August 5, 1988

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 & LIGHT COMPANY	(ASLBP No. 88-560-01-LA)
(St. Lucie Plant, Unit No. 1)	

LICENSEE'S MOTION FOR SUMMARY DISPOSITION OF INTERVENOR'S CONTENTIONS

Pursuant to 10 C.F.R. § 2.749, Florida Power & Light Company ("FPL" or "Licensee") hereby moves for summary disposition of each of Intervenor's contentions in the above-captioned proceeding. For the reasons set forth herein, it is Licensee's position that there is no genuine issue of fact material to any of Intervenor's contentions, and that FPL is entitled to a decision in its favor on each of the contentions as a matter of law. This motion is supported by "Licensee's Statement of Material Facts as to Which There Is No Genuine Issue To Be Heard with respect to Intervenor's Contentions" ("Licensee's Statement"); "Licensee's Memorandum of Law in Support of Licensee's Motion for Summary Disposition of Intervenor's Contentions" ("Licensee's Memorandum"); and the following affidavits:

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for buckling of the liner plate, as well as stresses in welds and embedded metal associated with the liner system. The analysis showed that there would be no loss of function under all postulated conditions. Weber Affidavit 3, ¶ 18.

The conclusion can therefore be made that the St. Lucie 1 reinforced concrete spent fuel pool structure will withstand the thermal loads expected as a result of the spent fuel pool expansion since the start of spent fuel storage until the expiration of the operating license on March 1, 2016. The expected service temperatures are well below the level that may cause deterioration of the concrete, and the strength and serviceability of the structural concrete will not be reduced. Weber Affidavit 3, ¶ 22.

There are no anticipated effects from alpha and beta radiation on the stainless steels used in the spent fuel pool racks and fuel pool liner. Fast neutrons can only affect these stainless steels when the integrated fast fluence reaches levels of 10^{17} to 10^{18} neutrons/cm^2 or higher. Only at this threshold level can measurable changes in the properties be detected. As indicated earlier, the maximum integrated fluence for the St. Lucie 1 spent fuel pool through the expiration of the operating license is at least an order of magnitude below this threshold level. Hence, it is readily evident that the fast neutron exposure will have essentially no effect on the St. Lucie 1 spent license (March 1, 2016). The radiation levels and heat loads are well below those that may cause deterioration of the concrete, and the strength and serviceability of the structural concrete will not be reduced. Weber Affidavit 3, ¶¶ 21 & 22.

With respect to the strength of the pool under the physical forces associated with thermal stress, the load-carrying capacity of the pool structure was evaluated by conducting a detailed computer analysis as part of the overall evaluation of the St. Lucie 1 spent fuel pool for increased capacity. All loads potentially imposed on the structure were considered, including the effect of heat from the pool water and the concrete heating from gamma and neutron energy deposition from the spent fuel; static loads resulting from the total dead weight; dynamic loads associated with postulated earthquakes and cask drop accidents; and the most severe thermal load on the structure, caused by temperature gradients which would occur due to boiling of the pool water when the prevailing ambient temperature outside the pool was conservatively assumed to be 40°F. Weber Affidavit 3, ¶ 17. The results of the analysis show that the pool will maintain its structural integrity even under the severe conditions assumed. Id., ¶ 18.

The liner plate was conservatively not considered to provide structural capability in the structural analysis in the pool. However, a separate analysis was performed to determine the effects of thermal and other loads on the functionality of the liner plate system. This analysis considered the potential experience no significant damage to any attribute when exposed to a neutron integrated fluence of 3×10^{20} neutrons/cm^2. Id., ¶ 10. Since the total gamma dose and neutron radiation exposure experienced by the spent fuel pool structural concrete as a result of reracking through the expiration of the St. Lucie 1 operating license is much lower than the total exposure that may cause radiation damage to the concrete, there will not be any appreciable deterioration of the structural concrete due to radiation. Id., ¶ 11.

During normal refueling discharge and full core discharge to the spent fuel pool, the temperature of the concrete will not exceed the appropriate limits established in the ASME Code, Section III, Division 2, Concrete Reactor Vessels and Containments, and the ACI 349, Code Requirements for Nuclear Safety Related Concrete Structure, Appendix A. See Weber Affidavit 3, ¶¶ 12-16. Since the structural concrete temperatures during both normal spent fuel discharge and full core discharge do not exceed the code requirements, the strength and serviceability capabilities of the St. Lucie 1 spent fuel pool structural concrete will be retained and the concrete temperatures will not cause deterioration of the concrete. Weber Affidavit 3, ¶ 20.

The conclusion can therefore be made that the St. Lucie 1 reinforced concrete spent fuel pool structure will withstand the radiation levels and heat loads expected as a result of the spent fuel pool expansion through the expiration of the operating

exposed to radiation dose levels of less than 10^{11} rads through the expiration of the St. Lucie 1 operating license. The steel liner and concrete structure of the pool will be exposed to lower radiation levels because of the shielding effect of the water. Turner Affidavit, ¶ 39. Calculations show that the maximum gamma radiation total dose that could be experienced by any portion of the structural concrete will be less than 3×10^{10} rads. Weber Affidavit 3, ¶ 9.

Gamma and neutron radiation are the only types of radiation that need to be considered when evaluating radiation effects on structural concrete. Alpha and beta radiation are of no consequence since they are absorbed before they can reach the spent fuel pool structural concrete. Weber Affidavit 3, ¶ 7. As stated above, the maximum gamma radiation total dose that could be experienced by any portion of the structural concrete through the expiration of the St. Lucie 1 operating license (March 1, 2016) is less than 3×10^{10} rads. This could occur only under an assumption that the most recently discharged fuel assemblies would be stored in the same Region 1 location closest to the walls every refueling outage and left there until the next refueling outage. Id., ¶ 8-9. Experiments have indicated that concrete will experience no significant damage to any attribute when exposed to a total gamma dose of up to 3×10^{11} rads. Id., ¶ 9. The maximum neutron integrated fluence that would be experienced by any portion of the structural concrete is 9.8×10^{14} neutrons/cm^2. Experiments have indicated that concrete will temperature will also decrease with time as the temperature of the water in the pool falls. The maximum temperature of the concrete with gamma and neutron (nuclear) heating will be less than 160°F and will decrease with time as the temperature of the water in the pool decreases. Weber Affidavit 3, ¶¶ 12-13. The spent fuel pool liner and rack materials will also be exposed to heat due to the temperature of the pool water. Portions of the racks could also be subject to slightly higher temperatures (a few degrees at most) due to localized gamma heating. Kilp Affidavit 3a, ¶ 10.

The temperatures in the spent fuel pool are far less severe than temperatures experienced in a reactor. Commercial reactors generally operate at coolant outlet temperatures in the 500-640°F range, which is to be compared to the short duration maximum temperature associated with pool boiling, which might occur for a loss of forced cooling event. The fuel assemblies and fuel cladding were designed and utilize materials to withstand the temperatures and heat loads present in the reactor. Kilp Affidavit 3b, ¶ 8.

Neutron flux levels in the spent fuel pool are low, reaching a very conservatively estimated maximum integrated fluence of less than 5 x 10^{15} neutrons/cm^2 through the expiration of the St. Lucie 1 license. Fuel assembly structural materials and fuel cladding will receive an estimated dose of less than 5 x 10^{10} rads while in the spent fuel pool through the expiration of the St. Lucie 1 operating license. Boraflex is expected to be physical characteristics of Boraflex allow fabrication and handling in continuous pieces. Singh Affidavit 3/6, ¶ 13. Boraflex has been the preferred material for neutron absorption in spent fuel racks within the United States, as evidenced by the fact that over 85% of all storage racks ordered by U.S. utilities since 1980 have incorporated Boraflex. Id., ¶ 31.

Shortly after completion of a normal refueling discharge into the spent fuel pool, the temperature of the water in the pool could rise to a maximum temperature of 133.3°F. After storage in the pool for approximately 8 days, the temperature of the water in the pool will then decrease to approximately 128°F. Shortly after completion of a full-core discharge into the spent fuel pool, the temperature of the water in the pool could rise to a maximum temperature of 150.8°F. After storage in the pool for approximately 9 days, the temperature of the water in the pool will decrease to approximately 141°F. Weber Affidavit 3, ¶¶ 12-14. Under loss of forced (pumped) cooling conditions, temperatures could reach boiling. Kilp Affidavit 3a, ¶ 10.

