel would tend to promote the generation of multiple cracks or gaps. The interior cell construction necessitated spot welds at 6-inch distances along the edge of the stainless steel wrapper (12 inches along each side staggered). On shrinking, the Boraflex panels may encounter these spot welds, and local stresses might appear along the axial length of the panels. Singh, ff. Tr. 139 at 11.

5. THE POTENTIAL EFFECTS OF GAP FORMATION ON REACTIVITY

Licensee has evaluated the consequences of various scenarios involving the formation of gaps in the Boraflex panels and loss of borated water in the spent fuel pool. Turner on 3 & 6, p. 7, 17, and Table 1, p. 19. Assuming 4 percent Boraflex shrinkage distributed in 0.5 inch gaps at 12-inch intervals, with gaps at the same elevation in all panels, the calculations show a maximum k-eff of 0.771 under normal operating conditions in Region 1 of the spent fuel pool. Adding to this, a loss of all borated water in the pool results in a k-eff of 0.948, a value still within the acceptable bounds for reactivity. Id. Calculations for Region 2, where Licensee states that gaps are precluded because the panels are fully free to contract, show a k-eff of 0.760 for normal operating conditions and a value of

0.944 for loss of all soluble borate in the fuel pool. Id.

The Staff sees no criticality concerns because the Staff's criteria for k-eff (not greater than 0.95) would not be exceeded. Kopp at Tr. 535. Dr. Turner also calculated the reactivity coefficient for a condition of 4 percent shrinkage of the entire 144-inch panel (5.72-inch shrinkage) occurring at the most reactive position in the same axial plane in all the panel in Region 1 (5.72-inch gaps in all panel at the same elevation) and with no Boron in the spent fuel pool water. Under these extremely unlikely conditions, he calculated a k-eff of 0.992, a value below criticality. Turner at Tr. 412. The k-eff for the same 5.72-inch gap condition with water borated at 1720 ppm would be considerably less. Id., at Tr. 413.

6. THE IN-SERVICE SURVEILLANCE PROGRAM AT ST. LUCIE 1

Long-term and synergistic effects of factors such as radiation, heat, and atmosphere are, at best, very difficult to determine in the short term. It is therefore necessary to employ accelerated testing as a necessary technology to obtain data which can be used with some confidence in an operational situation. To this end, an in-service surveillance testing program will be conducted at St.

Lucie 1. The program is designed to verify the physical characteristics and neutron absorbing properties of the Boraflex utilized in both Regions 1 and 2 of the St. Lucie 1 fuel storage racks.

The Boraflex used in the surveillance program is representative of the absorber material within the storage racks. It is of the same composition, produced by the same method, and certified to the same criteria as the production lot material. The sample coupons are the same thickness as the poison employed within the storage system, and approximately 5" in width, and 15" in length. Each Boraflex specimen is encased in a stainless steel jacket of an austenitic stainless steel alloy identical to that utilized in the storage racks, formed so as to encase the Boraflex material and fix it in a position and with tolerances similar to the design utilized in the racks. The jacket permits wetting and venting of the specimens in a manner similar to that which occurs in the actual rack environment. Weinkam Testimony, ff. Tr. 139 at 4.

In the current program, two types of tests for each Region are planned: a long term test, with coupons surrounded by the same spent fuel assemblies during the entire irradiation period; and an accelerated test, with coupons surrounded by freshly discharged spent fuel assemblies during each refueling. The long-term test coupon examination frequency is after nominal irradiation times of 90 days, 180 days, 1 year, 5 years, 10 years, 15 years, 25 years and 35 years. The accelerated test coupon examination frequency is after each discharge from the second discharge to ninth discharge after the rack installation. Id. at 5.

The coupons will be carefully examined for the following properties:

1. Visual examination intended to reveal any surface or excessive edge deterioration that might appear and to provide supporting information to assist in interpreting any degradation suggested by other measurements.

2. Dimensional measurements to provide a continuing measure of Boraflex shrinkage. The length measurement is of particular importance as an indicator of the potential for gap formation in excess of that accommodated in the design.

3. Neutron attenuation measurements will be made for establishing areal density to confirm that boron is not being lost from the Boraflex. Although previous irradiation tests indicate that boron is retained, this is perhaps the single most important measure of the ability of Boraflex to continue to serve its intended function.

