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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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USNRC

ATOMIC SAFETY AND LICENSING BOARD '88 OCT 18 A9:53

Before Administrative Judges:

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

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

SERVED OCT 18 1988

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| 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) | October 14, 1988 |

# MEMORANDUM AND ORDER
(Ruling on Motions for Summary Disposition)

## I. INTRODUCTION

Licensee, Florida Power & Light Company, has moved for summary disposition of the six remaining contentions in this proceeding.[1] Those contentions, filed by Campbell Rich, a nearby resident, challenge whether specific aspects of Licensee's plan to rerack the spent fuel pool at its St. Lucie Unit 1 plant will adequately protect the public health and safety. The reracking at issue would

[1]Of the 7 contentions originally admitted to the proceeding, Florida Power & Light Co., 27 NRC 452 (LBP-88-10A, 1988), Contention 2, concerning a temporary storage crane, was dismissed by unpublished order dated July 27, 1988, at the request of the Intervenor when he learned that the crane had been removed.



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> ... authorize the licensee to increase the spent fuel pool storage capacity from 728 to 1706 fuel assemblies. The proposed expansion is to be achieved by reracking the spent fuel pool into two discrete regions. New, high density storage racks will be used.

52 Fed. Reg. 32852.

In the six contentions at issue here, Mr. Rich asserts that: (a) there is a danger of radiation releases from a cask drop accident (Contention 1); (b) analyses of materials deterioration or failure of materials integrity are inadequate (Contention 3); (c) the high density design, through various mechanisms, could cause a major release of radioactivity into the environment (Contention 4); (d) the cooling system will be inadequate in the event of a single failure of the pumps or the electrical system, thus creating greater potential for an accidental radioactive release (Contention 5); (e) the high density storage racks are a new and unproven technology (Contention 6); and (f) the increased number of fuel rods will increase the probability of a criticality accident (Contention 7). Mr. Rich's response to the motion for summary disposition only addressed Contentions 3, 6, and 7. The Nuclear Regulatory Commission Staff supported the Licensee's motion on all six contentions.

For the reasons set forth within, we grant the motion as to Contentions 1, 3 (in part), 4, and 5. We find that there is no issue of material fact as to those contentions and that Licensee is entitled to a decision thereon as a matter of law. Licensee's motion is denied as to parts of Contention 3 and 7 and Contention 6 in its entirety.

## II. THE SPENT FUEL POOL

The spent fuel pool at issue in this proceeding is adjacent to Unit 1 of the St. Lucie Plant 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. Licensee submitted nine affidavits[2] averring facts not contested by Intervenor unless otherwise

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[2]Licensee's affidavits are as follows:

1. Affidavit of Stephen Marschke on Admitted Contention 1 (Mr. Marschke is a Supervising Engineer in the Nuclear Licensing Department of Ebasco Services Inc., New York, NY);

2. Affidavit of Murray Weber on Admitted Contention 1 (Mr. Weber is a supervising civil engineer in the Nuclear Licensing Department of Ebasco Services, Inc., New York, NY);

3. Affidavit of Murray Weber on Admitted Contention 3;

4. Affidavit of Dr. Gerald R. Kilp on Admitted Contention No. 3 (Kilp Affidavit 3a) (Dr. Kilp is an Advisory Engineer in the Engineering Department of the Westinghouse Electric Corp., Pittsburgh, PA);

5. Affidavit of Dr. Gerald R. Kilp on Admitted Contention 3 (Kilp Affidavit 3b);

6. Affidavit of Dr. K. P. Singh on Admitted Contentions 3 and 6 (Dr. Singh is President of Holtec International, Mount Laurel, NJ);

(Footnote Continued)

noted in the text. We find from an examination of each affiant's education and experience that each is qualified to testify as an expert witness.

Licensee's uncontested affidavits establish the following facts about the spent fuel pool. The pool itself 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. Weber-3, ¶ 4.

A separate but adjacent fuel cask storage area is located at the northeast corner of the spent fuel pool. It is 10 feet long, 12 feet wide, and 3 feet 6 inches deep. Weber-3, ¶ 4. The south and west walls of the fuel cask storage areas are over 14 feet high and are made of steel plate lined with 1/4 inch stainless steel. The spent fuel cask weight is limited to 25 tons. Weber-1, ¶¶ 8 and 9.

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(Footnote Continued)

7. Affidavit of Dr. K. P. Singh on Admitted Contentions 4 & 5;

8. Affidavit of John B. Houghtaling on Admitted Contentions 4 & 5 (Mr. Houghtaling is a Project Manager with Ebasco Services Inc., Kenner, LA); and

9. Affidavit of Dr. Stanley E. Turner on Admitted Contentions 6 and 7 (Dr. Turner is Chief Scientist of Holtec International, Mount Laurel, NJ).

The fuel assembly structures containing the spent fuel to be stored in the pool are made of stainless steel and inconel. The cladding on the assemblies is made of 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. Kilp-3b, ¶ 4. 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. Kilp-3a, ¶ 10; Kilp-3b, ¶ 8; Houghtaling-4 & 5, ¶ 14.

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 per cent 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 SER is attached to License Amendment 91 for the St. Lucie Plant.

The essential difference between Region 1 and Region 2 rack modules is that the Region 1 racks are provided with additional neutron absorbing insulation (Boraflex) so as to absorb the higher neutron concentrations that would come from 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 graphi ' displayed in Figure 5.6-1 of Amendment 91 to License DPR-67, p. 5-6b. ine racks, as installed, are designed to provide storage up to the year 2008, assuming full core offload capability. 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. Singh-4 & 5, ¶ 9, 10 and Fig. 1. Decay heat is transferred to the pool water and hence to materials in contact with the water. Secondary heat sources are the gamma and neutron bombardment of pool materials. Kilp-3a, ¶ 12; Kilp-3b, ¶ 5; Weber-3, ¶ 17.