Considering only heating due to the temperature of water in the pool, the maximum temperature of the structural concrete at the interior surface in contact with the liner under the normal refueling discharge will be approximately 133°F, and it will decrease with time as the temperature in the pool falls in temperature. Likewise, the maximum temperature of the structural concrete at the interior surface in contact with the liner under full-core discharge will be approximately 150°F, and that and fuel cladding were designed and utilize materials to withstand the temperature and heat loads present in the reactor. These are far more severe than those present in the spent fuel pool. Id., ¶ 8.

The spent fuel storage racks (excluding the Boraflex) are made from Type 304L stainless steel, except for the feet which are made from Type 17-4 Ph (or 17:4 Ph) stainless steel. The minimum thickness of the racks at any point is 0.08 inches. Type 304L stainless steel differs from Type 304 only in that the former has a slightly lower carbon content. The Type 17-4 Ph stainless steel has comparable corrosion properties to Types 304 and 304L. It is used for the feet on the racks because it can be heat-treated to a designated hardness, which makes it useful for threaded connections with Type 304L. The harder Type 17-4 Ph prevents galling (adhesion of contacting surfaces) when threaded into Type 304L. Type 304, Type 304L and Type 17-4 Ph stainless steels are also used within the primary system of pressurized water reactors. Kilp Affidavit 3a, ¶ 6; Singh Affidavit 3/6, ¶ 11.

The Boraflex in the spent fuel storage racks serves a neutron attenuation function. Singh Affidavit 3/6, ¶ 12. Boraflex is composed of a polymeric silicone encapsulant entraining and fixing fine particles of boron carbide in a homogeneous, stable matrix. The inherent stability of both silicones and carbides results in compatibility with a variety of chemical environments concurrent with strong ionizing radiation. The experimental facilities and nuclear fuel fabrication plants. Concrete possesses many of the physical attributes of an ideal shielding material. Concrete can also be readily designed to resist temperature gradients and resulting thermal stresses through the structural elements by the addition of steel reinforcement. Weber Affidavit 3, ¶ 5. The use of the concrete as a radiation shield is recognized by industry and the NRC. Id., ¶ 6.

The St. Lucie 1 spent fuel pool liner is made from Type 304 stainless steel, which is a preferred material for use in applications such as spent fuel pool liners because of, among other things, its exceptional corrosion resistance to high temperature water. Stainless steel has a demonstrated ability to perform in nuclear power plant applications which are more severe than those in the spent fuel pool. Kilp Affidavit 3a, ¶¶ 4 & 5.

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

Memorandum and Order, p. 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. Id. at 19.

The St. Lucie 1 spent fuel pool is 37 feet long by 33 feet wide and 40 feet 6 inches deep, with reinforced concrete walls 6 feet thick, and a reinforced concrete floor and foundation mat 9 feet 6 inches thick. The fuel cask storage area is located in the northeast corner of the spent fuel pool and is 10 feet long by 12 feet wide with a floor which is a depression in the mat that is 3 feet 6 inches deep. The spent fuel pool floor and bottom of the walls are lined with 1/4 inch thick stainless steel and the remainder of the walls, except for the cask storage area, is lined with 3/16 inch stainless steel. The depression part of the fuel cask storage area in the mat is lined with 1/2 inch thick stainless steel plate on the walls. The floor of the cask storage area is lined with 1 inch thick stainless steel plate. The fuel cask storage area is enclosed on the south and west sides with built up steel walls lined with 1/4 inch stainless steel plate. The walls are 6 7/8 inches thick and 14 feet 9 inches high. Weber Affidavit 3, ¶ 4.

The St. Lucie 1 spent fuel pool walls and floor structural elements are reinforced concrete. Reinforced concrete is especially suited to resist the effects of radiation and heat in the spent fuel pool and is the most widely used material for shielding in nuclear power plants, as well as other nuclear facilities, such as hot laboratories, radiochemical plants, 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 a 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."

would not, 1/ and no fuel assemblies would be uncovered. Weber Affidavit 1, ¶ 12. The cask drop scenario considered in the structural analysis is the bounding scenario for cask drop accidents. Id., ¶ 13.

In sum, Licensee performed analyses of postulated cask drop accidents for damaged fuel. The analyses were conservatively performed. Each showed that the radiation releases would meet NRC guidance and criteria, being well within the dose guidelines specified in 10 C.F.R. Part 100. Marschke Affidavit, ¶ 19. Further, the probabilities and uncertainties identified in the BNL Report with respect to a severe spent fuel accident are not of concern, since even a conservatively assumed worst case cask drop accident would not cause structural failure and draining of the pool. Weber Affidavit 1, ¶ 15.

B. 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).

1/ To the extent Admitted Contention 1 expresses concern over a criticality accident in the spent fuel pool in case of pool draining, such an accident is impossible. Criticality cannot occur since the water in the pool is the only moderator available. Without any moderator, criticality is impossible. Weber Affidavit 1, ¶ 6.

¶ 5. Probability estimates of spent fuel pool structural failure resulting from a cask drop accident were considered in the BNL Report. Id., ¶ 7.

At the outset, it should be noted that the St. Lucie 1 spent fuel pool has a spent fuel cask handling and storage area located in the northeast corner of the pool. This area is separated from the remainder of the pool by walls higher than the top of the stored fuel. Thus, it is of a configuration which the BNL Report recognizes as presenting zero risk to the main pool from a cask drop. See BNL Report, p. 37, n. a; Weber Affidavit 1, ¶ 8.

Nevertheless, the results of a cask drop on the structural integrity of the St. Lucie 1 pool were analyzed. A scenario was considered such that the cask would be dropped from the maximum height available for a crane lift, a total 60.46 feet, and strike perpendicular to the pool bottom mat -- at its thinnest location -- for maximum impact energy. Moreover, the cask was conservatively assumed to be a rigid body absorbing no impact energy. Weber Affidavit 1, ¶ 10. The scenario also assumed the load combination giving the maximum stresses. Id., ¶ 11.

All safety factors for this load combination were greater than one, so that the pool could not fail from a cask drop. At most, hairline cracking might occur, but pool draining

2, it was assumed that one-third of the core was placed in the spent fuel pool during refuelings and that, then, the entire core was removed to the spent fuel pool, filling it. Marschke Affidavit, ¶¶ 13 & 17. In both cases, a non-mechanistic release of all of the radioactive noble gases and iodine stored in the gas gap of the fuel rods from all of the stored fuel in the pool was assumed. Id., ¶ 13.

In this connection, it is relevant to note that, in the case of any real cask drop, only a small percentage of fuel assemblies could actually be damaged. This is due to the fact that cask movement is limited to the spent fuel storage area in the northeast corner of the pool, and is physically prevented from being directly over any stored fuel. Marschke Affidavit, ¶ 15.

Under the postulated accident conditions, criticality would not occur. Turner Affidavit, ¶ 34. Further, exclusion area boundary doses were determined to be less than 0.1 rem whole body and 30 rem to the thyroid and, thus, "well within" the 10 C.F.R. Part 100 exposure guidelines values for both Case 1 and Case 2. Marschke Affidavit, ¶¶ 16, second paragraph numbered 17 & Table 1. Thus, regulatory requirements and guidance are met. Marschke Affidavit, ¶¶ 9, 10, & 19.

The BNL Report attempted to assess the likelihood and consequences of a severe accident in a spent fuel storage pool involving the complete draining of the pool. Weber Affidavit 1, 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 Water Reactors" (1972), is identified in the SRP, section 15.7.5, as the source of appropriate conservative assumptions to be utilized in calculating potential radiological consequences from a postulated cask drop accident. Appropriate assumptions from Regulatory Guide 1.25 were utilized in the St. Lucie 1 radiological consequences analysis. Marschke Affidavit, ¶ 11.

The radiological impact of a postulated accident involving a cask's dropping into a spent fuel storage pool and damaging fuel assemblies depends upon the amount of fission products of concern released from the fuel assemblies in storage. In evaluating the consequences of a cask drop accident, the release fractions specified in NRC Regulatory Guide 1.25 were utilized, along with the assumption that all fuel assemblies in the completely filled spent fuel pool were damaged. Marschke Affidavit, ¶ 12.

Two cases, using different sets of conservative assumptions, were considered for the St. Lucie 1 cask drop accident analysis. Each assumed that all fuel assemblies were damaged, an extremely conservative assumption. Marschke Affidavit, ¶¶ 13, 15 & 17. For Case 1, it was assumed that one-third of the core was placed in the spent fuel pool during each refueling until the pool was filled. Id., ¶¶ 13 & 15. For Case

Affidavit, ¶ 5. Further, any radiological releases resulting from damage to the fuel would be well within acceptable limits. Id., ¶ 19.

