ATTACHMENT A

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-335-OLA
Florida Power and Light Company Company	)	
	)	ASLBP No. 88-560-01-LA
St. Lucie Plant, Unit No. 1	)	
	)	

Affidavit of Edmond G. Tourigny
In Support of Motion for Summary
Disposition

Mr. Tourigny being duly sworn according to law states as follows:

1. My name is Edmond G. Tourigny. I am employed in the Office of Nuclear Reactor Regulation (NRR) of the U.S. Nuclear Regulatory Commission. I am presently the Project Manager for Florida Power and Light Company's St. Lucie Plant, Unit Nos. 1 and 2. I have been the Project Manager for the St. Lucie Plant since May 1986. A summary of my qualifications and experience is attached hereto as Exhibit A, which is incorporated herein by reference.

2. The responsibilities of my present position include reviewing changes to the Technical Specifications (TS) proposed by the licensee, preparing notices of Significant Hazards Consideration, preparing safety evaluations, environmental assessments (when required), and environmental impact statements (when required) on the proposed change(s), and if the proposed change is acceptable, preparing the license amendment and supporting documentation for issuance by the Project Director. Assistance in preparing the safety evaluation and environmental assessment or environmental impact statement (when required) is obtained, as appropriate, from technical specialists in NRC's Office of Nuclear Reactor Regulation.

   I managed the safety and environmental review of the St. Lucie spent fuel pool rerack amendment request. In addition, I wrote parts of the safety evaluation and environmental assessment. Because of the complexity of review in this matter, a team of technical specialists was assembled to evaluate the licensee's submittals. This team was augmented by contractual support from Brookhaven National Laboratory. The team consisted of personnel highly experienced in structures, geosciences physics, radiation, dose assessment, thermal-hydraulics, chemistry and materials. The team members performed their own analyses of various aspects of the licensee's submittal. For example, one member of the team performed independent calculations of radiation dose to members of the public under normal and accident conditions. One team member performed

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his own physics analysis. One team member independently calculated pool heating and cooling under various scenarios. Thus, in all significant matters, the team was able to evaluate on its own the acceptability of the rerack of the spent fuel pool. I reviewed the work of every team member and agreed with their analyses and conclusions. The results of the team's efforts were then summarized in the safety evaluation and environmental assessment which supported authorizing the licensee to rerack the spent fuel pool.

3. The purpose of this affidavit is to respond to the contentions admitted by the Atomic Safety and Licensing Board in its April 20, 1988 Memorandum and Order, and to support Licensee's Motion of August 5, 1988 for Summary Disposition. The admitted contentions 3, 4, 6, 8, 9, 11 and 15 were renumbered as 1 through 7, respectively.

   As part of the discovery process, the intervenor was afforded the opportunity to respond to the licensee's interrogatories. I have reviewed the intervenor's response and concluded that no new information has been identified, nor has the intervenor identified any witnesses. Based upon these considerations, the staff's safety evaluation and environmental assessment continues to serve as an adequate basis to authorize the spent fuel pool rerack. It should be noted that the rerack effort has been completed by the licensee.

4. Admitted Contention 1

   Admitted Contention 1 reads as follows:

   Admitted Contention 1

   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 the 10 CFR Part 100 criteria. (Originally Amended Petition Contention 3.)

Memorandum and Order, Appendix A. p. 1, (April 20, 1988). The bases for the contention read as follows:

Bases for Contention

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.

Amended Peition, p. 4.

In admitting this Contention, the Licensing Board stated that

> Licensee's response . . . should show that its analysis of a cask drop accident bounds those uncertainties that are identified in the BNL Report and listed as the bases for this contention. Thus, by such conservatisms and analysis, Licensee must demonstrate compliance with 10 CFR Part 100 (1987).

Memorandum and Order, p. 13, (April 20, 1988).