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; Houghtaling-4 & 5, ¶¶ 5 and 6; Singh-4 & 5, ¶ 12 and 13.

## III. CONTROLLING LAW

The requests for summary disposition at issue here are filed pursuant to 10 C.F.R. 2.749(a) which authorizes any party to move for a decision "... in that party's favor as to all or any part of the matters involved in the proceeding." The purpose of the summary disposition procedure is to avoid holding hearings on issues where there is no genuine dispute of material fact. "Statement of Policy on Conduct of Licensing Proceedings, 13 NRC 452, 457 (CLI-81-8, 1981). See Houston Lighting and Power Co., 11 NRC 542, 550 (ALAB-590, 1980).

The rule requires that each party must file a statement of material facts as to which they believe there is no genuine issue to be heard, and "All material facts set forth in the statement ... will be deemed to be admitted unless controverted ... ." Moreover, under subsection (b) of section 2.749

> ... When a motion for summary disposition is made and supported as provided in this section, a party opposing the motion may not rest upon the mere allegations or denials of his answer; his answer ... must set forth specific facts showing there is a genuine issue of fact. If no such answer is filed, the decision sought, if appropriate, shall be rendered.

Nevertheless, the movant, not the opposing party, has the burden of showing the absence of a genuine issue as to any material fact. Cleveland Electric Illuminating Co., 6 NRC 741, 753 (ALAB-443, 1977). Conclusionary statements in the motion, unsupported by factual showings as, for example, in affidavits, will not support a decision on the motion in favor of the movant. Thus, where the evidentiary material in support of a motion does not establish the absence of a genuine issue of material fact, summary disposition must be denied, even if no opposing evidentiary material is presented. Adickes v. Kress & Co., 398 U.S. 144, 159 (1970) cited in Cleveland Electric Illuminating Co., supra. Conversely, our rule, at 10 CFR 2.749(b)(1988) makes absolutely clear that the opponent to a properly supported motion for summary disposition may not rest upon "mere allegations or denials", but must answer setting forth "specific facts showing there is a genuine issue of fact." Virginia Electric and Power Co., 11 NRC 451, 453 (ALAB-584, 1980).

## IV. RULINGS ON MOTION

### A. Contention 1

Contention 1 avers:

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

Florida Power & Light Co., 27 NRC 452, 470 (LBP-88-10A, 1988). As the bases for this contention Intervenor asserts that:

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

Request for Hearing and Petition for Leave to Intervene, p. 4 (Served January 21, 1987) (hereinafter, "Amended Petition").

1. Radioactive Releases

(10 C.F.R. Part 100)

Licensee asserts through uncontroverted affidavits and an uncontroverted statement of material facts that its

> analyses of postulated cask drop accidents for damaged fuel were conservatively performed and showed that any radiation releases would be well within 10 C.F.R. Part 100 guideline values.

Licensee's Statement of Material Facts as to Which There Is No Genuine Issue To Be Heard With Respect to Intervenor's Contentions, Contention 1, ¶ 23 (hereinafter, "Licensee's Facts"). For the reasons set out below, we agree.

Licensee's analysis begins with a recitation of the dose limitations in 10 C.F.R. § 100.11(a) (1988) for: (1) an individual located at a point on the exclusionary area boundary; and (2) an individual located at any point on the outer boundary of the low population zone. Licensee notes that the dose for the former would control for a cask drop accident and avers that criticality would not occur under the postulated accident conditions. Marschke-1, ¶¶ 5, 6.

For its radiological consequence analysis, Licensee draws from the assumptions in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Reactors," the document referenced by § 15.7.5 of the Standard Review Plan (NUREG-0800) ("SRP"). Using the fission product release fractions specified in Regulatory Guide 1.25, Licensee assumed conservatively "that all fuel assemblies in the completely filled fuel pool were ruptured" and that all of the radioactive noble gases and iodine in the gas gap of the fuel rods were released. Id., ¶¶ 11-13, 15, 17. These assumptions were made for two cases, both assuming the pool was filled, in one instance after 1/3 of the fuel assemblies had been removed to the pool from the core and in the other in which all fuel assemblies had been removed to the pool from the core. Because of the delay in transferring spent fuel from the core (whether 1/3 or the entire core) to the pool, mandated by the St. Lucie Unit 1 technical specification, Licensee's analysis concluded that the exclusion area exposures would be

10% or less of the 10 C.F.R. Part 100 guidelines. Id., ¶¶ 10, 13 & Table 1. Thus, Licensee concluded that the exclusionary area boundary doses were determined to be well within 10 C.F.R. Part 100 exposure guidelines and that, consequently, NRC regulatory requirements and guidance have been met. Id., ¶ 19.

We agree. We find that licensee's assumptions, analysis, and conclusions are quite reasonable and are otherwise uncontroverted. Licensee has met its burden of proof in this portion of its motion in demonstrating that its calculations of radiological consequences with respect to releases from a cask drop accident satisfy the regulatory requirements set out in 10 C.F.R. Part 100. We are satisfied that there is no genuine issue of material fact remaining as to this portion of Contention 1 and grant summary disposition in favor of Licensee thereon.