Part 100 provides that, in the case of a postulated accident: (a) an individual located at any point on the exclusion area boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure; and (b) an individual located at any point on the outer boundary of the low population area who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose of 300 rem to the thyroid from iodine exposure. See Marschke Affidavit, ¶ 6. For a cask drop accident at St. Lucie 1 the dose at the exclusion area boundary is controlling. Marschke Affidavit, ¶ 8.

Under paragraph 15.7.5.II.1 of the NRC Standard Review Plan (NUREG-0800) ("SRP"), the potential radiological consequences of a postulated cask drop accident should be "well within" the exposure guideline values of 10 C.F.R. Part 100. Paragraph 15.7.5.II.1 defines "well within" to mean 25 percent or less of the 10 C.F.R. Part 100 exposure guideline values, i.e., 75 rem for the thyroid and 6 rem for the whole-body doses. Marschke Affidavit, ¶ 9.

compliance with 10 CFR Part 100 (1987).

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

In essence, the contention questions whether the potential radiological consequences of a cask drop accident have been conservatively calculated and will meet the criteria of 10 C.F.R. Part 100 ("Part 100"). Intervenor cites as the bases of his contention uncertainties identified in the Brookhaven National Laboratory report entitled: "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," NUREG/CR-4982, BNL-NUREG-52093 (July 1987) ("BNL Report").

Intervenor's basic concern -- based on his understanding of the BNL Report -- is that a cask drop accident will violate the integrity of the pool, resulting in its complete and essentially instantaneous draining and possible criticality. Draining of the pool, however, can only result from structural failure of the pool. As discussed below, the pool would not fail structurally in the case of a cask drop. Further, even if the pool were somehow to drain, criticality could not occur since there would be no water in the pool to act as a neutron moderator. Without any moderator, criticality is impossible. Weber Affidavit 1, ¶ 6.

Since there would be no structural failure, the water would remain in the pool following a cask drop accident. However, criticality still would not occur. This is because of both the nature of the damage to the fuel which would result, and the presence of neutron absorbing poisons in the pool. Marschke

II. Discussion of Contentions and Argument

A. 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 with 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 Petition, 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

Intervenor as a party to this proceeding and admitting Intervenor's Contentions 3, 4, 6, 8, 9, 11, and 15, renumbered, respectively, as Admitted Contentions 1 through 7.

The Board deferred ruling on Intervenor's Contention 5. On May 31, 1988, the Board dismissed Contention 5 due to Intervenor's failure to advise the Board, pursuant to its April 20 Memorandum and Order, that he wished to pursue the contention.

On June 24, 1988, Intervenor requested that he be permitted to withdraw Admitted Contention 2, stating:

> It has come to my attention that the temporary crane that was installed in the spent fuel pool storage area to facilitate the reracking procedure has been removed from the storage area. In light of this, I would ask that the Board withdraw from contention, admitted Contention 2, which concerns the damages that might have existed from the presence of the crane. Obviously, these concerns are no longer sensible.

Upon consideration of Intervenor's request, the Licensing Board issued an Order on July 27, 1988, dismissing Admitted Contention 2 "with prejudice as moot."

On June 2, 1988 FPL served "Licensees First Set of Interrogatories to Intervenor." The "Intervenors Response to Licensee's Interrogatories" ("Intervenor's Response") was submitted on July 6, 1988. There are no outstanding discovery requests. Intervenor's contentions are ripe for summary disposition.

submitted Revision 1 to the SAR ("SAR, Rev. 1"). The purpose of SAR, Rev. 1 was to incorporate into the SAR modifications resulting from exchanges between FPL and the NRC Staff during the latter's review of the amendment request.

A notice of FPL's request for a license amendment was published in the Federal Register at 52 Fed. Reg. 32,852 (1987). On September 30, 1987 Mr. Campbell Rich ("Intervenor") addressed a letter to the Secretary of the Nuclear Regulatory Commission, asking that a public hearing be held concerning the requested spent fuel pool expansion amendment. In responsive pleadings filed November 4 and 9, 1987, both the NRC Staff and FPL pointed out that the letter failed to meet the requirements of 10 C.F.R. § 2.714 and that, therefore, the request should be denied. Thereafter, pursuant to a Licensing Board Memorandum and Order dated November 13, 1987, Intervenor submitted a "Request for Hearing and Petition for Leave to Intervene" ("Amended Petition"). The Amended Petition contained 16 contentions which Intervenor proposed be admitted in this proceeding.

Following completion of its review, the NRC Staff determined that the requested amendment involved no significant hazards consideration, and issued Amendment 91 to Facility Operating License No. DPR-67 on March 11, 1988, accompanied by a Safety Evaluation ("SE"). Thereafter, following a prehearing conference on March 29, 1988, the Licensing Board issued a Memorandum and Order, dated April 20, 1988, accepting the

1. Affidavit of Stephen Marschke on Admitted Contention 1 (July 25, 1988) ("Marschke Affidavit");

2. Affidavit of Murray Weber on Admitted Contention 1 (July 26, 1988) ("Weber Affidavit 1");

3. Affidavit of Dr. Gerald R. Kilp on Admitted Contention No. 3 ("Kilp Affidavit 3a") (July 22, 1988);

4. Affidavit of Dr. Gerald R. Kilp on Admitted Contention 3 ("Kilp Affidavit 3b") (July 22, 1988);

5. Affidavit of Murray Weber on Admitted Contention 3 (July 27, 1988) ("Weber Affidavit 3");

6. Affidavit of Dr. K. P. Singh on Admitted Contentions 3 and 6 ("Singh Affidavit 3/6") (July 28, 1988);

7. Affidavit of Dr. K. P. Singh on Admitted Contentions 4 & 5 (July 28, 1988) ("Singh Affidavit 4/5");

8. Affidavit of John B. Houghtaling on Admitted Contentions 4 and 5 (July 28, 1988) ("Houghtaling Affidavit 4/5"); and

9. Affidavit of Dr. Stanley E. Turner on Admitted Contentions 6 and 7 ("Turner Affidavit").

I. Background of This Proceeding

By means of a letter dated June 12, 1987, from C.O. Woody to the Nuclear Regulatory Commission ("NRC"), FPL submitted a request to amend the Operating License for St. Lucie Plant, Unit No. 1 ("St. Lucie 1") to modify the existing spent fuel storage facility for the unit in order to increase its storage capacity. In support of this request, FPL submitted a "Spent Fuel Storage Facility Modification Safety Analysis Report" ("SAR"). Under cover of a letter dated January 29, 1988, FPL

fuel pool liner and stainless steel rack materials. Kilp Affidavit 3a, ¶ 11. Gamma radiation has no physical effects on the stainless steel materials used in the St. Lucie 1 pool liner and racks. Kilp Affidavit 3a, ¶ 12.

The stainless steels used in the St. Lucie 1 pool liner and storage racks are virtually immune to corrosion at spent fuel pool temperatures. No significant galvanic attack is expected. Kilp Affidavit 3a, ¶ 14. The possibility of stress-corrosion cracking of sensitized Type 304 stainless steel adjacent to welds in the pool liner is remote and would only be minor, should it occur. Id., ¶ 16. Type 304L stainless steel has virtually no possibility of stress-corrosion cracking. Id., ¶ 17. Hydriding is of no concern for the stainless steels under the duties of interest for the pool liner and racks. Id., ¶ 15.

In sum, there are no materials degradation concerns for the stainless steels used in the St. Lucie 1 spent fuel pool liner and storage racks. Kilp Affidavit 3a, ¶ 20. Support for this conclusion is drawn from the extensive experience accumulated all over the world in spent fuel pools similar to the St. Lucie 1 spent fuel pool. Id., ¶ 19.

It is evident that any radiation effects on fuel cladding and fuel assembly material attributable to increasing the capacity of the spent fuel pool are negligible when compared to prior reactor exposure. The materials used in the fuel cladding and fuel assemblies are virtually immune to changes due to alpha, beta and gamma radiation, and the fast neutron levels in the spent fuel pool are so small as to be inconsequential. The added neutron exposure due to storage in the spent fuel pool would be equivalent to less than two minutes in the reactor at full power. Hence, there is no threat to the integrity of these materials caused by the reracking. Kilp Affidavit 3b, ¶¶ 6-7.

The corrosion rate of Zircaloy in water or steam is approximately 1/100,000 inch per year at 500°F and results in an amount of corrosion to the Zircaloy that is well under 10% of the thicknesses of the fuel cladding remaining after discharge of the spent fuel from the reactor. The corrosion rate is at least an order of magnitude lower at the much lower temperatures predicted for the spent fuel pool. Kilp Affidavit 3b, ¶ 9. Additionally, the stainless steel materials are virtually immune to corrosion at spent fuel pool temperatures. The corrosion rate for Type 304 stainless steels has been shown not to exceed 6/10,000 inches after 100 years in an oxygenated borated water environment similar to that in the St. Lucie 1 spent fuel pool. Corrosion rates for the Inconels are at least as low as those for stainless steels. Id., ¶ 11.