The intervenor contends that a cask drop accident will cause radiological consequences greater than 10 CFR Part 100 criteria. The Brookhaven National Laboratory (BNL) report on Severe Accidents is used as the basis for the contention. The BNL report assumed a number of events would occur in order to postulate significant radiological releases. Such events include failure of the pool, thereby causing rapid loss of all contained cooling water, eventual fuel heatup, and a subsequent zirconium cladding fire.

I do not believe that a cask drop accident will cause a radiological release greater than 10 CFR 100 guidelines, provided that the licensee adheres to Technical Specification 3.9.13 which reads "The maximum load which may be handled by the spent fuel cask crane shall not exceed 25 tons" and Technical Specification 3.9.14 which reads "The irradiated fuel assemblies in the fuel storage pool shall have decayed for at least 1180 hours, unless more than one-third [of the] core is placed into the pool, in which case the irradiated fuel assemblies shall have decayed for 1490 hours."

The basis for this maximum load technical specification reads, in part, "structural damage caused by dropping a load in excess (sic) of a loaded single element cask could cause leakage from the spent fuel pool in excess of the maximum makeup capability." The 25 ton limit was placed in the technical specifications when the plant was licensed in 1976 as a result of staff analysis of the spent fuel pool structure. Thus this limit assures that the initial initiating event described in the BNL report (failure of the pool thereby causing rapid loss of all contained water) will not occur.

The basis for the spent fuel decay time technical specification reads, in part, "the minimum requirements for decay of the irradiated fuel assemblies in the entire spent fuel storage pool prior to movement of the spent fuel cask into the fuel cask compartment ensure that sufficient time has elapsed to allow radioactive decay of the fission products." The decay time was reevaluated as part of the spent fuel pool rerack request and found still to be acceptable. Given a 1490 hour decay time and assuming the cask would damage all the spent fuel in the pool (an impossible event[1] but an acceptable assumption

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(1) The spent fuel cask crane cannot carry the cask over most of the spent fuel pool because the door of the fuel handling building limits cask entry into only a portion of the building; the cross sectional area of the cask would also be smaller than the cross sectional area of the spent fuel pool.

from a design basis perspective), the resulting dose to an individual at the exclusion area boundary will be 21 rem to the thyroid and less than 0.1 rem to the whole body. These doses are well below the 10 CFR 100 guideline doses of 300 rem to the thyroid and 25 rem to the whole body.

I have reviewed the affidavits provided on behalf of the licensee in response to this contention and I agree with the conclusions that a cask drop accident would cause radiological releases within the dose guidelines specified in 10 CFR Part 100.

In summary, a cask drop accident will not cause radiation doses to exceed 10 CFR 100 guidelines provided that the technical specifications on allowable cask loads and minimum spent fuel decay time are adhered to.

5. Admitted Contention 2

Admitted Contention 2 reads as follows:

Admitted Contention 2

> That the consequences of a cask drop accident or an accident similar in nature and effect are greatly increased due to the presence of a large crane to be built inside the spent fuel pool building in order to facilitate the reracking. (Originally Amended Petition Contention 4).

The licensing Board issued an Order on July 27, 1988, dismissing Admitted Contention 2" with prejudice as moot."

6. Admitted Contention 3

Admitted Contention 3 reads as follows:

Admitted Contention 3

> 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. (Originally Amended Petition Contention 6).

Memorandum and Order, Appendix A, p. 1, (April 20, 1988).

The bases for the contention state:

Bases for Contention

> 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:

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 as result of exposure to increased heat over extended periods of time.

Amended Petition, pp. 5-6. Of this contention, the Licensing Board said:

> Petitioner argues that the pool was designed to store lesser quantities of spent fuel for a shorter period of time and that licensee has failed to adequately analyze problems that may result from exposure to the increased amount of decay heat and radiation emitted by the larger number of spent fuel assemblies stored. Petitioner specifies three problems: (1) deterioration of fuel cladding; (2) loss of integrity of materials making up the storage rack and the pool liner; and (3) deterioration of the concrete of which the pool is constructed. Amended Petition, 5-6. 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."

Memorandum and Order, page 17 (April 20, 1988). The Licensing Board limited the scope of the contention to the length of time authorized by the licensing amendment at issue.