4. Neutron radiography provides supporting information on neutron attenuation and is intended to reveal any non-uniformities in the boron distribution within the Boraflex that might not be uncovered in the attenuation measurements.

5. Shore A hardness measurements will be performed on a continuing basis. Although the Boraflex is expected to become fully hard in the first few

cycles of irradiation, continued measurement is intended to uncover any softening or friability as an indicator of excessive degradation.

6. Weight and specific gravity measurements are supporting measurements intended to reveal any significant loss of Boraflex material or the development of more open porosity than expected.

Turner Testimony, ff. Tr. 139, at 16, 17.

Although Boraflex is expected to satisfactorily perform its intended function, the surveillance program assures that any radiation effects beyond those expected and accommodated in the design, will be detected well in advance (probably years) of the need for remedial action. This surveillance program is consistent with the program described by EPRI in its study with respect to all parameters relevant to the performance of Boraflex as a neutron absorber. Id., at 17.

7. OAT CORPORATION RACKS AS NEW AND UNPROVEN TECHNOLOGY

Intervenor contends that because of the changes made in the fabrication process as a result of problems identified during in-service use of Boraflex in high-density racks, the technology employing the Boraflex is new and unproven. Licensee and NRC Staff disagree. Both contend that high-density spent fuel racks with Boraflex panels as the neutron absorber have been in use since the early 1980's and are not unproven technical innovations or unproven technology. Tourigny on 6 at p. 10; Singh on 3 & 6, at 4-7 and 17.

The Joseph Oat Corporation ("the Corporation"), the St. Lucie 1 rack manufacturer, has had extensive experience with the fabrication of spent fuel pool racks. Prior to the early 1980's when the Corporation began using Boraflex in high-density fuel storage racks, the Corporation was involved in the fabrication of "new fuel racks" which employ the same technological base as spent fuel racks. Additionally, the Corporation has decades of experience in the fit-up, cleaning and handling of stainless steel components, and in the welding processes used in fabricating from stainless steel in sheet metal form, such as in fuel storage rack applications. Singh on 3 & 6, pp. 4-6. Rigorous quality control procedures have been employed at Oat for decades. Their Quality Assurance Program has been reviewed by the survey team of the American Society of Mechanical Engineers (ASME) at 3-year intervals since 1969. The Corporation has passed all of its ASME surveys. Hundreds of pieces of Corporation equipment have been used in nuclear and non-nuclear plants for years. There is undisputed testimony in this record that not a single case of equipment failure leading to plant shutdown has been ascribed to Corporation-supplied equipment. Id.

The Board agrees with Licensee and Staff that utilization of high density racks designed and fabricated by the Joseph Oat Corporation is not utilization of a new and unproven technology.

B. Erroneous Fuel Assembly Storage and Criticality

We adopt Licensee's and Intervenor's agreed statement of the Contention 7 issues[4] as follows:

Contention 7.

1. The mechanisms which prevent the erroneous insertion of a fuel assembly into a storage cell such that the prescription of Standard Review Plan ("SRP") Section 9.1.2, Part III.2.b., that it not be possible for "a fuel assembly ... (to) be inserted anywhere other than a design location", have not been demonstrated; and

2. It has not been shown why criticality will not occur in the spent fuel pool in the absence of a moderator.

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[4] In our October 14, 1988 Memorandum and Order Ruling on Motions for Summary Disposition, we granted summary disposition of Contention 7 with the exception of the two issues discussed in this decision as to which there remained a dispute of fact. Florida Power & Light Co., 28 NRC 455, 473-475 (LBP-88-27, 1988).

Standard Review Plan, Sec. 9.1.2, Part II, 2.b, requires that "The design of the storage racks is such that a fuel assembly cannot be inserted anywhere other than in a design location." The St. Lucie pool racks are divided into two regions, Region 1, in which any of the St. Lucie fuel assemblies can be stored, including fresh fuel, and Region 2, in which only fuel that has reached the burnup requirements set forth in the "Initial Enrichment vs. Burnup Requirements for Storage of Fuel Assemblies in Region 2" curve in Technical Specification 5.6.1.b, Fig. 5.6-1. Tourigny Testimony ff. Tr. 110, at 13.