The hydriding rate of Zircaloy in the St. Lucie 1 spent fuel pool is virtually nil and Zircaloy is considered virtually immune to stress-corrosion cracking. Kilp Affidavit 3b, ¶ 10. For the stainless steels, Zircaloy and Inconel, no significant galvanic attack is expected. Id., ¶ 11. From numerous studies of the performance of Type 304 stainless steel in fuel storage pools throughout the world, there is no evidence that stress-

corrosion cracking of stainless or the other assembly components is occurring to any significant degree. Even if localized stress-corrosion cracking were assumed to occur, it would not affect the fuel rods. Id., ¶ 12.

Support for the above conclusions is drawn from the extensive experience accumulated all over the world in safely storing nuclear fuel in pools similar to the St. Lucie 1 spent fuel pool. Kilp Affidavit 3b, ¶ 14. It can be concluded, therefore, that the fuel assemblies can be stored safely in the St. Lucie 1 spent fuel pool from the present to the expiration of the St. Lucie 1 operating license (March 1, 2016). Id., ¶ 15.

Substantial information pertaining to the performance characteristics of Boraflex has been developed from testing. The results of this testing demonstrate the suitability of Boraflex for use as a neutron absorber in the spent fuel pool environment. Singh Affidavit 3/6, ¶ 15.

Under testing, Boraflex exhibits excellent heat aging characteristics. Testing has also been performed confirming the stability of the material in various chemical environments, including high-temperature, borated water. Id., ¶ 16. Neutron absorption of Boraflex was measured at various boron^{10} (or B^{10} or boron-10) loadings to confirm the absorptive characteristics of the material. Id., ¶ 17. Radiation exposure tests of Boraflex at total equivalent doses of 10^{12} rads were performed during 1979-1981. This test program was designed to identify the physical and chemical characteristics of Boraflex under a variety of radiation levels, radiation rates, and environments. All evidence from the tests suggests that, at the exposure levels expected, Boraflex maintains sufficient bend tolerance to withstand normal and anticipated conditions of service in storage rack applications. Evaluation of the data reveals no discernible effects of either environment or irradiation on neutron absorption. The tests resulted in no significant leaching of either boron or halogens. Gas generation due to irradiation has been evaluated and found to be negligible. Id., ¶¶ 18 & 19.

Most recently, Boraflex has been tested under carefully controlled conditions to determine its dimensional changes in a precise manner under irradiation. Analysis indicates the total in-plane shrinkage to be less than 2.5%. This shrinkage was anticipated and accounted for in the St. Lucie 1 rack design. Singh Affidavit 3/6, ¶ 20.

As discussed above, the maximum cumulative radiation dose to the Boraflex material in the St. Lucie 1 pool for all fuel discharges until the end of the licensed life of the unit is not expected to exceed 10^{11} rads. This is merely 10% of the 10^{12} rad equivalent radiation dose given to this material in the laboratory tests discussed above. Singh Affidavit 3/6, ¶ 21.

The rack technology employed for producing the St. Lucie 1 racks utilizes a proven and widely used technology and reflects established industry practice. Singh Affidavit 3/6, ¶ 22. Consistent with practice in the industry, the racks used in the St. Lucie 1 spent fuel pool are of two basic types, commonly referred to as Region 1 and Region 2 types. Id., ¶ 23. Region 1 modules are of the so-called "flux-trap" construction, which is an industry standard for modules of this type. Id., ¶ 24. Each Region 1 fuel assembly storage box is equipped with a sheet of Boraflex on each of its four sides. The Boraflex panel is positioned in its place by a stainless steel sheathing which also serves to protect the Boraflex material from accidental dents. The stainless steel box surrounded by Boraflex on four sides is also a universally employed technology. Id., ¶ 25.

Region 2 rack modules are designed to store fuel with a specified minimum burnup and are constructed from the same basic elements as the Region 1 racks. The flux-trap gap (water gap) is not required for Region 2 modules and, therefore, not provided. The Boraflex panels are positioned in place by the contiguous walls of the boxes and suitably located peripheral strips. Id., ¶ 28. All of the above mentioned features of rack construction are routinely used in the industry, and represent total conformance with the industry norm. Id., ¶ 29.

For the St. Lucie 1 storage racks, established industry practice in rack construction was followed in the installation of Boraflex, with one difference based in experience. Previous manufacturing practice called for the use of a silicone based adhesive to cement the material in its place. For example, this practice was followed for the Quad Cities spent fuel racks, which experienced gaps in the Boraflex panels. The experience gained with Boraflex in operating plants indicated that the twin effects

of irradiation, namely shrinkage and hardening, are best accommodated by providing complete freedom for the Boraflex panels to undergo in-plane dimensional changes, and by installing Boraflex panels large enough to allow the neutron absorption function to be carried out even after shrinkage. Accordingly, the cementing glue was eliminated for the installation of the Boraflex panels in the St. Lucie 1 spent fuel racks, and provision was made in the manufacturing process to install the Boraflex panels in-place, with minimal surface loading. Singh Affidavit 3/6, ¶ 30. The St. Lucie 1 racks are designed to provide for in-plane dimensional changes to Boraflex. Id., ¶ 35.

It is therefore concluded that the design and fabrication of the St. Lucie 1 racks incorporate proven technology for Boraflex installation and positioning. The underlying causes leading up to the Quad Cities Boraflex problems, as described in NRC Information Notice No. 87-43, have been eliminated. The rack modules at St. Lucie 1 are based on refinement of established technology in light of operating experience. All aspects of their construction are based upon proven design ideas and well-established fabrication techniques. Singh Affidavit 3/6, ¶ 36. In sum, the results of the Boraflex testing program and the design and fabrication of the St. Lucie 1 racks, incorporating experience and proven technology for Boraflex installation, have demonstrated the suitability of using Boraflex as a neutron absorber for the storage of spent fuel in the St. Lucie 1 spent fuel pool. Id., ¶¶ 15 & 36.

In sum, the St. Lucie 1 reinforced concrete spent fuel pool structure will withstand the radiation and heat levels expected as a result of the spent fuel pool expansion through the expiration of the St. Lucie 1 operating license (March 1, 2016). It will also withstand the expected thermal loads. There are no materials degradation concerns for the stainless steels in the St. Lucie 1 spent fuel pool liner and storage racks. Any radiation and heat effects on the fuel cladding and fuel assembly materials attributable to increasing the capacity of the spent fuel pool are negligible when compared to prior reactor exposure. The suitability of using Boraflex as a neutron absorber for the storage of spent fuel in the St. Lucie 1 spent fuel pool has been demonstrated.

C. 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 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, 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 decay of fission products in spent fuel assemblies produces heat. Singh Affidavit 4/5, ¶ 9. The amount of decay heat generated by the spent fuel assemblies diminishes very rapidly with time after they are taken out of the reactor. Singh Affidavit 4/5, ¶ 10. In fact, two weeks after discharge to the spent fuel pool, a single assembly will have decreased its heat generation to approximately 15%, after one month to approximately 10%, and after one year to approximately 2%, of its original value. Singh Affidavit 4/5, Figure 1. Therefore, because of this rapid reduction in heat generation over time, a batch of 80 assemblies discharged and stored in the fuel pool for two years will generate less heat than a single assembly one hour after discharge. Singh Affidavit 4/5, ¶ 11.

When the fuel assemblies are first discharged to the spent fuel pool, their heat generation rate is at its peak. Singh Affidavit 4/5, ¶ 14. Thus, at the beginning of storage the heat generation rate exceeds the heat removal rate. Id. This initial surplus of heat generated over the heat removed results in a gradual rise of the pool water temperature. Id. At the same time, the spent fuel pool cooling system heat removal rate increases. Id. After a relatively short time, the cross over point between the heat generation rate and heat removal rate is reached, and it is this point which marks the maximum pool temperature. Id. See also Singh Affidavit 4/5, Figure 3.

The spent fuel pool cooling system, itself, is a closed loop consisting of two full capacity pumps, one heat exchanger and associated valves, piping and instrumentation. This system is designed to transfer decay heat from spent fuel in the spent fuel pool to the component cooling water ("CCW") system. The spent fuel pool cooling system draws water from the spent fuel pool near the surface and returns it to the bottom on the

opposite side of the spent fuel pool after passing the water through the heat exchanger to remove decay heat. A completely separate loop with its own pump, filters, demineralizer, piping and valves is used to purify the water and maintain spent fuel pool clarity. Houghtaling Affidavit 4/5, ¶ 5; Singh Affidavit 4/5, ¶¶ 12 & 13 & Figure 2.