2. Cask Drop Accident Consequences

Contention 1 sets out as the mechanism for the foregoing radioactive releases, a cask drop accident in the spent fuel pool, relying on the uncertainties identified in a report by the Brookhaven National Laboratory entitled, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue No. 82", NUREG/CR-4982, BNL-NUREG-52093 (July 1987) (hereinafter "BNL Report"). The contention appears to assume a total loss of water in the pool caused by structural damage to the pool resulting from the cask drop. Since we have found that any releases would not exceed regulatory requirements, there is no need to examine Licensee's analysis of a cask drop accident at Unit 1, particularly since Intervenor has not controverted the affidavits and facts submitted by Licensee in support of its motion for summary disposition of the question.

Nevertheless, in light of our obligation under 10 C.F.R. § 2.749(a) to make an affirmative finding of the absence of any genuine issue of material fact, it is worth reciting some of the significant differences between the BNL Report's cask handling assumptions and the actual configuration of cask handling at the St. Lucie Unit 1 spent fuel pool described in Licensee's motion. The BNL Report states that

> ... some spent fuel pools have a special section for the shipping cask separated from the main pool by a wall with a weir or gate. For such a configuration the number of passes over the pool edge would be zero and hence the risk to the main pool from a cask drop would be zero.

BNL Report, p. 37, n. "a". As described above, the St. Lucie plant is so configured.

The BNL Report also appears to assume an accident involving a 100 ton cask. As noted, casks at Unit 1 are limited to a weight of 25 tons. Weber-1, ¶ 9. See p. 4, supra.

Based on an assumed maximum distance drop of a 25 ton cask onto the thinnest part of the mat at the bottom of the pool, Licensee concluded, using conservative assumptions as to factors such as energy loss, cask rigidity, temperature, and live and dead loads, that "All safety factors are greater than one and therefore the cask drop accident will not cause spent fuel pool structural failure." Weber-1, ¶ 12. Licensee considered other possible scenarios but found that none of them equalled the impact energy generated by the dropping of a cask from the maximum height possible. Id., ¶ 13.

Licensee acknowledges the possibility of "fine hair line cracks" developing in the cask storage area but concludes that the amount of water leakage would be so low in volume as to be easily handled by the makeup water capability of the pool's makeup system. Id., ¶ 12. We agree.

Other possible scenarios for a cask drop accident have been considered, but the impact energy would be less than the scenario discussed above. We find that any threat of radiological release is bounded by that discussion and that other cask scenarios would not cause radiological consequences exceeding the 10 C.F.R. Part 100 criteria. See Id., ¶ 13.

Accordingly, as to Contention 1, we are satisfied that the showing made by Licensee establishes that there are no genuine issues of material fact and that Licensee is entitled to judgment as a matter of law. Licensee's motion is granted as to Contention 1, and it is dismissed with prejudice.

B. Contention 3

Admitted Contention 3 states as follows:

> The Licensee and Staff have not adequately considered or analyzed materials deterioration 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.

Florida Power & Light Co., 27 NRC 452, 470 (LBP-88-10A, 1988). As bases for the Contention, Intervenor asserts that

> The spent fuel pool 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.

Amended petition, pp. 5-6.

1. Background

At oral argument Petitioner asserted that the normal temperature of the pool would be increased, subjecting the pool materials, particularly the concrete, to greater stress. Petitioner asserted that the calculation of these factors were "clearly inadequate." Florida Power & Light Co., 27 NRC 452, 462 (LBP-88-10A, 1988). The Board limited the scope of the contention to the length of time authorized by the licensing amendment at issue, March 1, 2016. Id. at 463. Weber-3, ¶ 14.

Licensee submitted four affidavits in support of its motion for summary disposition of Contention 3. Each of Licensee's affiants addressed the impacts of radiation and heat on one or more of the materials comprising the fuel pool.[3]

The NRC Staff supports Licensee's motion and submitted the affidavit of Edmond G. Tourigny.[4] Mr. Tourigny generally supported each of Licensee's affiants in their submittals on this contention.

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[3]Weber-3; Kilp-3a; Kilp-3b; and Singh-3 & 6. See fn. 2, supra.

[4]Mr. Tourigny is the Nuclear Regulatory Commission Project Manager for both units at the St. Lucie plant. He has been employed by the Commission in various capacities for 12 years and has Masters of Engineering degrees in both the industrial and the nuclear fields.

Intervenor filed a timely response in opposition to the motion. Intervenor's reply addressed only the impact of the increased generation of heat and radioactivity on the neutron absorbing material Boraflex.

2. Uncontested Issues

a. Levels of Radiation and Heat

Four types of radiation (alpha, beta, gamma, and neutron) will be present in the spent fuel pool to varying degrees. The alpha and beta radiation are of no concern because they cannot penetrate the fuel cladding surface to any significant depth and hence will not have any effect on the cladding outer surface or any materials outside of the cladding surface. Kilp-3b, ¶ 5. The situation is different with gamma and neutron radiations. Kilp-3a, ¶ 8. The effects of each of these types of radiation on materials are a function of cumulative exposure. Gamma radiation dosage is expressed in rads (roentgen absorbed dose), while the cumulative neutron exposure is expressed in neutrons per square centimeter. The estimated maximum accumulated gamma radiation dosage for the reracked pool ranges from $5x10^{10}$ rads for the fuel cladding and assemblies to $3x10^{10}$ rads for the fuel pool structural concrete. The maximum accumulated neutron dose for the reracked pool ranges from $5x10^{15}$ to $9.8x10^{14}$ neutrons per square centimeter. Kilp-3a, ¶ 11; Turner-6 & 7, ¶ 39; Weber-3, ¶¶ 9, 10.