The racks themselves are designed such that it is physically impossible to insert a fuel assembly in any place other than the storage cells. It is, however, possible to insert an assembly with less than the requisite burnup into Region 2. It is also physically possible to lower a fuel assembly into the shipping cask area and a small area between the east wall of the pool and rack modules E_1 and H_1. There are no racks in those areas. Weinkam Testimony ff. Tr. 21, at 3-4; Tourigny Testimony ff. Tr. 110, at 12-13.

The Standard Review Plan (SRP) requires the Licensee to develop and employ a system which prevents improper placement of a fuel assembly through the use of administrative controls, physical restraints, or by a combination of both. SRP 9.1.2, "Spent Fuel Storage," NUREG-0800. Tourigny Testimony ff. Tr. 110, at 12.

NRC Staff guidance, however, allows for administrative controls, utilizing written procedures, to prevent the misplacement of fuel in the pool. (See Turner on Contention 7, ff. Tr. 21, at 17-18; Tourigny on Contention 7, ff. Tr. 110, at 13.) The Licensee's fuel-handling methods are by administrative control. Licensee described its methodology as follows:

> Each fuel assembly arrives at St. Lucie 1 with a unique serial number which is engraved on it. The serial number remains visible regardless of storage location within the pool to facilitate identification. The Licensee tracks the location of a fuel assembly throughout its life by its serial number.
>
> Fuel is moved to, and inserted into, a spent fuel rack cell location with a spent fuel pool machine which consists of a rolling bridge which spans the pool, and a fuel lifting device. The fuel lifting device may be positioned by a spent fuel machine operator over any rack cell location in Regions 1 or 2. Each cell location within the racks is identified by a region-unique index system, which uses a grid for Region 1 and another for Region 2. Fuel assemblies are tracked within the pool by maintaining records of their serial numbers on maps indicating the cell locations and associated alpha-numeric index codes where the assemblies are located. Location of new and burned fuel assemblies, stored in the spent fuel racks, are tracked by serial numbers which are reported in fuel status report records and spent fuel pool fuel locations maps. The transfer of assemblies to predetermined locations is conducted by an NRC-licensed

> operator under the direction of the licensed Control Room operator.
>
> Following refueling, an independent verification (by a remotely controlled camera) of the location of the fuel assemblies in the reactor core and the spent fuel pool is conducted, and fuel status records are updated to reflect any assembly location changes. In addition, an audit of the spent and new fuel in storage must be completed at least annually in accordance with 10 C.F.R. Part 75.

Weinkam Testimony ff. Tr. 21, at 3-7.

The Board finds that the foregoing procedures and restraints used in the handling of fuel assemblies in the spent fuel pool are adequate to provide reasonable assurance that fuel will be stored in the prescribed areas of the pool. The procedures satisfy the guidelines of SRP 9.1.2 and will ensure against improper storage of fuel assemblies.[5] This issue under Contention 7 is dismissed.

The second issue under Contention 7 to be resolved arises out of Licensee's statement, in several places in its

---

[5] It is also pertinent to note that, even if a fresh fuel assembly were to be mislocated within the storage pool in the worst possible location, the maximum k-eff would remain below 0.8, taking into account the presence of soluble boron in the pool water. Turner on Contention 7, ff. Tr. 21, at 18-19; Turner, Tr. 92-93. Even in the absence of soluble boron, the misinsertion of a fresh fuel assembly into a Region 2 location would not result in criticality. Turner, Tr. 92-93. Multiple misinsertions would be necessary. (Id.) With the prescribed soluble boron in the pool, criticality would not occur even if fresh fuel were misinserted into each and every Region 2 cell. Turner, Tr. 55-57.

motion for summary disposition, that in the absence of a moderator it would not be possible for the St. Lucie fuel assemblies to form a critical mass in any configuration. The Intervenor questioned this statement, and pointed out that a Dr. Slotin was fatally injured in an incident which resulted in a "dry" criticality at Los Alamos in 1947, and that atomic weapons achieve criticality with no moderator present.