The intervenor believes that the long-term performance of the materials associated with the spent fuel pool has not been demonstrated. In the basis statement, the fuel cladding, storage racks, pool liner, and pool structure concrete were specifically noted. The fuel cladding in the St. Lucie case is zircaloy and the rack and liner material is stainless steel.

I believe that the successful long-term performance of spent fuel pool materials has been demonstrated as reported in a number of Commission documents, particularly in the Final Generic Environmental Impact Statement on Handling and Storage of Spent Fuel Light Water Power Reactor Fuel, NUREG-0575, August 1979*

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*This was a major source document used in the preparation of the staff's Environmental Assessment in support of the Amendment Request.

and the Waste Confidence Decision and Requirements for License Actions Regarding the Disposition of Spent Fuel Upon Expiration of Reactor Operating Licenses, Federal Register Notice, Volume 43, No. 207, Friday, August 31, 1984, beginning at 34658. In addition, the staff specifically focused its attention on material compatability and chemical stability of the stainless steel racks and Boraflex in its Safety Evaluation in support of the amendment.

The following statement was contained in NUREG-0575, Appendix H, Section 3.0 entitled Cladding Stability During Storage of Spent Fuel: "Fuel handling experience in the U.S., going back to 1959, has not revealed any instance where zircaloy clad uranium oxide fuel has undergone observable corrosion or other chemical degradation during pool storage."

The following statements were contained in the staff's safety evaluation in Section 3. entitled "Material Compatability and Chemical Stability." "The austenitic stainless steel used in the spent fuel storage racks is not susceptable to stress corrosion cracking and thus, corrosion in the spent fuel storage pool environment should be of little significance during the life of the plant. The spent fuel pool water is processed by filtration and demineralization to maintain water purity and clarity. Dissimilar metal contact corrosion (galvanic attack between the stainless steel rack assemblies and zircaloy in the fuel assemblies) should not be significant because the materials are protected by highly passivating oxide films and are, therefore, at similar galvanic potentials". Similar statements could be made for the pool liner. In regard to Boraflex, "Qualification tests have shown that Boraflex does not possess leachable halogens that could be released into the spent fuel pool water in the presence of radiation. Similar conclusions have been made regarding the leaching of boron from the Boraflex." The staff has reviewed the proposed surveillance program for monitoring the Boraflex in the St. Lucie Plant spent fuel storage pool and concludes that the program can reveal deterioration that may lead to loss of neutron absorbing capability during the life of the spent fuel racks. In the unlikely event of Boraflex deterioration, the monitoring program will detect such deterioration and the licensee will have sufficient time to take corrective action.

The reinforced concrete which supports the liner, pool water, racks, and spent fuel assemblies is massive in the X-Y plane and in the vertical direction below the pool. There will be no deterioration of the concrete as a result of the reracking.

The Commission's Waste Confidence Decision goes one step further (as related in this matter which addresses the operating license life of the facility) in considering the long term effects of spent fuel storage. The fourth Commission Finding states "The Commission finds reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations." (underlining added for emphasis). In support of this finding an analysis was presented on the long-term integrity of spent fuel under water pool storage conditions in addition to an analysis of structure and component safety for extended facility operations for storage of spent fuel in water pools.

Based upon the undisputed facts set forth above, the materials associated with the St. Lucie pool and its structure have been adequately evaluated and will perform their designed function for a few decades.

I have reviewed the affidavits provided on behalf of the licensee in response to the contention. The analyses present a more specific case for the St. Lucie pool and structure materials, whereas, I took a more generic approach. I agree with the licensee that (1) the reinforced concrete spent fuel pool structure will withstand the radiation and heat levels expected as a result of the rerack, (2) the structure will withstand the expected thermal loads, (3) there are no materials degradation concerns for the stainless steels in the pool liner and storage racks, (4) any radiation heat effects on the fuel cladding and fuel assembly materials attributable to increasing the capacity of the pool are negligible when compared to prior reactor exposure, and (5) the suitability of using Boraflex as a neutron absorber for the storage of spent fuel in the pool has been demonstrated.