Radioactive decay of the unstable nuclides present in the spent fuel is the energy source responsible for the thermal loading in the fuel pool. Immediately after a normal refueling discharge into the fuel pool, the temperature of the pool water could rise to a maximum of 133.3° F. After 8 days this would then decrease to approximately 128° F. Weber-3, ¶ 12. A full core discharge into the pool could raise the temperature to a maximum of 150.8° F which would decrease to approximately 141° F within 9 days. Id., ¶ 13.

b. Effects on Non-Boraflex Materials

Licensee has estimated the "worst case" gamma and neutron doses to the fuel pool materials and compared those doses to the results of dose experiments on the same materials. The following Table illustrates the principal comparisons.

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RADIATION DOSE AND EFFECT

| Material | Estimated Max. Accum. Dose Over License Period | No Significant Effects Dose | References |
|---|---|---|---|
| Fuel cladding & fuel assemblies | Neutrons $5 \times 10^{15}$ n/cm2 | (see fn.*) | Kilp-3a, ¶¶ 11, 12 Kilp-3b, ¶ 5, Turner-6 & 7, ¶ 39 |
| | Gamma radiation $5 \times 10^{10}$ rads | No effect | |
| Steel liner & rack materials | Neutrons $5 \times 10^{15}$ n/cm2 | $10^{17-18}$ n/cm2 | Kilp-3a, ¶¶ 11, 12 Turner-6 & 7, ¶ 39 |
| | Gamma radiation $5 \times 10^{10}$ | No effect | |
| Concrete | Neutrons $9.8 \times 10^{14}$ n/cm2 | $3 \times 10^{20}$ n/cm2 | Weber-3, ¶ 9, 10 |
| | Gamma radiation $3 \times 10^{10}$ rads | $3 \times 10^{11}$ rads | |

*The maximum integrated fluence dose to the fuel cladding and fuel assemblies accumulated in the spent fuel pool over the license duration is estimated to be $5 \times 10^{15}$ neutrons per square centimeter. The difference between reactor neutron fluences and spent fuel pool fluences is several orders of magnitude. The added neutron exposure attributed to as long as 40 years in the spent fuel pool is equivalent to less than two minutes in the reactor when it is at full power. Kilp-3a, ¶ 9; Turner-6 & 7, ¶ 39. A typical fuel assembly will receive approximately $10^{22}$ neutrons per square centimeter during its stay in the reactor as compared to $10^{15}$ n/cm2 maximum over a period of 40 years in a spent fuel pool environment. Kilp-3b, ¶ 6.

The results of the comparisons demonstrate that the concrete, including imbedded steel, stainless steel storage racks, pool liner, fuel assemblies and cladding will not be affected in any significant way by the radiation exposure accumulated over the operating license period. Weber-3, ¶ 6-11, 17, 21; Kilp-3a, ¶¶ 4-12, 18, 19; Kilp-3b, ¶¶ 4-7, 13, 15.

Regarding temperature effects, Licensee demonstrates that adequate consideration was given in design and selection of non-Boraflex materials used in the fuel pool. Temperature over the range expected will cause no change in material properties and will have no detrimental effect on the integrity of the materials or the ability of the material to perform its intended function over the operating license period. Weber-3, ¶¶ 12-16, 17-19, 20, 22; Kilp-3a, ¶¶ 10, 13-19; Kilp-3b, ¶¶ 8-12, 14, 15.

3. Contested Issue: Boraflex

Colloquially referred to as a "poison", Boraflex is an effective entrapper of neutrons. It is produced by uniformly dispersing Boron carbide particles in a silicone-polymeric matrix. Singh-3 & 6, ¶ 12. Since the early 1980's Boraflex has been the "poison" of choice for high density fuel racks at many U.S. commercial nuclear power plants. Id. Licensee argues that the impacts of heat and radiation on the Boraflex have been adequately considered and describes test programs designed to confirm the neutron absorptive characteristics and identify the physical and chemical characteristics under a variety of radiation levels, radiation rates, and environments. Id., ¶¶ 15-20.

Last year in connection with the Turkey Point proceeding (Docket No. 50-250 and 50-251), the NRC published information concerning a potentially significant problem pertaining to the formation of gaps in the Boraflex absorber (separations of the neutron absorbing material) in high density storage racks. "Board Notification Regarding Anomalies in Boraflex Neutron Absorbing Material" BN-87-11, June 15, 1987; see also, NRC Information Notice 87-43, " Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks" (SSINS No. 6835), issued September 8, 1987. The problem was identified at the Quad Cities Plant, a commercial reactor with high density storage racks similar in design to the St. Lucie racks. The Joseph Oat Corporation manufactured the racks at both Quad Cities and St. Lucie.

Licensee argues that the problems identified at Quad Cities have been resolved and will not occur at St. Lucie. In brief, Licensee asserts that the principal reason for the cracking and gap formation following irradiation at the Quad Cities plant was excessive restraint of the Boraflex plate which was caused by the fabrication process and the use of adhesive during Boraflex installation. However, Licensee notes that the St. Lucie racks are designed to provide complete in-plane dimensional changes to Boraflex. Singh-3 & 6, ¶¶ 33-36. Intervenor disagrees that the problems identified at Quad Cities have been resolved and cites several sections of a Quad Cities report assessing the Boraflex performance. "Preliminary Assessment of Boraflex Performance in the Quad Cities Spent Fuel Racks", Report No. NET-042-01, Revision 0, dated April 10, 1987 (hereinafter "Quad Cities Report"). See pages 2-7 of Intervenor's Response to Licensee's Motion for Summary Disposition of Intervenor's Contention 3.