Licensee and Staff dispute Intervenor's assertion and deny the relevance of the two examples he cites. Licensee's witness, Dr. Turner, testified to Licensee's underlying criticality theory as follows:

> The term "fissile material" refers to material the atoms of which are capable of being split or fissioned with the attendant production of large quantities of heat energy (the useful product from the reactor) upon the capture (absorption) of neutrons. The primary fissile material in new fuel assemblies of most nuclear power reactors, including St. Lucie 1, is a nuclide of uranium called uranium-235. In natural uranium, the uranium-235 is present at a concentration less than 1 percent by weight, with almost all of the remainder being the uranium-238 nuclide. To be useful in a light-water nuclear power reactor, natural uranium is enriched in uranium-235. The nuclear fuel utilized at St. Lucie 1 may be enriched up to 4.5 percent by weight of uranium-235, with almost all of the remaining 95.5 percent being the uranium-238 nuclide.
>
> In general, when a neutron is absorbed by uranium-235, there is a high probability that uranium-235 will undergo fission, resulting in the release of energy, fission products and more neutrons. These neutrons, in turn, can (1) be absorbed by uranium-235 or other

fissile nuclides, (2) be absorbed by uranium-238 nuclides, resulting in virtually no additional fission, (3) be absorbed non-productively by non-fissile materials called "poisons" (resulting in no additional fission), or (4) escape without being absorbed (i.e., leakage, which also results in no additional fission).

As a practical matter, not all neutrons released as a result of fission will cause additional fissions. Uranium-238 nuclides, poison materials and leakage inhibit the fission process by reducing the number of neutrons available to cause fissions. If fewer neutrons are being produced as a result of fission than are leaking and being absorbed, the fission process will not sustain itself; this condition is called "subcriticality." In contrast, if the rate of neutron production as a result of the fission process is equal to the rate of neutron absorption and leakage, the fission process will sustain itself, and the condition is referred to as "critical."

The term "effective multiplication factor" is defined as the ratio of the number of neutrons per unit of time produced in the fission process, to the number of neutrons per unit of time absorbed and escaping. The effective multiplication factor, commonly called k-effective (or k_{eff}), is a measure of the ability of a system to sustain a fission reaction. Criticality occurs whenever the effective multiplication factor reaches or exceeds a value of 1.0 because at least as many neutrons are being produced as are being lost by absorption and leakage. For a k_{eff} less than 1.0, the fission rate cannot be sustained. The margin below a k_{eff} of 1.0 is the safety margin to criticality, and this subcritical margin is the difference between a k_{eff} of 1.0 and the k_{eff} of a given system.

Turner Testimony ff. Tr. 21, at 5-7.

U-235, the reactive isotope of uranium used in the reactor system, is a poor absorber of the "fast" neutrons produced in the fission process, but is a very good absorber of "slow" or "thermal" neutrons. U-238, which comprises the bulk of the uranium in the fuel, is, conversely, a very good absorber of fast neutrons, but a poor absorber of thermal neutrons. Unless some mechanism is brought into play which will slow down the fast neutrons to allow neutron absorption by U-235, the fraction of neutrons absorbed by U-235 is small compared with the absorption by U-238.[6] Turner Testimony ff. Tr. 21, at 19-20. This requires the presence of a moderator.

A moderator is a material consisting of light elements which scatter and slow down the neutrons, but which do not absorb many of the neutrons in the process. Turner, Tr. at 60. There are only a few good moderators. The only ones that are in common use are water (hydrogen), heavy water (deuterium), graphite, and beryllium. The moderator used in the St. Lucie reactor and fuel pool is light water. Turner, Tr. 60-62.

Intervenor attempted to establish that if the fuel melted and slumped to the floor of the pool that there would be sufficient zirconium, air, wood and concrete in the mass to act as moderators. Both Licensee's and Staff's witnesses denied this, saying that while there might be some small moderation by these materials, in practice it would be

[6]In order to simplify this discussion, the possibility of escape or non-fission capture of neutrons, neither of which produce new neutrons, is ignored.

negligible and insignificant. Turner, Tr. at 62; Kopp, Tr. at 116-119. Intervenor also questioned the amount of plutonium in spent fuel. Dr. Turner replied that the total amount of fissionable material in spent fuel, including both U-235 and the fissionable plutonium isotopes was about the equivalent of fresh fuel enriched to about 1.7%. Turner, Tr. at 67. This is far less reactive than fresh fuel. Intervenor then asked about the total weight of uranium oxide, plutonium, fission products and zirconium in the pool. Licensee's witness had no figures, but stated that the total amounts were irrelevant, as in his calculations he assumed an infinite mass as a matter of conservatism. The conservatism in assuming infinite mass is that neutron leakage, i.e., a net neutron loss, is ignored. Turner, Tr. at 66.