The spent fuel pool heat exchanger is of a horizontal shell and tube design with a two-pass tube side. The spent fuel pool water circulates through the tube side and CCW circulates through the shell side. The heat exchanger is constructed of a carbon steel shell and stainless steel tubes, and has been fabricated to Section III Class C requirements. The spent fuel pool pumps are of the horizontal centrifugal type with mechanical seals. Each pump is capable of pumping 1500 gpm at a 70 foot head. The pumps are constructed of cast stainless steel. Each pump is driven by a 40 HP, 3-Phase, 460 Volt AC motor. All piping in the spent fuel pool cooling system is constructed of seamless stainless steel with welded joints, except for the pump connectors which are flanged. Houghtaling Affidavit 4/5, ¶ 6.

The operation of the cooling system is controlled manually from a local control panel. High spent fuel pool temperature, high/low spent fuel pool water level and low spent fuel pool cooling pump discharge pressure are annunicated in the control room which is continuously manned. In addition, the opening of spent fuel pool pump breakers, and high/low CCW flow alarms are annunciated in the control room. This instrumentation is sufficient to alert the operators in the event of abnormal conditions in the spent fuel pool. Local indication is also provided for pump discharge pressure, heat exchanger inlet temperature and heat exchanger outlet temperature. Houghtaling Affidavit 4/5, ¶ 6.

In the highly unlikely case of an extended loss of forced cooling, the spent fuel pool might boil. However, there are several sources of water on the site available to the pool; namely, the Refueling Water Tank ("RWT"), the Primary Water Tank ("PWT"), and the city water tank. In addition, there is water available via a crosstie to the intake cooling water ("ICW") system. This system is seismic Category I and is capable of providing the design capacity of 150 gpm to the spent fuel pool. Adequate time exists for makeup water sources to be utilized and, 150 gpm is more than adequate to maintain the spent fuel pool level under maximum abnormal heat load (full core discharge) conditions. Houghtaling Affidavit 4/5, ¶ 8.

The spent fuel pool cooling system normally operates with one pump in service. Water in the pool circulates around the fuel bundles removing heat by forced convection. The heated water is drawn from the pool by the spent fuel pool cooling pump. It then passes through the spent fuel pool heat exchanger, transferring the decay heat to the CCW system. Failure of a spent fuel pool cooling pump or loss of CCW to the heat exchanger is annunicated in the control room. Sufficient time exists (on the order of hours) for the operators to diagnose and resolve the

problem. The CCW system, which removes heat from the spent fuel pool heat exchanger, also removes heat from the safety related and other systems in the plant during normal modes of operation, and removes decay heat and provides containment cooling after a design basis accident. Houghtaling Affidavit 4/5, ¶ 9.

The CCW system consists of three loops: an A safety loop, a B safety loop and an N common non-safety loop. The A and B loops each have their own pumps, which are independently powered. There is a C pump which can be aligned either to the A or B loop. The A and B loops are redundant and are capable of removing the abnormal maximum heat load from the spent fuel pool during operation with only 1 loop operating. The CCW system water is treated to inhibit corrosion. The system is monitored on a routine basis. Houghtaling Affidavit 4/5, ¶ 10.

Section 9.1.3 of the SRP provides acceptance criteria for the NRC Staff's review of the heat transfer capability of the spent fuel pool cooling system under normal and abnormal discharge conditions. The acceptance criteria are derived from 10 CFR Part 50, Appendix A, General Design Criterion 44. Section 9.1.3 of the SRP states that the temperature of the pool should not exceed 140°F under normal discharge conditions and that, under abnormal (full core) discharge conditions, the pool temperature should not exceed boiling. Singh Affidavit 4/5, ¶ 22.

The Licensee has performed a heat transfer analysis of the spent fuel pool under normal and abnormal discharge conditions. This analysis, as required by Section 9.1.3, is modeled after NRC Branch Technical Position ASB 9-2 and shows that the maximum temperature which the pool will reach under normal discharge conditions is 133.3°F. Singh Affidavit 4/5, ¶ 23. Under abnormal, or full core, discharge conditions, the maximum calculated temperature is 150.8°F. Id. Clearly, these values do not exceed the acceptance criteria set forth in Section 9.1.3 of the SRP. Furthermore, based on its own independent calculations, the NRC Staff, in its March 11, 1988 Safety Evaluation "concludes that the licensee has properly calculated the heat generation rate in accordance with the guidelines of the SRP." Singh Affidavit 4/5, ¶ 25.

The increase in fuel storage will also not affect the reliability and testability of the spent fuel pool cooling system. First, all of the system's components, including electrical components, have been specified to operate continuously without degradation at their maximum design temperature. Houghtaling Affidavit 4/5, ¶ 17. These design temperatures are above the expected maximum operating temperatures. Id. Second, the electrical equipment associated with the spent fuel pool cooling system is remotely located and, thus, is not affected by the pool area environmental conditions. Id. Third, all critical

components are tested regularly, as provided in the Plant Technical Specifications. Id. There has been no change in any of the testing requirements. Id.

In addition to concerns regarding heat transfer calculations and the effects of environmental conditions on pool cooling system components, the Intervenor also asserts, in Part b of his bases, that a delay in providing makeup emergency water could cause the zirconium fuel rod cladding to fail, and possibly even "catch on fire."

A zirconium cladding/water reaction, however, is only possible at temperatures above those which can be achieved when the spent fuel pool water level is such that the fuel assemblies are partially covered. Singh Affidavit 4/5, ¶ 28.

The Licensee has performed a loss of forced cooling analysis and determined that the time from loss of forced pool cooling until the pool water boils for the normal discharge condition is approximately 13 hours. For the abnormal (full core) discharge condition, it is 5 hours. Singh Affidavit 4/5, ¶ 26. Moreover, if the pool were to boil, it would take an additional 92 hours for the normal discharge case and 46 hours for the abnormal discharge case, before the fuel would begin to be uncovered. Singh Affidavit 4/5, ¶ 27. These long lead times would provide sufficient time to allow appropriate action, such as providing makeup water. Singh Affidavit 4/5, ¶ 28; Houghtaling Affidavit 4/5, ¶ 18.

In particular, with respect to makeup water, multiple sources are available. As noted earlier, these sources include the RWT, the PWT and the city water tank. Houghtaling Affidavit 4/5, ¶ 8. In addition, there is makeup water available from the ICW system at a rate of 150 gpm. Id. Given the time available for remedial action, there is sufficient time to supply makeup water to the pool in the event of loss of forced cooling and pool boiling, and 150 gpm is more than adequate to maintain pool water level under maximum abnormal heat load conditions. Houghtaling Affidavit 4/5, ¶¶ 8 & 18; Singh Affidavit 4/5, ¶¶ 26 & 28.

In addition, no fuel damage will result from the boiling of water in the pool itself. Before boiling occurs, heat is transferred to the pool water by convection. As the water temperature increases, so does the temperature of the fuel until bubbles of steam begin to form on the surface of the fuel rods. This is called Nucleate Boiling. Singh Affidavit 4/5, ¶ 29.

With the onset of Nucleate Boiling, heat moves rapidly into the water. The Nucleate Boiling mode of heat transfer is very effective for cooling the fuel rods, and their surface temperature would stabilize at less than 300°F, well below the temperature at which any cladding damage can occur. Further, at this temperature the heat transfer mode is well within the Nucleate Boiling regime, with no possibility of the occurrence of Departure from Nucleate Boiling (DNB). Singh Affidavit 4/5, ¶ 30.

In sum, operation of the spent fuel pool cooling system will maintain pool temperatures below the NRC requirements stated in the SRP, for both normal and abnormal discharge conditions. There is no increased risk of loss-of-cooling or reduction in system reliability or testability due to the reracking. Furthermore, in the event of a loss of forced cooling, the long lead times before the onset of pool boiling will allow the multiple sources of makeup water to be utilized in sufficient time to prevent boiling. Even should forced cooling not be restored and boiling occur, the fuel would remain well covered with water, at a safe temperature.

D. 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 core discharge conditions. Memorandum and Order, p. 21 (April 20, 1988).