Licensee filed 71 statements of material fact as to which it claims there are no genuine issues to be heard with respect to intervenor Contention 3. Mindful of the standards of review discussed on pp. 5-6, supra, the Board agrees with and accepts Licensee's findings numbered 1-7, 9-52, and 62 through 67. Most, but not all, of these findings pertain to non-boraflex materials, and the Board concludes that summary disposition of Contention as regards non-boraflex materials is warranted.

As to the effect of radioactivity and heat on Boraflex, sufficient question is raised in the Intervenor's filings to require denial of summary disposition as to Boraflex. Accordingly, Licensee's motion for summary disposition is granted in part and denied in part. Because Licensee has not demonstrated that there are no outstanding safety problems regarding the performance of Boraflex, that issue remains to be resolved in Contention 3.

C. Contention 4

In Contention 4 Intervenor asserts

> That the high-density design of the fuel storage racks will cause higher heat loads and increases in water temperature which could cause a loss-of-cooling accident and/or challenge the reliability and testability of the systems designed for decay heat and other residual heat removal, which could, in turn, cause a major release of radioactivity into the environment.

Florida Power & Light Co., 27 NRC 452, 471 (LBP-88-10A, 1988). As bases for its contention, intervenor recites that

> a) The NRC has stated in numerous documents that the water in spent fuel pools would normally be kept below 122 degrees F. The present temperature of the water at the St. Lucie Plant, Unit No. 1 is estimated to be 110 degrees F. After the reracking, the temperature of the water would rise to 152 degrees F on a normal basis, and could reach 182 degrees F with a full core load added.
>
> b) There is also the possibility that a delay in the make-up emergency water could cause the zirconium cladding on the fuel rods to heat up to such high temperatures that any attempt at later cooling by injecting water back into the pool could hasten the heat up, because water reacts chemically with heated zirconium to produce heat and possible explosions. Thus, the zirconium cladding could catch on fire especially in a high-density design and create an accident not previously evaluated.

Amended Petition, p. 6. Intervenor did not submit any filing in opposition to Licensee's motion for summary disposition on this contention. However, as previously noted, we are constrained to examine Licensee's filings in support of its motion to determine that there is no genuine issue of material fact and that summary disposition is appropriate. 10 C.F.R. §2.749(b) (1988).

In support of its motion, Licensee has established the following material facts. Fuel assemblies, which are the source of heat generation in the spent fuel pool, are removed to the spent fuel pool after they have burned up a certain amount of the uranium oxide of which they are composed. However, the fission products accumulated in the fuel assembly continue to decay, producing the heat at issue here. That heat must be continuously removed. Singh-4 & 5, ¶ 9. The heat generated is greatest when the fuel rods are first transferred into the pool after removal from the reactor, so much greater that the heat generated by a single fuel assembly that has been in the pool only one hour is greater than the heat generated by 80 assemblies that have been in the pool for two years. Id., ¶¶ 10, 11 and Figure 1. Consequently, the increased storage capacity of the spent fuel pool, from 708 to 1706 cells, does not mean a correspondingly proportional increase in the total heat generated in the pool. Id., ¶ 11.

The spent fuel pool cooling system ("SFPCS") uses water to transport heat generated by the stored assemblies from the pool to the power plant heat sink. The heated water is drawn from the top of the pool at a steady rate, circulated through the tube side of a tubular heat exchanger where it exchanges heat with a closed loop, component cooling water system ("CCWS"), and is then returned to the bottom of the pool. Houghtaling-4 & 5, ¶ 5; Singh-4 & 5, ¶ 12, 13. Considerable quantities of heat are drawn off through evaporation and through the pool slab and walls, but these leakages are ignored for conservatism in the design of the cooling system. Singh-4 & 5, ¶ 19. Similarly the design assumes a deposit on heat transfer surfaces and ignores film coefficient benefits. Id., ¶¶ 20, 21.

The cooling system is comprised of two parallel pumps that together can pump 3,000 gallons per minute through the heat exchanger. One pump can handle a normal fuel assembly offload, but both are needed for a full core discharge. Id., ¶ 17. The CCWS can handle heat removal in abnormal conditions with only one loop. The CCWS is powered from independent, safety-related power supplies. These are loaded onto the emergency diesel generators in the event of a loss of offsite power. The SFPCS pumps are powered by independent electrical sources and can receive back-up power from the emergency diesel generators. The SFPCS is controlled and monitored from a local instrument panel supplemented by annunciator alarms in the control room to alert operators to any unusual conditions in the SFPCS or the fuel pool itself. Houghtaling-4 & 5, ¶¶ 6, 7, 9, 10.

Reracked Pool Temperatures

Licensee has conducted a heat transfer analysis of the fuel pool system for both a "normal batch" offload of fuel assemblies and a full core offload pursuant to the standards set out in the SRP and the NRC Branch Technical Position referenced in the SRP. The peak temperatures derived from that analysis are within the maximums permitted by the foregoing standards and descend rapidly after peaking. The NRC Staff's independent analysis corroborates Licensee's analysis. Singh-4 & 5, ¶¶ 23-25; SER-Amendment 91, p. 8.