Based upon the above, I believe that materials used in the pool and structure will fulfill their design function for the length of time authorized by the license amendment at issue.

7. Admitted Contention 4

Admitted Contention 4 reads as follows:

> 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. (Originally Amended Petition Contention 8).

Memorandum and Order, Appendix A, p. 1 (April 20, 1988). The bases for Admitted Contention 4 are stated as follows:

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 temperature 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, pp. 6-7.

In admitting this Contention, the Licensing Board stated that it expects the Intervenor to present direct technical testimony for the record. Memorandum and Order, p. 20 (April 20, 1988). In this regard, the Intervenor clarified his contention during the March 29, 1988 Prehearing Conference by stating that his basic concern with the pool cooling system "comes down to technical calculations." Tr. 69. Intervenor further indicated that he or his experts will provide substantial technical evidence that will show (1) the Licensee's temperature calculations are inadequate and will be exceeded, and (2) the Licensee's calculations do not adhere to the guidelines as set forth in the SRP, Section 9.1.3. Id.

The intervenor contends that the rerack 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. The above sequence of events would lead to a major release. The basis for the contention consists of higher than normal pool water temperature and the possible delay in makeup water leading to zirconium cladding fire.

The spent fuel pool water temperature changes with time and is dependent upon how much spent fuel is placed in the pool, the decay time of each individual assembly before it is placed in the pool, and whether one or two spent fuel pool pumps are operating with the single heat exchanger.

The maximum pool water temperature guidelines allowed by the Standard Review Plan (SRP) are 140°F under normal refueling conditions (approximately 1/3 core offload) and below boiling for an abnormal refueling condition (full core offload). A single active failure to the system is assumed in the normal refueling condition and a single active failure need not be considered under the abnormal refueling condition. Calculations were performed for both of these conditions and the results showed that the acceptance criteria of the SRP would be met. The maximum water temperature (as a function of time) was determined to be 134°F for the normal refueling condition and 167°F for the abnormal refueling condition. These temperature calculations assumed only one of two pumps were operating. I am in agreement with the staff calculations and resultant values. Thus, there would be no loss of coolant because of boiling under these conditions and the reliability and testability of the systems designed for decay heat removal would not be affected.

The spent fuel pool cooling system at the St. Lucie Plant, Unit No. 1 is not seismic Category I. As such, the SRP identifies the need to address this contingency should an earthquake occur. This SRP contingency can be considered similar to the intervenor's concern of delay of makeup emergency water, should it be needed. Implicit with a delay in makeup water is the assumption that the cooling system has failed. The same two conditions postulated above were assumed coincident with an earthquake which was assumed to disable the spent fuel pool cooling system. I and the staff reviewed the licensee's analysis. The calculated boil-off rates and time to boil were estimated to be 33.9 gpm and 16.79 hours for the normal refueling condition and 69.5 gpm and 7.47 hours for the abnormal refueling condition. In addition, there is a significant delay between the time the pool starts boiling and the water level recedes appreciably. The licensee calculated results are acceptable.

The licensee identifies a number of makeup water sources. These are the Refueling Water Tank, Primary Water Tank, City Water Tank, and Seismic Category I, Intake Cooling Water System. The Intake Cooling Water System can provide 150 gpm makeup in a reasonable period of time. The three pumps associated with the system can be powered by the emergency diesel generators and any one pump can fulfill the function. Thus, I conclude that adequate makeup can be provided to the pool so that the fuel would not be put in jeopardy in the highly unlikely event of failure of the cooling system.

I have reviewed the affidavits provided on behalf of the licensee in response to this contention and I agree with the conclusions that operation of the spent fuel pool cooling system will maintain pool temperatures below the values stated in the SRP and should forced cooling be lost and boiling occurs, the fuel would remain well covered with water, at a safe temperature.