As a basis for his thesis that a moderator was not necessary for criticality, Intervenor asserted that several incidents had occurred where criticality was achieved without a moderator. Intervenor's Response to Licensee's Motion for Summary Disposition of Intervenor's Contention 7, ¶ 9. One, a criticality accident at Los Alamos in 1947, involved experiments with a supercritical mass of highly enriched plutonium metal in a form capable of attaining "dry" criticality. Turner Testimony ff. Tr. 21, at 21, 22.

That material has no relationship to the low-enriched St. Lucie 1 uranium fuel. Similarly, the fact that nuclear weapons do not use a moderator is irrelevant. Weapons use either highly enriched U-235 or plutonium metal, which is not the case at St. Lucie. Three Mile Island and Chernobyl, both mentioned by the Intervenor, were moderated, the former with water and the latter with graphite, and do not apply to Intervenor's assertion that criticality could occur in the St. Lucie spent fuel pool if no moderator were present.

The Board has reviewed the entire record on the criticality issue and has found no basis to question Licensee's position. The Staff agrees with Licensee that in a dry fuel pool there is no danger of accumulating a critical mass of fissile material. We therefore find that Licensee has met its burden of proof in this matter and find in favor of Licensee on Contention 7.

The Board finds that Licensee has met its burden on each of the admitted contentions and operation of the spent fuel pool as modified is and would be in compliance with the Rules and Regulations of the Commission.

IV. CONDITION

However, there is one aspect of the Application which was the subject of much discussion at the evidentiary hearing and by the Licensing Board following the hearing. That is the matter of the "controlled gap formation" in the interior Boraflex panels in Region 1. Licensee's experts argue that the construction technique used in Region 1, while required because of the manner in which the cells are held together, is such that if the Boraflex panels are subjected to gamma radiation sufficient to cause shrinkage and sufficient stress at the weld connection points, they would selectively break at the weld point locations. Weld connections are located at 12- inch spacing staggered along each side of the Boraflex cover panel (6-inch vertical spacing staggered along the panel length). Licensee's witnesses contend that the panel, if stressed sufficiently to cause rupture, would break at the weld connection on 6 or 12-inch intervals. Assuming 4 percent shrinkage and stress relief at 12-inch spacing, they calculated a gap size of 0.5 inches. Singh ff. Tr. 139 at 11; Turner ff. Tr. 139, at 19. The NRC Staff did not address this aspect of Licensee's design. Written and oral testimony by Staff witnesses stated that no mechanism for gap formation existed and therefore no gaps should be formed in the Boraflex panels.

This Staff assertion was reiterated on the stand even after the rack designers described the system for controlling the location of gaps in Region 1. Wing ff. Tr. 110, at 4; Tr. 544-45.

The controlled gap system is unique and has not been tested. As far as the Board is aware, there is only one practical way to determine the effectiveness of the Licensee's method for controlling gaps and that is to measure the ability of the Boraflex panel to absorb neutrons by a technique known as "blackness testing". Both Licensee and Staff argue that the predicted 0.5 inch gaps would not be detectible by blackness testing and therefore it is not necessary. Tourigny at Tr. 552; Turner at Tr. 321-22. Licensee further argues that the Region 1 pool is generally not subject to irradiation. Spent fuel is normally discharged to Region 2, while Region 1 is used to store fresh fuel prior to refueling and for contingencies such as the possible need for a full core offload. Weinkam at Tr. 140; Turner at Tr. 3[illegible]. Because of the normal use of Region 1, shrinking and subsequent gap formation should thus be nonexistent or minimal in the Region 1 racks. In the Region 2 racks Boraflex is unconstrained and no gapping should occur. Singh ff. Tr. 139, at 11. The one exception which does result in some gamma irradiation of Region 1 cells occurs because of the in-service surveillance program which Licensee has undertaken. This program includes two cells in Region 1 with separate sets of sample coupons. Turner ff. TR. 139 at 15-16; Weinkam ff. Tr. 139 at 5.

The Board agrees that, without gamma irradiation, the Boraflex in Region 1 should not form gaps. The Board also agrees that even with irradiation the unconstrained exterior Boraflex panels in Region 1 and all the panels in Region 2 should not form gaps. Gamma irradiation of the interior panels in Region 1, however, poses a different situation.