Licensee also performed a loss of cooling analysis postulating a failure of the cooling system when the water in the pool had reached maximum temperature. Two significant conclusions were reached. First, the pool would not reach boiling for over 13 hours when loaded with a normal batch and 5 hours with a full core offload. Given the backup sources of makeup water, those time frames are adequate to take remedial action. Singh-4 & 5, ¶ 26. Second, even unattended pool boiling with a full core offload would require 46 hours before the 23 feet of water over the fuel assemblies would evaporate. Licensee concludes that under those circumstances ample time is available to remedy the situation. NRC Staff agrees. id.; Tourigny, ¶ 7.

Finally, Licensee concludes that boiling will not damage the fuel itself because of the effects of nucleate boiling, i.e., the formation of steam bubbles on the fuel rod surface which more efficiently transfers heat from the rods. Surface temperature would stabilize at 300 degrees F, "well below the temperature at which any cladding damage can occur." Id., ¶ 29, 30.

Given the time frames before the fuel assemblies would be exposed, the multiple sources of makeup water, the temperature of any makeup water, and the high temperatures required before cladding damage can occur, we find no basis for Intervenor's allegation that a delay in makeup water could cause any cladding damage, much less an explosion or fire. Houghtaling-4 & 5, ¶ 18. The electrical systems are remotely located and subject to regular testing under the Plant Technical Specifications. Id., ¶ 17. We perceive no scenario that would change the reliability and testability of the heat removal systems resulting from the high density reracking, and Intervenor has suggested none.

For all the foregoing reasons, we find no genuine issue of material fact with respect to Contention 4. Licensee has affirmatively demonstrated the safety of the spent fuel pool cooling system with respect to any possible threat from heat loads or loss of cooling accidents. Accordingly, Licensee's motion is granted as to Contention 4, and it is dismissed with prejudice.

### D. Contention 5

Admitted Contention 5 reads as follows:

> That the cooling system will be unable to accommodate the increased heat load in the pool resulting from the high density storage system and a full core discharge in the event of a single failure of any of the pumps or the electrical power supply to the pumps on the shell side of the cooling system and/or in the case of

> a single failure of the electrical power supply to the pumps on the pool side of the spent fuel pool cooling system. This inability will, therefore, create a greater potential for an accidental release of radioactivity into the environment.

Florida Power & Light Co., 27 NRC 452, 471 (LBP-88-10A, 1988). No basis was specified to support Amended Contention 5.

In admitting this Contention, the Licensing Board stated that the "Licensee's evidence on this contention should be directed toward applicability of and compliance with Criterion 44 of 10 CFR Part 50, Appendix A." Id. at p. 464. At the March 29, 1988 Prehearing Conference, Mr. Rich emphasized his concern over the alleged vulnerability of the electrical power supply, in particular vulnerability to the effects of humidity, wear, corrosion, elevated temperatures, and exposure to radiation on components. Tr. 80. Essentially, the Contention alleges that, if a pump or pump power supply fails, then the spent fuel pool cooling system will be unable to accommodate the increased heat load associated with the higher density fuel storage under full core discharge conditions. Id.

In its Motion for Summary Disposition, Licensee asserts that

> ... consistent with the requirements of Criterion 44 of Appendix A to 10 C.F.R. Part 50, the SFPCS is capable of maintaining the fuel pool temperature within acceptable limits, even under full core discharge conditions, with a single active failure on the shell and/or pool side of the heat exchanger. Even if a complete loss of forced cooling were to occur, the fuel would be kept covered and maintained at a safe temperature, given the multiple sources of

> makeup water and the long lead time before the pool water level could reach an unacceptable level.

Licensee's Facts, Contention 5, ¶ 16. We agree. Licensee's Affidavits and Statement of Material Facts were uncontroverted. As noted above, Intervenor failed to respond to Licensee's Motion for Summary Disposition on Contention 5.

Licensee first provides a complete description of the Spent Fuel Pool Cooling System ("SFPCS"). Briefly, it consists of two 1500 gallons per minute ("gpm") centrifugal pumps, a heat exchanger wherein the SFPCS rejects excess pool water heat to the component cooling water system ("CCWS"). The component cooling water, in turn, rejects its heat to the Intake Cooling Water ("ICW") system. Each of these systems has at least two independent loops. Houghtaling-4 & 5, ¶¶ 4-6, 10-11.

During normal operation, only one loop in the SFPCS is in operation. Id., ¶ 9. During abnormal conditions, such as full core discharge, both loops are placed in operation. Singh-4 & 5, ¶ 17. The CCWS has enough capacity that only one loop is needed to remove fuel pool heat during abnormal conditions. Houghtaling-4 & 5, ¶ 10.

The SFPCS is controlled and monitored from a local instrument panel. In addition, alarms are provided by annunciators in the control room sufficient to alert operators to any unusual conditions in the

SFPCS or the fuel pool itself. High and low CCWS flow is also annunciated in the control room. Id., ¶¶ 6, 9.

The SFPCS pumps are powered by independent electrical supplies, and are capable of receiving backup power from the emergency diesel generators. The CCWS components are powered from independent safety related power supplies. These are loaded onto the emergency diesel generators in the event of loss of offsite power. Id., ¶ 7.

A single active failure of a SFPCS pump or its power supply would reduce pool flow capability to that of one pump, but even in the case of the abnormal, full core offload situation, the pool water would remain within acceptable temperature limits as allowed in the SRP. A single active failure of a pump in the CCWS would not reduce cooling of the pool water because the CCWS is fully capable of removing the abnormal case heat load with only one pump operating. Id., ¶¶ 14, 15.

We further note that all components in the SFPCS and the CCWS are designed to operate continuously without degradation at temperatures above the expected maximum temperatures. The electrical equipment associated with the SFPCS and CCWS are remotely located and would not be affected by pool environmental conditions. All critical components of the systems are routinely tested in accordance with plant technical specifications. Houghtaling-4 & 5, ¶ 17.