The mechanical portions of the spent fuel pool cooling system ("SFPCS") have been described in detail in connection with Admitted Contention 4, supra. Essentially, the SFPCS consists of two 1500 gpm centrifugal pumps, one heat exchanger, piping and associated valves and instrumentation. Houghtaling Affidavit 4/5, ¶¶ 5 & 6. During normal conditions, one fuel pool pump and the heat exchanger are placed in service. Houghtaling Affidavit 4/5, ¶ 9. During abnormal, or full core discharge, conditions, two fuel pool pumps and the heat exchanger are in service. Singh Affidavit 4/5, ¶ 17. The fuel pool water is drawn from the pool near the surface and is circulated by the pump(s) through the heat exchanger where the decay heat is rejected to the CCW system. Houghtaling Affidavit 4/5, ¶ 5. The cooled pool water is returned to the bottom of the fuel pool at the opposite end of the pool from the intake. Id.

The CCW system takes the decay heat from the SFPCS through the heat exchanger and rejects it to the ICW system. Houghtaling Affidavit 4/5, ¶¶ 11. The CCW flows on the shell side of the heat exchanger, while the SFPCS water flows in the tubes. Singh Affidavit 4/5, Figure 2. The CCW system consists of three loops: an A safety loop, a B safety loop and an N common non-safety loop. The A and B loops each have their own pumps, which are independently powered. There is a C pump which can be aligned either to the A or B loop. The A and B loops are redundant and are capable of removing the abnormal maximum (full core discharge) heat load from the spent fuel pool during operation with only 1 loop operating. The CCW system is monitored on a routine basis. Houghtaling Affidavit 4/5, ¶ 10.

The SFPCS is controlled and monitored from a local instrument panel, except that the following alarms are annunciated in the control room: (1) low head pressure for the pool pump(s) discharge; (2) high pool temperature; (3) low and high pool water level; and (4) opening of the fuel pool pump(s) breakers. Houghtaling Affidavit 4/5, ¶ 6. These instruments are sufficient to alert operators of unusual conditions in the spent fuel pool. Id. In addition, high and low CCW flow is annunicated in the control room. Houghtaling Affidavit 4/5, ¶¶ 6 & 9.

The spent fuel pool cooling pumps are powered by independent power supplies. The pumps are capable of receiving backup power from the Emergency Diesel Generators, and sufficient time and capacity exist to do so. All active components in the

CCW system are powered from safety related power sources. All A train components receive A power and all B train components receive B power. Both the components and power supplies are separate and independent. The CCW system is loaded onto the Emergency Diesel Generators in the event of loss of offsite power. Houghtaling Affidavit 4/5, ¶ 7.

A single active failure of a SFPCS pump, either the pump or power supply, will reduce the cooling flow to that of one pump. Houghtaling Affidavit 4/5, ¶ 15. However, if this occurs the maximum pool temperature will be 133.3°F for the normal discharge case, and less than 167°F for the abnormal, full core, discharge case. Id. Both of these temperatures are acceptable under the guidance provided in the SRP, even when no failures are assumed. Houghtaling Affidavit 4/5, ¶ 14.

A single active failure of a pump in the CCW system (on the shell side of the SFPCS heat exchanger), would not further reduce cooling to the spent fuel pool because there is a spare pump available to the CCW system. Houghtaling Affidavit 4/5, ¶ 15. A failure of a power supply to one train of the CCW system could reduce the CCW system to one pump. Id. This is because, depending on the failure, it might not be possible to align the spare pump to the electrical bus being used by the operating CCW pump. Id. However, even under this scenario, the CCW system is fully capable of removing the required heat load, even under full-core off load conditions, with only one pump operating. Id.

The above discussion demonstrates that a total loss of forced cooling due to pump failure or loss of power is unlikely. Houghtaling Affidavit 4/5, ¶ 16. There are two SFPCS pumps, which are independently powered. For both pumps to fail by either a mechanical failure or loss of power supply is considered unlikely. The CCW system, to which the SFPCS transfers decay heat, is fully redundant, and no single active failure can reduce its capacity below that required. Id.

In addition, the threat of a loss of cooling event is further reduced because all of the components in the SFPCS and CCW system have been specified to operate continuously without degradation at their maximum design temperatures; temperatures that are above the expected maximum operating temperatures. Houghtaling Affidavit 4/5, ¶ 17. Moreover, the electrical equipment associated with the SFPCS is remotely located and, thus, is not affected by pool area environmental conditions. Id. All critical components are tested on a regular basis in accordance with the Plant Technical Specifications. Id. These accepted testing criteria have not changed as a result of the increase in fuel storage and, thus, the reliability and testability of the system will not be reduced. Id.

As discussed in detail in connection with Admitted Contention 4, supra, even a total loss of forced cooling would not result in fuel damage. Briefly, in the unlikely event of a total loss of forced cooling, it would take approximately 13 hours under normal discharge conditions and 5 hours under abnormal discharge conditions for the spent fuel pool to boil. Houghtaling Affidavit 4/5, ¶ 18. Furthermore, even if the pool were to boil, it would take approximately 52 additional hours to reach the minimum acceptable water level of 10 feet above the fuel (as specified in the SRP) for the normal discharge case. Id. For the abnormal discharge case, it would take approximately 26 additional hours to reach 10 feet above the fuel. Id. This amount of time is sufficient to provide makeup water to the spent fuel pool. Id.

There are many sources of makeup water available, including: the RWT; the PWT; and the city water tank. Houghtaling Affidavit 4/5, ¶ 8. In addition, makeup water is available via a crosstie to the ICW system. Id. This system alone is capable of providing 150 gpm to the spent fuel pool, which is more than adequate to maintain spent fuel pool level even under maximum abnormal heat load conditions. Id.

Given the multiple sources of makeup water and the fact that these sources are capable of supplying sufficient water to maintain the pool level even under the worst postulated conditions, the fuel will remain covered and the bulk pool temperature will not exceed boiling. Houghtaling Affidavit 4/5, ¶ 18. Even if the pool boils, the maximum fuel cladding temperature will be maintained well below the point where any fuel damage would occur because the spent fuel pool is not pressurized and, therefore, the bulk pool temperature cannot exceed the boiling temperature. Id.; Singh Affidavit 4/5, ¶¶ 27-30.

In sum, and 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, in the event of a single failure of any of the pumps on the shell side of the SFPCS heat exchanger and/or in the case of a single failure of the electrical power supply to the pumps on the pool side of the heat exchanger. Houghtaling Affidavit 4/5, 4 & 15. In addition, even if a complete loss of forced cooling were to occur, the fuel would be kept covered and maintained at a safe temperature. Id., ¶ 19. This conclusion is based on the multiple sources of makeup water available and the long lead time before the pool water could reach an unacceptable level. Therefore, the increased fuel storage capacity has not affected the SFPCS in an unacceptable way.

E. 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, p. 24 (April 20, 1988).

Intervenor, in responding to specific interrogatories concerning Admitted Contention 6, offered definitions of the terms "untested" and "new and unproven" as follows:

> "Untested," implies that the means by which the Boraflex material was determined to be a suitable material for extended use as a neutron poison in a highly radioactive environment was inadequate in simulating the stresses to be encountered in actual usage.
>
> "New and unproven," means that the durability and neutron attenuation capability of the Boraflex material over the projected service

> life of the material has not been substantial or demonstrated by any credible, testing program.

Intervenor's Response at 3. Intervenor also stated that: "Licensee does not adequately address the issues of shrinkage, degradation, inservice surveillance, durability, corrective actions to be taken in the case of degraded, Boraflex specimens and the structural integrity of the Boraflex material and its cover plate under all design conditions." Intervenor's Response at 3.

In his bases for Admitted Contention 6, Intervenor referenced NRC Information Notice No. 87-43, "Gaps in Neutron Absorbing Material in High-Density Spent Fuel Storage Racks," ("Information Notice"). The purpose of the Information Notice was to:

> alert recipients to a potentially significant problem pertaining to gaps identified in the neutron absorbing component of the high-density spent fuel storage racks at Quad Cities Unit 1. The safety concern is that certain gaps might excessively reduce the margin of nuclear subcriticality in the fuel pool.

Information Notice at 1. The gaps were attributable to mechanical restraint of Boraflex undergoing shrinkage caused by irradiation. Turner Affidavit, ¶ 36.

As noted above, in its April 20, 1988, Memorandum and Order the Licensing Board cited the NRC Staff's "Board Notification regarding Anomalies in Boraflex Neutron Absorbing Material (BN-87-11)" ("Board Notification"). This Board

Notification indicated that "[t]he results of inspections performed by two utilities of Boraflex neutron absorber (poison) material used in their spent fuel pools (SFPs) have identified anomalies in the Boraflex." Board Notification at 1. The Board Notification also provided the reports of the two utilities concerning the Boraflex anomaly and stated:

> The Point Beach Report indicates that small samples in which significant deterioration was identified were not found to be representative of the full size Boraflex sheets used in their SFP racks. The Quad Cities Report indicated that numerous gaps were found in the Boraflex in the racks due to shrinkage of the Boraflex material. The report further indicated that the design still maintained the SFP's criticality below 0.95 k-eff.