We, therefore, impose the following condition on the license amendment: In the event that any of the Region 1 Boraflex test coupons are found to be subjected to gamma irradiation equal to or greater than $1x10^8$ rads, Licensee is directed within 30 days to prepare a study program to be approved by NRC Staff and performed by the Licensee to assess the effect of the irradiation on the integrity of the Boraflex panels. The study program should include blackness testing or a state-of-the-art equivalent approved by the NRC Staff.

ORDER

For all the foregoing reasons and upon consideration of the entire record in this matter, it is, this 9th day of May, 1989

ORDERED

1. That judgment is granted for Licensee on the matters remaining at issue in Contentions 3, 6, and 7, except as to the condition imposed in paragraph 3 below;

2. That License Amendment No. 91 to License No. DPR-67, issued by the NRC Office of Nuclear Reactor Regulation on March 11, 1988 shall remain in full force and effect as issued;

3. That in the event that any of the Region 1 Boraflex test coupons are subjected to gamma irradiation equal to or greater than $1x10^8$ rads, Licensee is directed to prepare within 30 days a study program to be approved by the NRC Staff and performed by the Licensee to assess the effect of the irradiation on the integrity of the Boraflex

panels. The study program should include blackness testing or a state-of-the-art equivalent approved by the NRC Staff; and

4. That, pursuant to 10 C.F.R. § 2.760 (1988) of the Commission's Rules of Practice, this Initial Decision shall become effective immediately. It will constitute the final decision of the Commission forty-five (45) days from the date of issuance, unless it is appealed in accordance with 10 C.F.R. § 2.762[7] (1988) or the Commission

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[7] Any party may appeal from this decision by filing a notice of appeal within ten (10) days after service of this Initial Decision. Pursuant to 10 C.F.R. § 2.762 (1988), each appellant must file a brief supporting its position on appeal within thirty (30) days after filing its notice of appeal (forty (40) days if the Staff is the appellant). Within thirty (30) days after the period has expired for the filing and service of the briefs of all appellants (forty (40) days in the case of the Staff), a party who is not an appellant may file a brief in support of, or in opposition to, the appeal of the other party. A responding party shall file a single, responsive brief <u>only,</u> regardless of the number of appellant's briefs filed.

directs otherwise. See also, 10 C.F.R. §§ 2.764, 2.785, and 2.786 (1988).

ATOMIC SAFETY AND LICENSING BOARD

B. Paul Cotter Jr.

B. Paul Cotter, Jr., Chairman
ADMINISTRATIVE JUDGE

Glenn O. Bright

Glenn O. Bright
ADMINISTRATIVE JUDGE

Dr. Cole was not availabe to sign this decision but fully concurs in the result.

Dated at Bethesda, Maryland,

this 9th day of May, 1989.

Appendix A

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

B. Paul Cotter, Jr., Chairman
Glenn O. Bright
Dr. Richard F. Cole

In the Matter of:	Docket No. 50-335-OLA
FLORIDA POWER & LIGHT COMPANY	(ASLBP NO. 88-560-01-LA)
(St. Lucie Plant, Unit No. 1)	May 9, 1989

TRANSCRIPT CORRECTIONS

Pursuant to motions by Licensee and the NRC Staff, as well as on its own initiative, the Board makes the following corrections to the transcript of the January 24 to 26, 1989 hearing in the captioned proceeding. Some changes proposed by the parties were not accepted, not so much because the Board disagreed with them in the context but because there was no evidence to conclude other than that the witness simply misspoke.

Page	Line	Change
Tuesday, January 24, 1989		
6	13	Change "polimer (?) to "polymer"
6	22	Change "fuel of ray" to "K-effective"
12	23	Add between "comprehensive" and "analysis" the word "safety"