It can be concluded that the pool water temperature can be adequately controlled and cooling of the spent fuel can be adequately accomplished under various refueling conditons, even assuming an earthquake or other initiating event causing all loss of forced pool cooling.

8. Admitted 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 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. (Originally Amended Petition Contention 9).

Memorandum and Order, Appendix A, p. 2 (April 20, 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." Memorandum and Order, p. 22 (April 20, 1988). At the March 29, 1988 Prehearing Conference, Intervenor emphasized his concern over the alleged vulnerability of the electrical power supply; in particular, to the effects which humidity, wear, corrosion, elevated temperatures and exposure to radiation would have 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 discharge conditions.

Memorandum and Order, p. 21 (April 20, 1988).

The contention interrelates with Admitted Contention 4 because heat loading and single failure analysis are addressed in the calculation of pool water temperatures under normal refueling conditions, abnormal refueling conditions, and loss of all spent fuel pool cooling. The contention goes one step further and raises the concern of accidental release of radioactivity into the environment. As stated in the previous contention, the SRP does not assume a single active failure in the abnormal refueling condition.

I concluded in the previous contention that the spent fuel pool cooling system will maintain pool temperatures below the values stated in the SRP and should forced cooling be lost and boiling occurs, the fuel would remain well covered with water, at a safe temperature.

The radiological dose case which fully addresses the contention is the abnormal refueling condition coupled with complete loss of the spent fuel pool cooling system. This is a very unlikely event because full core offloading is typically performed only a few times over the life of the facility and the spent fuel pool cooling system is highly reliable. Makeup water is supplied to the pool to compensate for boil off. The makeup water could be supplied by the Seismic Category I Intake Cooling Water system which can be powered by the emergency diesel generators.

The results of this analysis were previously sent from Mr. B. H. Vogler to the intervenor by letter dated April 15, 1988. I prepared the enclosure to the letter and I will summarize the results here. The total radiation dose to the thyroid from iodine exposure was calculated to be approximately 0.1 rem. The total radiation dose to the whole body was calculated to be approximately 10^{-5} rem. The radiation dose values are applicable to both the exclusion area boundary and the low population zone boundary. These values are well below the 10 CFR 100 guidelines doses of 300 rem to the thyroid and 25 rem to the whole body.

I have reviewed the affidavits provided on behalf of the licensee in response to the contention. The affidavits also focused on electrical power supply failure considerations for various systems. As specified above, the SRP does not call for assumption of a single active failure for the abnormal refueling condition case. Thus, the affidavits provided on behalf of the licensee go beyond the SRP guidance in this case. The affidavits did not address the expected radiological doses as a result of pool boiling. My affidavit does. I agree that the spent fuel pool water will be maintained within acceptable limits and if complete loss of forced cooling were to occur, the fuel would be kept covered and maintained at a safe temperature.

In conclusion, and as concluded in Admitted Contention 4, the pool water temperature can be adequately controlled and cooling of the spent fuel can be adequately accomplished under various refueling conditions, even assuming an earthquake or other initiating event causing loss of all forced pool cooling. In addition, in the highly unlikely event of loss of forced pool cooling, the radiation doses would be very small.

9. Admitted Contention 6

Admitted Contention 6 reads as follows:

Admitted Contention 6

> The proposed use of high-density racks designed and fabricated by the Joseph Oat Corporation is utilization of an essentially new and unproven technology. (Originally Amended Petition Contention 11).

Memorandum and Order, Appendix A, p. 2 (April 20, 1988). The bases for the contention read:

Bases for Contention

> As recently as 8 September 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 23 October 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 Corp. 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, page 8. In admitting this contention the Board stated:

> While the use of Boraflex may not be considered "new technology," the problems identified in the NRC Staff Board Notifications concerning the reports on the Quad Cities and Point Beach plants raise specific questions about the use of Boraflex in the Joseph Oat storage racks.

Memorandum and Order, pg 24 (April 20, 1988).

The intervenor contends that the use of high-density racks designed and fabricated by the Joseph Oat Corporation is utilization of an essentially new and uproven technology. The basis of the contention hinges on problems found with Boraflex, a neutron absorption material that is an integral part of the racks.