_Id_. In response to the information, the Board Notification indicated that the NRC Staff had requested additional information from utilities "to determine the significance of the identified anomalies, any changes in their existing inservice surveillance programs for Boraflex and any corrective actions to be taken if determined necessary." _Id_. at 2.

Only the gap anomaly in the _Quad Cities_ storage racks was ultimately determined to be of significance. Discoloration of the Boraflex and the absorption of spent fuel pool water observed at _Point Beach_ were determined to have no effect on the neutron absorption capability of the Boraflex. Turner Affidavit, ¶ 37.

Experimental irradiation programs conducted on Boraflex subsequent to the discovery of gaps at Quad Cities have shown that, upon irradiation, Boraflex undergoes shrinkage and hardens. The material then exhibits increased compressive strength and reduced tensile strength. Gaps may occur if the Boraflex panels are too rigidly constrained mechanically, preventing their free contraction when shrinkage occurs under irradiation. However, the same irradiation programs (and prior evidence) confirmed that there is no loss in the boron-10 content and, therefore, that the Boraflex is capable of continuing to perform its intended function of maintaining reactivity within the acceptable limit. In the manufacture of the St. Lucie 1 storage racks, care was exercised to avoid mechanical constraint that might contribute to the formation of significant gaps in the Boraflex. Turner Affidavit, ¶¶ 37 & 38.

Neutron attenuation is achieved in the St. Lucie 1 spent fuel storage racks through the combined action of water and a widely used neutron absorbing material -- Boraflex. Since the early 1980's, Boraflex has become the preferred neutron attenuation material in high-density racks. Singh Affidavit 3/6, ¶ 12. Twenty-seven nuclear power reactors have used Boraflex as a neutron absorbing material. Singh Affidavit 3/6, Table A.

Boraflex is comprised of a polymeric silicone encapsulant entraining and fixing fine particles of boron carbide in a homogenous, stable matrix. The inherent stability of both silicones and carbides results in their compatibility with a variety of chemical environments concurrent with strong ionizing radiation. The physical characteristics of Boraflex allow fabrication and handling in continuous pieces. Singh Affidavit 3/6, ¶ 13.

Substantial information pertaining to the performance characteristics of Boraflex has been developed from extensive testing. The results of this testing demonstrate the suitability of Boraflex for use as a neutron absorber in spent fuel pool environments. Singh Affidavit 3/6, ¶ 15. Under testing, Boraflex exhibits excellent heat aging characteristics. Testing has also been performed confirming the stability of the material in various chemical environments, including high-temperature, borated water. Id., ¶ 16. The neutron absorption of Boraflex was measured at various boron-10 loadings to confirm the absorptive characteristics of the material. The measurements were made at neutron energies representative of thermal neutrons which could cause fission. Id., ¶ 17. Radiation exposure tests of Boraflex at total equivalent doses of 10^{12} rads were performed. This test program was designed to determine the physical and chemical characteristics of Boraflex under a variety of radiation levels, radiation rates and environments. Id., ¶ 18.

The samples were evaluated for the effects of irradiation on a number of physical and chemical characteristics. All evidence from the tests suggest that, at the exposure levels expected, Boraflex maintains sufficient bend tolerance to withstand normal and anticipated conditions of service in storage rack applications. Evaluation of the data reveals no discernible effect of either environment or irradiation on neutron absorption. The tests resulted in no significant leaching of either boron or halogens. Gas generation due to irradiation has been evaluated and found to be negligible. Singh Affidavit 3/6, ¶ 19.

Most recently, Boraflex has been tested under carefully controlled conditions to determine precisely its dimensional changes under irradiation. Analysis indicates total in-plane shrinkage to be less than 2.5%. This shrinkage has been anticipated and accounted for in the St. Lucie 1 rack design. Singh Affidavit 3/6, ¶ 20.

The maximum cumulative radiation dose to the Boraflex material in the St. Lucie 1 pool for all fuel discharges until the end of the life of the unit is not expected to exceed 10^{11} rads. This is merely 10% of the equivalent radiation dose given to this material in the laboratory tests, described above. Singh Affidavit 3/6, ¶ 21.

An inservice surveillance program at St. Lucie 1 has been established to periodically verify the continued integrity of the Boraflex neutron absorber material. Surveillance coupons (test specimens), mounted in stainless steel jackets representative of the actual rack materials and configuration, are suspended in the pool so as to be exposed to the same or greater irradiation than the Boraflex in the racks. These surveillance coupons are removed periodically, and are tested and evaluated to provide an indication of the condition and integrity of the bulk

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Boraflex in the racks. These tests include a determination of any dimensional changes, material hardness, and both neutron radiography and measurement of neutron absorptivity to assure the continuing effectiveness of the Boraflex in providing reactivity control. Turner Affidavit, ¶ 40.

All 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 Affidavit 3/6, ¶ 22. Consistent with practice in the industry, the St. Lucie 1 rack modules are of two basic types, commonly referred to as Region 1 and Region 2. Id., ¶ 23. Region 1 is designed to accommodate fresh unirradiated fuel and Region 2 is designed for spent fuel of a minimum specified burnup that, in turn, depends upon the initial enrichment of the particular fuel batch. Each of the two regions of the fuel storage rack, therefore, has different design criteria, provides for a different boron-10 loading in the Boraflex absorbers, and utilizes different fuel assembly spacings. Turner Affidavit, ¶ 6. Region 1 modules are of the so-called "flux-trap" construction, which is an industry standard for modules of this type. In this construction, square cross-section tubes of 8.65 inch inside dimension are produced by seam welding two identical channels. The seam welding equipment and process used are examples of standard technology used in the manufacturing of the racks of the Joseph Oat Corporation. Singh Affidavit 3/6, ¶ 24.

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[illegible]

The Region 1 fuel assembly storage boxes are arranged on a nominal center-to-center spacing of 10.12 inches. Turner Affidavit, ¶ 7.

Every Region 1 box is equipped with a sheet of Boraflex on each of its four sides. The Boraflex sheet, or panel, is positioned in place by a stainless steel sheathing which also serves to protect the Boraflex material from accidental dents. The stainless steel boxes surrounded by Boraflex on four sides are also a universally employed technology. Singh Affidavit 3/6, ¶ 25. Joining of the boxes to produce a honeycomb construction is yet another aspect of proven and widely used technology employed in the fabrication of the St. Lucie 1 racks by the Joseph Oat Corporation. Id., ¶ 26.

Region 2 rack modules are designed to store fuel with a specified minimum burnup. These modules are constructed from the same basic elements as the Region 1 rack; namely, a solid base plate, seam welded boxes, adjustable support legs and Boraflex. A flux-trap gap (water gap) is not required for Region 2 modules and, therefore, is not provided. Singh Affidavit 3/6, ¶ 28. As a result, the Region 2 boxes are arranged on a nominal center-to-center spacing of 8.86 inches. Turner Affidavit, ¶ 9. The Boraflex panels are positioned in place by the contiguous walls of the boxes, and suitably located peripheral strips. Singh Affidavit 3/6, ¶ 28.

All of the above-mentioned features of the St. Lucie 1 storage rack construction are routinely used in the industry, and represent total conformance with the industry norm. Singh Affidavit 3/6, ¶ 29. Joseph Oat Corporation has extensive experience in the manufacturing of spent fuel storage racks using Boraflex panels. Singh Affidavit 3/6, ¶ 31. All significant construction features of the Region 1 and Region 2 racks for St. Lucie 1 are direct adaptations of established technology. Moreover, the production control methods in use at the Joseph Oat Corporation are derived from two decades of nuclear component manufacturing experience. Id., ¶ 32.

For the St. Lucie 1 storage racks, established industry practice in rack construction was followed in the installation of Boraflex, with one difference based on experience. Previous manufacturing practice called for use of a silicone based adhesive to cement the material in its place. For example, this practice was followed for the Quad Cities spent fuel racks. The experience gained with the use of Boraflex in operating plants indicates that the twin effects of irradiation, namely shrinkage and hardening, are best accommodated by providing complete freedom of Boraflex panels to undergo in-plane dimensional changes, and by installing Boraflex panels large enough to allow the neutron absorption function to be carried out even after shrinkage. Accordingly, for the St. Lucie 1 racks, the cementing glue was eliminated, and provision was made in the manufacturing process to install Boraflex in-place, with minimal surface loading. Singh Affidavit 3/6, ¶ 30.