Even if the highly unlikely event of loss of all forced cooling occurred, there would be adequate time available to obtain water from several sources: the refueling water tanks, the primary water tank, and even the city water supply. There is even water available through a crosstie with the ICW system which could supply 150 gpm to the pool which is more than adequate to maintain the pool level under abnormal conditions. Id., ¶ 8.

Given these multiple sources of makeup water even under the worst conditions, the fuel will remain covered and the bulk temperature of the pool water will not exceed boiling. Id., ¶ 18. The fuel cladding temperature will therefore be maintained well below the point where any fuel damage would occur, inasmuch as the pool is unpressurized and thus the bulk pool temperature could not exceed the boiling temperature. Id.; Singh-4 & 5, ¶¶ 27-30.

Therefore, as to Contention 5, we conclude that the above material presented by the Licensee establishes that there are no genuine issues of material fact and that Licensee is entitled to judgment as a matter of law. Licensee's Motion for Summary Disposition of Contention 5 is granted, and it is dismissed with prejudice.

E. Contention 6

Contention 6 avers:

> The proposed use of high-density racks designed and fabricated by the Joseph Oat Corporation is utilization of an essentially new and unproven technology.

Florida Power & Light Co., 27 NRC 452, 471 (LBP-88-10A, 1988). As bases for the contention Intervenor asserted that:

> As recently as September 8, 1987, the NRC has provided information concerning these racks to all nuclear power reactor facilities warning of a "... potentially significant problem pertaining to gaps ... ." "The concern is that separation of the neutron absorbing material used in high-density fuel storage racks might compromise safety." (NRC Information Notice No. 87-43. SSINS NO.: 6835). Again on October 23, 1987, the NRC is requiring more information of FP&L in order to assess the integrity of the Boraflex system. The answer to this latest inquiry has not yet been made available to the public.
>
> FP&L's response to these and other problems relating to the use of Boraflex incorporated in a system designed by the Joseph Oat Corporation represents an essential modification of the current technology to such an extent that it, in fact, represents utilization of a new technology and fabrication process that is, thus, unproven and untested.

Amended Petition, p. 8.

Licensee filed two affidavits in support of its motion for summary disposition: "Affidavit of Dr. Stanley E. Turner on Admitted Contentions 6 and 7", and "Affidavit of Dr. K. P. Singh on Admitted Contentions 3 and 6." See fn. 2, supra. Dr. Singh's affidavit was proffered to demonstrate that the problems encountered with the use of Boraflex in high-density storage racks at other nuclear power plants have been resolved and will not occur at St. Lucie 1. Dr. Turner's affidavit addresses criticality analyses and conformance with NRC criteria and applicable industry standards.

NRC filed the affidavit of Edmond G. Tourigny which supported Licensee's motion. Mr. Tourigny stated that on October 29, 1987, he visited the Joseph Oat Corporation and observed racks during various stages of fabrication and agrees with Licensee that all aspects of rack construction embody proven design concepts and well-established fabrication techniques and that the racks incorporate proven technology for Boraflex installation and positioning. He concludes by stating that he does not believe that the use of high-density racks designed and fabricated by the Joseph Oat Corporation constitutes the use of a new and unproven technology. Tourigny, ¶ 9.

Licensee argues that the aspects of rack technology employed for producing the St. Lucie 1 racks utilize a proven and widely applied technology, and reflect established industry practice. Singh-3 & 6, ¶ 22. Licensee states that the rack manufacturer, the Joseph Oat Corporation, has extensive experience in the manufacturing of spent fuel storage racks using Boraflex panels, that all significant construction features of the racks are direct adaptations of established technology and the production control methods in use at the Joseph Oat Corporation are derived from two decades of nuclear component manufacturing. Id., ¶¶ 31, 32. Licensee also states that the only departures from established industry practice in the construction of the St. Lucie racks were the elimination of the silicone based adhesive used to cement the Boraflex in place and modification of the manufacturing process so as to install Boraflex in place with minimal surface loading. Id., ¶ 30. It is precisely those changes that Intervenor focuses on in its claim that the use of the St. Lucie Unit 1 racks constitutes utilization of an essentially new and unproven technology. See, Intervenors Response to Licensee's Motion for Summary Disposition of Intervenor's Contention 6, pp. 2-12.

Licensee filed 35 statements of material fact as to which it claims there are no genuine issues to be heard with respect to Intervenor Contention 6. Many of these findings were filed with the motion for summary disposition of Contention 3 and were rejected there on the basis of questions raised by Intervenor. The Board agrees with and accepts Licensee's findings numbers 1, 7, 8, 12, 16, 19, 20, 22, 27, and 29. Because Licensee has not established that there are no outstanding safety issues concerning the use of Boraflex the remaining findings state the subject matter for evidentiary hearing where Intervenor will be provided an opportunity to pursue questions raised as to those issues. Licensee's motion for summary disposition of Contention 6 is denied.

F. Contention 7

Admitted Contention 7 reads as follows:

> That the increase of the spent fuel pool capacity, which includes fuel rods that are more highly enriched, will cause the requirements of ANSI-N16-1975 not to be met and will increase the probability that a criticality accident will occur in the spent fuel pool and will exceed 10 CFR Part 50, A 62 criterion.