The use of the high-density racks designed and fabricated by the Joseph Oat Corporation is not a utilization of a new and uproven technology. I also acknowledge that Boraflex gap problems were found elsewhere. The Boraflex is used for criticality control. Boraflex has been used in racks fabricated by various companies including Joseph Oat Corporation since the early 1980's.

The St. Lucie racks were fabricated using proven technology. I visited the Joseph Oat Corporation on October 29, 1987 and observed racks during various stages of fabrication. Stainless steel half channels were mated together to form a channel or box. A Boraflex strip was placed on each side of a Region I channel and secured by a cover plate. Region II channels do not have cover plates and the Boraflex is held in place by the mating of two channels. The channels are subsequently welded together; the module is subsequently welded to the base plate and girdle bars are attached. Quality control personnel independently checked all the above described steps.

The Boraflex gap problem identified elsewhere should not occur at St. Lucie. The gap problem was due to the restraining of the Boraflex during the fabrication of the racks and that upon irradiation, the Boraflex shrunk and gaps formed because of restraint on parts of the Boraflex. In the St. Lucie case, the use of adhesives in the attachment of slightly oversized Boraflex to the rack cell was not permitted and the manufacturing process avoided techniques that could pinch the Boraflex. Therefore, the Boraflex should be able to slightly shrink unrestrained under irradiation. Lastly, I consider this change of technique in placing the Boraflex on the channel and not permitting restraint to be a refinement of the whole fabrication process. It, in itself, does not make the entire fabrication process a new and uproven technology.

The licensee has developed a surveillance program to verify the performance of the Boraflex. In the event of any observed deterioration of the specimen that could have an effect on nuclear criticality, an immediate inspection of the poison panels in the rack will be performed. The NRC will be advised accordingly. I believe the surveillance program as outlined in the FP&L submittal of October 20, 1987, to the staff to be acceptable. In addition, in the highly unlikely event that the Boraflex will not be able to fulfill its design function at some time in the future, the licensee's implementation of the corrective actions described in the October 20, 1987 submittal will suffice.

I have reviewed the affidavits provided on behalf of the licensee in response to this contention, and I agree that all aspects of rack construction embodies proven design concepts and well-established fabrication techniques and that the racks incorporate proven technology for Boraflex installation and positioning. Lastly, the surveillance program is adequate to detect greater than expected degradation and sufficient opportunity is available to take any corrective action that may be necessary.

In summary, I do not believe that the use of high-density racks designed and fabricated by Joseph Oat Corporation to be an essential new and unproven technology.

10. Admitted Contention 7

Admitted Contention 7 reads as follows:

Admitted Contention 7

> 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. (Originally Amended Petition Contention 15).

Memorandum and Order, Appendix A, p. 2 (April 20, 1988). The bases for the contention read as follows:

Bases for Contention

> 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 into the environment in excess of the 10 CFR 100 criteria.

Amended Petition, p. 11. The Licensing Board amended Intervenor's contention as originally filed to delete reference to failed fuel, and admitted the contention. See Memorandum and Order, p. 28 (April 20, 1988).

In admitting this contention, the Licensing Board 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. The contention is therefore admitted." Memorandum and Order, p. 28 (April 20, 1988).

The intervenor is concerned that a criticality accident has a higher probability of occurring as a result of increasing the spent fuel capacity and allowing higher enriched fuel in the pool.

Criticality consideration is a prime input in the design of the spent fuel racks. Since the spacing between the fuel assemblies in a given rack is now smaller and higher enriched fuel would be permitted, a neutron absorbing medium (Boraflex in this case) for both Regions must be used in order to meet regulatory requirements. In addition, credit for burnup is also permitted for the Region II racks. The K_{eff} of an individual fuel assembly decreases while the fuel assembly is producing energy in the reactor.