17	9	Change "MS-1050" to "NS-1050"
23	14	Change "1714" to "0.71"
24	6	Change "neither" to "either"
24	17	Add "in the reactor" after "correct" ??
30	13	Change "fuel" to "reactor"
35	24	Change "plutonium 239" to "plutonium"
50	1	Change "more" to "less" ??
60	14	Change "residence" to "resonance" ??
76	11	Change "fuel" to "reactor"
95	1	Add between "criticality" and "under" the words "could occur"
101	11	Change "deform gaps" to "will form gaps and"
101	11	Change "these gaps" to "these spot welds"
101	14-15	Change "two above does" to "two (above and below) do"
109	6	Change the word "Wing" to "Wade"
120	16	Change "using" to "losing"
124	20	Change "is" to "has" and "process" to "cross section"
128	23	Delete the "U" between the word "plutonium" and the numbers "239"
129	10	Change the word "they" to "we"
129	16	Delete the second "this"
145	10	Change "austenetick" (??) to "austenitic"

146	17	Change "austenetick" to "austenitic"
167	15	Change "95" to ".95"
183	1	Change "add" to "as"
185	2	Change "aerial" to "areal"
196	1	Change "modulers" to "modulus"
196	3	Change "austenetick" to "austenitic"
196	7	Change "modeli" to "moduli"
197	8	Change "double up again" to "develop gapping"
204	8, 11	Change "aerial" to "areal"
208	12	Change "BRW" TO "BWR"
211	24	Delete "three" ??

Wednesday, January 25, 1989

256	21	Add between "says" and "rapid" the word "no"
296	25	Delete "time"
297	11	Change "shrunk" to "gapped"
297	22	Change "does have inflamed forces" to "does not have in-plane forces"
300	4	Change ".2" to "2."
308	14	Change "moving" to "removing"
308	15	Change "in essence in" to "in essence is"
308	16	Change "increases" to "increase in"
308	16	Change "site" to "sight"
309	21	Change "hemispherical" to "semicircular"

315	9	Change "gaps" to "cutouts"
319	19	Change "extends" to "tends"
347	24	Change "probability" to "factor"
348	23	Change "cooperation" to "reactor operation"
351	25	Change "siding" to "surrounding"
355	3	Add "not" between "why" and "in"
367	18	Add "and the shrinkage" between "panel" and "was"
402	6	Change "difference" to "different" and "quantity" to "qualify"
404	21	Delete "to shrinkage"
412	10	Change "COLE" TO "TURNER"
412	11	Change "5.72 gaps" to "5.72 inches"
412	19	Change "incident rate" to "infinite array"
414	18	Change "used" to "made"

Thursday, January 26, 1989

Index	Between lines 5-6	The word "Edmont" should be "Edmond"
430	11	Change "on" to "one"
437	23	Delete the "?" at the end of this line
439	15	Insert the word "the" between the word "to" and the word "eleventh"
439	22	Change "conversion" to "composition"
442	13	Change "for" to "at"
442	22	Change "was" to "were"

453	18	Add the word "to" at the end of this line after the word "have"
455	13	Change "is" to "as"
455	15	Change "and" to "at"
455	18	Delete the "?" at the end of this line after the word "Cities"
Index	Between 3-4	The word "Edmont" should be lines "Edmond"
467	18	Change "cutoff" to "cutout"
478	4	Change "irradiated" to "unirradiated"
483	11	Change "seasoning" to "scissioning"
483	12	Change "is seasoning" to "if scissioning"
489	18	Change "late" to "plate"
499	6	Change "two" to "team"
520	5	Delete the letter "Q" at the beginning of the line and connect the text on this line to the text on line 4
536	9	Change "so" to "show"
544	25	Change "deal with" to "produce"
548	9	Change "of what" to "on that"
551	13	Insert the word "it" between the words "that" and "has" at the beginning of this line
556	15	Change "gaps" to "space"

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of

FLORIDA POWER AND LIGHT COMPANY

(St. Lucie Plant, Unit No. 1)

Docket No.(s) 50-335-OLA

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing INITIAL DECISION DTD 5/10/89 have been served upon the following persons by U.S. mail, first class, except as otherwise noted and in accordance with the requirements of 10 CFR Sec. 2.712.

Administrative Judge
Thomas S. Moore, Chairman
Atomic Safety and Licensing Appeal Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Administrative Judge
Alan S. Rosenthal
Atomic Safety and Licensing Appeal Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Administrative Judge
Howard A. Wilber
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Administrative Judge
B. Paul Cotter, Jr., Chairman
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Docket No.(s)50-335-OLA
INITIAL DECISION DTD 5/10/89

Dated at Rockville, Md. this
10 day of May 1989

Pat Carey
Office of the Secretary of the Commission