Florida Power & Light Co., 27 NRC 452. 471 (LB-88-10A, 1988). The bases for the contention read as follows:

> The increase in the number of fuel rods stored and the fact that many of them may have experienced fuel failure or may be more highly enriched and have more reactivity will increase the chances that the fuel pool will go critical, and cause a major criticality accident and perhaps, explosion that will release large amounts of radioactivity to the environment in excess of the 10 CFR 100 criteria.

Amended Petition, p. 11. We amended Intervenor's contention as originally filed to delete reference to failed fuel, and admitted the contention. Florida Power & Light Co., 27 NRC 452, 468. In so admitting this contention, we stated that "[c]riticality control is one of the basic concerns when fuel is being stored, and the methods used to achieve this control are of great importance." Id.

Licensee states that

> It can be concluded that the design of the St. Lucie 1 storage racks conforms to safe and conventional practice in the industry, conforms to all applicable regulations and guidance, and provides assurance that a criticality accident cannot occur under any credible postulated accident condition. Turner Affidavit, ¶ 72.

Licensee's Facts, Contention 7, ¶ 29. We agree in large part, but will require clarification of two items noted below.

Licensee describes the pool design and neutronics behavior, as determined by analysis. Turner-6 & 7, ¶¶ 4-13. Licensee then describes the various regulations pertaining to fuel storage pools and NRC guidance for achieving the necessary conditions for meeting these requirements. Id., ¶¶ 17-19, 22, 23. Licensee's analyses' in light of that regulatory guidance were described in some detail. The results of the analyses showed that the maximum calculated values for k-eff in the pool were 0.9409 for Region 1 and 0.9435 for Region 2. Id., ¶¶ 23, 25, 28-30. These values are within regulatory requirements.

Licensee also analyzed the following conditions as potential causes of accidents: increased temperature, boiling, dropped assembly, and abnormal fuel location. Turner-6 & 7., 32. Licensee concluded that a k-eff will be maintained less than 0.95 under all credible accident conditions. Id., ¶ 42.

Licensee states that the largest reactivity effect would result from a fresh fuel assembly of 4.5 per cent enrichment being loaded into a Region 2 storage cell. Id., ¶ 33. Licensee's Facts, Contention 7, ¶ 25 is dependent upon restricting storage of different enrichment fuel in the proper region. Intervenor argues that the 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". Intervenor's Finding (Contention 7), ¶¶ 6, 12. Licensee has not shown how erroneous insertion could be avoided. We find that a full explanation of the measures that are taken by the Licensee to comply with this Standard Review Plan requirement must be provided.

Licensee states that even if "somehow the pool was allowed to drain criticality would not occur ... since, without water, there would be no neutron moderator in the pool--a necessary condition of criticality. Weber Affidavit-1, 6." Licensee's Facts, Contention 7, ¶ 24. Intervenor, in his "Intervenor's Response to Licensee's Motion for Summary Disposition of Intervenor's Contention 7", ¶ 9, questions this position, citing instances where criticality has occurred in the absence of water. We can find no factual evidence in the record to support Licensee's position; it is presented as a bald statement with no explanation as to why absence of a moderator will prevent criticality. We cannot accept Licensee's statement without adequate explanation of why moderation is necessary to achieve criticality.

In conclusion, we accept Licensee's Facts, Contention 7, ¶¶ 2-21, 26-28.[5] We reject Licensee's Facts, Contention 7, ¶¶ 22-25 pending further explanation as stated in our above consideration. Accordingly, Licensee's motion for summary disposition of Contention 7 is granted in part and denied in part as set out above.

## V. CONCLUSION

We find that there is no genuine issue of material fact and that Licensee is entitled to judgment as a matter of law with respect to:

1. Contentions 1, 4, and 5;
2. The non-Boraflex materials issues in Contention 3;
3. All issues in Contention 7 except:
    a. the danger of improper storage of fresh fuel assemblies; and
    b. the danger of a criticality accident in the absence of a moderator.

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[5]Licensee's Facts 1-3, 5, and 7-9 are not addressed here. They relate to the premise that the use of Boraflex will perform its intended function. The question of the suitability of Boraflex will be litigated under Contention 3 and 6, and will not be repeated under Contention 7.

Licensee's motion denied as to Contention 6 and will be granted as to all other Contentions with the exception of the matters listed above.

Pursuant to 10 C.F.R. §2.743(b), the parties shall file all direct written testimony of witnesses with the Board and serve copies on each other on or before November 22, 1988. Filed and served means received. Hearing on the issues remaining in this proceeding will commence on Tuesday, December 6, 1988 at 9:00 A.M. in the vicinity of the St. Lucie Plant. The parties will be notified of the location on or about November 7, 1988.

ORDER

For all the foregoing reasons, and upon consideration of the entire record in this matter, it is, this 14th day of October, 1988

ORDERED

1. That Licensee's Motion for Summary Disposition of Intervenor's Contentions is granted as to Contention 1, part of Contention 3, Contention 4, Contention 5, part of Contention 6, and part of Contention 7;

2. That Licensee's Motion for Summary Disposition is denied as to the issues concerning Boraflex in Contentions 3 and 6, and the

issues concerning the improper storage of fresh fuel assemblies and the danger of a criticality accident in the absence of a moderator in Contention 7; and

3. That hearing on the remaining matters at issue will commence on December 6, 1988 in the vicinity of the St. Lucie Plant.

ATOMIC SAFETY AND LICENSING BOARD

B. Paul Cotter, Jr.

B. Paul Cotter, Jr., Chairman
ADMINISTRATIVE JUDGE

Glenn O. Bright

Glenn O. Bright
ADMINISTRATIVE JUDGE

Richard F. Cole

Dr. Richard F. Cole
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 14th day of October 1988.