The SRP calls for K_{eff} to be not greater than 0.95 for normal storage conditions assuming that all the racks contain fuel assemblies and are fully covered with water. No credit is allowed for the boron in the pool water, but credit may be taken for the neutron absorbing capability of Boraflex in both regions and for fuel burnup in Region II. Credit may be taken for the boron in the pool water

for an abnormal storage condition since the assumption of two independent and concurrent failures need not be considered. The K_{eff} would decrease for most abnormal storage conditions but could increase for some. For example, a loss of the cooling system will result in an increase in pool water temperature but K_{eff} will decrease. Conversely, if a fresh assembly was placed in a Region II storage location, the K_{eff} would increase. However, the boron in the water is worth approximately 0.2 in K_{eff} and this would more than offset this abnormal storage condition.

The staff's review of the licensee's physics input parameters (including using a maximum of 4.5 w/o fuel), methodology (including uncertainty analysis), and results, concluded that the K_{eff} will be no greater than 0.95. The value calculated by the licensee was 0.9409 for Region I and 0.9435 for Region II. The staff concluded that most abnormal storage conditions will not result in an increase in K_{eff} but for some events, the K_{eff} could increase; however, the K_{eff} decrease caused by boron in the water more than offsets the reactivity addition caused by credible accidents. I am in agreement with the staff's conclusions and no credible criticality accident is possible.

The staff and I believe that subcriticality control is so important that technical specifications have been put in place. Technical Specification 5.6.1 requires that (1) a boron concentration of at least 1720 ppm will be kept in the pool water, (2) non-temporary placement of spent fuel in Region II will only be permitted if the assembly meets the burnup requirements of Figure 5.6-1, and (3) all fuel assemblies must have an initial U-235 enrichment less than or equal to 4.5 weight percent.

I have reviewed the affidavits provided on behalf of the licensee in response to this contention. I agree with the licensee that the design of the St. Lucie 1 storage racks provide assurance that a criticality accident cannot occur under any credible postulated condition.

Based upon the above discussion, I conclude that a criticality accident will not occur at the St. Lucie Unit No. 1. The applicable technical specifications must continually be met.

11. In summary, I am satisfied that the St. Lucie Unit 1 Spent Fuel Pool racks were adequately designed, fabricated, installed, and will operate as a safe spent fuel storage system until the expiration of the license in 2016 and that all NRC requirements have been met. I am satisfied that the contentions have been adequately addressed, and I support the licensee's Motion for Summary Disposition.

The foregoing statement are true and correct to the best of my knowledge and belief.

Edmond G. Tourigny

Subscribed and sworn to before me this 30th of August, 1988

Notary Public

My commission expires: 7/1/90

Exhibit A

Edmond G. Tourigny

Project Manager, U.S.N.R.C.

Washington, D. C.

Qualification and Experience

Experience

U.S. Nuclear Regulatory Commission; Washington, D.C., Operating Reactors Project Manager; (October 1981 - Present)

- Just under seven years operating nuclear power plant experience including:
  - managing hundreds of safety and environmental reviews associated with license amendments, NRC rule exemptions, and ASME code reliefs;
  - performing independent technical evaluations in support of license amendments, NRC rule exemptions, and ASME code reliefs;
  - conducting extensive plant system walkdowns comparing technical specifications, implementing procedures, PXID's, and FSAR descriptions;
  - evaluating plant operational events/licensee response/licensee resolution;
  - evaluating plant changes and modifications made under 10 CFR 50.59;
- Extensive knowledge of NRC power plant regulations, regulatory guides, and standard review plan;
- Able to perform evaluations associated with accident analysis, fuel reloads, inservice inspection, fire protection, equipment qualification, and missile protection;
- Received extensive NRC training in Westinghouse, Combustion Engineering, Babcock and Wilcox, and GE NSSS; and Balance of Plant;
- Trained on Sequoyah Simulator (W), Calvert Cliffs Simulator (C.E.), and Bellefonte Simulator (BXW);
- Project Manager assignments included St. Lucie, Unit Nos. 1 X 2, Forth Calhoun, ANO - 2, and Trojan;

U.S. Nuclear Regulatory Commission; Washington, D.C., Radioactive Waste Management Program Manager; (April 1979 - October 1981)

- Two and one half years low level waste management experience including:
  - development and maintenance of multi-year program plans which contained goals, objectives, planned accomplishments, and resource requirements;
  - coordination of technical work performed inside NRC and outside NRC by other agencies and contractors;
  - budget preparation and execution;
- Interfaced with outside groups such as state legislators, public advisory groups, etc.;
- Evaluated proposed Congressional bills and prepared Congressional testimony;
- Interfaced with high level waste program managers.

U.S. Nuclear Regulatory Commission; Washington, D.C., Nuclear Fuel Cycle and Materials Program Analyst; (April 1977 - April 1979)

- Major author of Congressionally requested study entitled "Regulation of Federal Radioactive Waste Activities", (NUREG-0527). Was responsible for categorizing all federal nuclear waste by type and location. Has in-depth knowlege of DOE sites which process, store, and dispose nuclear waste.
- Major author of Commission requested study entitled "The License Renewal Study for Parts 30, 40, and 70 Licenses." Study was the first in-depth analysis of why the NRC issues Parts 30, 40, and 70 licenses for 5 years.
- Extensive knowledge of NRC fuel cycle and material regulations.

U.S. Nuclear Regulatory Commission, Washington, D.C., Special Assistant; (January 1976 - April 1977)

- Major author of first Commission requested NRC Five Year Plan.

The Pennsylvania State University, University Park, PA, Graduate Assistant; (January 1973 - December 1975)

- Served as a Graduate Assistant to the Director of the Nuclear Reactor Facility including:
  - development and utilization of nuclear design computer codes for in-core analysis of commercial LWR's;
  - assisting in the instruction of students in the use of computer codes for core design and fuel management.
- Graduate student in Industrial Engineering.
- Awarded full tuition and stipend.

Westinghouse Electric Corporation, Pittsburgh, PA, Marketing Engineer; (October 1971 - December 1972)

- Performed technical and economic analysis of nuclear fuel cycles.
- Provided support to nuclear fuel and nuclear plant sales managers.

Nuclear Materials and Equipment Cororation, Apollo, PA, Nuclear Engineer; March 1969 - October 1971)

- Two and one half years of power reactor design experience including:
  - proposal preparation to market commercial nuclear fuels;
  - reactor core physics analysis;
  - fuel management studies;
  - nuclear computer code benchmarking studies.

The Pennsylvania State University, University Park, PA, AEC Trainee in Reactor Operations; (June 1968 - March 1969)

- Completed reactor operator training program.
- Graduate student in Nuclear Engineering.
- Awarded full tuition and stipend.

Formal Education

MASTER OF ENGINEERING - INDUSTRIAL

The Pennsylvania State University, Industrial Engineering Department, 1976. Thesis: The Reliability With and Without Repair and Availability of Four Element Systems.

MASTER OF ENGINEERING - NUCLEAR

The Pennsylvania State University, Nuclear Engineering Department, 1970. Thesis: The Neutronics Calculation of a Pressurized Water Reactor.

BACHELOR OF SCIENCE - NUCLEAR

Lowell Technology Institute, Nuclear Engineering Department, 1968.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-335-OLA
FLORIDA POWER AND LIGHT COMPANY	)	
	)	(SFP Expansion)
(St. Lucie Plant, Unit No. 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "RESPONSE OF NRC STAFF IN SUPPORT OF LICENSEES MOTION FOR SUMMARY DISPOSITION" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or as indicated by an asterisk through deposit in the Nuclear Regulatory Commission's internal mail system, this 30th day of August, 1988:

B. Paul Cotter, Jr., Chairman*
Administrative Judge
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Richard F. Cole*
Administrative Judge
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing
Board Panel (1)*
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing
Appeal Panel (5)*
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Glenn O. Bright*
Administrative Judge
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Washington, D.C. 20555

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Benjamin H. Vogler
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