The Quad Cities racks, also built by Joseph Oat Corporation, utilized a different design from the St. Lucie 1 storage racks. The fabrication process associated with installation of the Boraflex was also different. The storage cells were not assembled from boxes. Singh Affidavit 3/6, ¶ 33. Rather, the Quad Cities racks employ the so-called "cruciform" construction wherein angles are welded together along the edges in a fixture to form a cruciform. The Boraflex is contained between the faces of the angle. The cruciforms are attached to each other by welding along their junction. This welding must be done remotely, and therefore, its quality depends on the flatness and straightness of the cruciform surfaces. It was this fabrication process, as well as the use of adhesive during Boraflex installation, that led to excessive restraint of the Boraflex panels and their subsequent cracking and gap formation following shrinkage upon irradiation at the Quad Cities facility. Singh Affidavit 3/6, ¶ 34. The unconstrained Boraflex panel at Point Beach showed no gaps or breakage when removed for inspection after extensive gamma and neutron exposure. Id., ¶ 35. The St. Lucie 1 spent fuel storage racks are designed to provide complete in-plane dimensional changes of Boraflex. Id., ¶ 35.

The driving mechanism of any Boraflex degradation is radiation induced change in the structure of the Boraflex material. Changes in Boraflex structure are significant only to the extent that they cause or result in the loss of boron and, hence, reduce the effectiveness of the Boraflex in controlling the reactivity. Tests have confirmed that no significant loss of boron occurs under irradiation up to total equivalent radiation levels in excess of those expected through the expiration of the St. Lucie 1 operating license on March 1, 2016. In some of these tests, the Boraflex has been irradiated to accumulated equivalent doses in excess of 10^{12} rads. By comparison, the materials in the St. Lucie 1 spent fuel storage rack, including Boraflex, are expected to be exposed to radiation dose levels less than 10^{11} rads through the expiration of the St. Lucie 1 operating license. Thus, the test irradiation programs have exposed Boraflex samples to the equivalent of hundreds of years use in the St. Lucie 1 spent fuel pool. Turner Affidavit, ¶ 39.

Based on the foregoing, it can be concluded that the design and fabrication of the St. Lucie 1 spent fuel storage racks incorporate proven technology for Boraflex installation and positioning. The problems identified in the NRC Staff Board Notification and Information Notice, referenced above, have been addressed in the design and fabrication of the storage racks for the St. Lucie 1 spent fuel pool expansion. The underlying causes of the Quad Cities Boraflex problems have been eliminated. The rack modules at St. Lucie 1 incorporate refinements of established technology. All aspects of their construction embody proven design concepts and well-established fabrication techniques. Singh Affidavit 3/6, ¶ 36.

The Boraflex absorber material is expected to perform its intended function for the storage lifetime of the St. Lucie 1 racks. Furthermore, it is not expected that any gaps that might form in the Boraflex absorber sheets would increase k-eff above the limiting value of 0.95, even if gaps as large as those observed at Quad Cities occurred. Further, the planned St. Lucie 1 Boraflex surveillance program is adequate to reveal the onset of any greater-than-expected degradation, well in advance of its becoming a significant problem. There would be ample opportunity to take any corrective action that might be necessary. Turner Affidavit, ¶ 41.

F. 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).

Intervenor, in his response to Licensee's interrogatories concerning Admitted Contention 7, in effect questioned whether the requirements of ANSI N16.1-1975 would be met; expressed a view that the greater number of fuel assemblies in the St. Lucie 1 pool would raise the effective multiplication factor (k-eff) closer to 1.0, i.e., criticality; and questioned whether General Design Criterion 62 of 10 C.F.R. Part 50, Appendix A would be met for the increased storage capacity. However, no specifics as to the bases for these concerns were provided. 2/ In addition, it appears from Intervenor's response to Licensee's specific interrogatory number six, concerning Admitted Contention 7, that Intervenor may believe that criticality can be caused by the loss of coolant accident (complete draining of the pool) considered in the BNL Report, referenced above. See Intervenor's Response at 4.

At the time the instant amendment to the St. Lucie 1 operating license was requested, the storage racks at St. Lucie 1 were approaching their capacity. A high-density storage rack design utilizing Boraflex absorber plates, comparable to designs currently being used in numerous other nuclear plants in this country and abroad, was employed in order to provide increased capacity for safe storage of spent fuel in the St. Lucie 1 spent fuel pool (from 728 fuel assemblies to 1706 assemblies). Turner Affidavit, ¶ 5. Reracking of a spent fuel storage pool, such as was undertaken at St. Lucie 1, will not result in a proportional increase in radiation intensity or heat generation rates, but will, in fact, result only in minor increases as aged fuel accumulates in the pool. Turner Affidavit, ¶ 14.

The expanded fuel storage racks at St. Lucie 1 are of the two-region design, with Region 1 racks being designed to accommodate fresh unirradiated fuel and Region 2 racks designed for storage of spent fuel of a minimum specified burnup that, in

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2/ In fact, Intervenor's references to "ANSI-N16-1975," in both the contention and response to interrogatories, appear to be in error. FPL assumes the reference is to ANSI N16.1-1975.

turn, depends upon the initial enrichment of the particular fuel batch. Region 1 racks and Region 2 racks meet different design criteria, provide for different boron-10 loadings in the Boraflex absorber and utilize a different fuel assembly spacing. Turner Affidavit, ¶ 6.

Region 1 is designed to safely accommodate fuel of the highest reactivity anticipated to be stored, i.e., fresh unirradiated (unburned) uranium fuel, enriched to 4.5 weight percent in the uranium-235 (U-235) isotope. Since fresh fuel is more highly reactive than fuel which has been in the reactor, fuel of any burnup may be stored in Region 1 with the assurance that the k-eff will be less than the maximum design case. Region 1, therefore, is intended to provide safe storage for fresh fuel to accommodate a full core off-load, when required, and to store fuel whose burnup does not satisfy the criteria for storage in Region 2 of the pool. Turner Affidavit, ¶ 7. Region 2 provides storage of spent fuel of sufficient burnups. Id., ¶ 8.

Reactivity of the fuel assemblies decreases substantially as burnup is accumulated (and fissile material is depleted). Region 2 is designed to safely store fuel of 4.5 weight percent initial enrichment which has accumulated a burnup of at least 36,500 megawatt days per metric ton of uranium (MWD/MtU). A similar minimum or limiting fuel burnup has also been established analytically for fuel of lower fuel enrichments, and these data define the bounding condition of burnup for acceptable storage in Region 2. For any given initial enrichment, fuel assemblies with burnup equal to or greater than the bounding condition may be safely stored in Region 2, while assemblies having less than the minimum required burnup will be stored in Region 1. Turner Affidavit, ¶ 8.

Fissile material refers to material, the atoms of which are capable of being split or fissioned, with the attendant production of heat energy, upon the capture (absorption) of neutrons. The primary fissile material in the new fuel assemblies of most nuclear power reactors, including St. Lucie 1, is uranium-235. The nuclear fuel used at St. Lucie 1 may be enriched up to 4.5% by weight of uranium-235. Turner Affidavit, ¶ 10.

The k-eff or reactivity of fuel assemblies is affected primarily by the uranium-235 enrichment of the fuel, and by the quantity of neutron absorbing materials (poisons) present. Changes in k-eff may be produced by several different mechanisms. Increasing the fuel enrichment increases the fuel's reactivity, as does increasing the density of fuel assemblies in the spent fuel pool, or decreasing the concentration of poisons. For this reason, Region 1 of the St. Lucie 1 racks is designed to safely store fuel assemblies of the highest enrichment permitted to be at the site. Conversely, the reactivity of fuel elements in a spent fuel pool can be decreased by decreasing the enrichment of uranium-235 in stored fuel assemblies, by decreasing the density of the stored fuel assemblies, or by increasing the concentration of poison. In this latter regard, neutron absorbing poisons may be intentionally installed in the storage racks to reduce the system's reactivity. This is accomplished in the St. Lucie 1 racks by the installation of Boraflex as a neutron absorbing or poison material, and by the use of soluble boron as additional poison material. Turner Affidavit, ¶ 16.

The effective multiplication factor, commonly called k-effective or k-eff, is a measure of the ability of a system to sustain a fission reaction. Criticality occurs whenever the k-eff reaches or exceeds a value of 1.0, because, under those conditions, at least as many new neutrons are being produced as are being lost by capture and leakage. For a k-eff less than 1.0, the fission rate cannot be sustained. The margin below a k-eff of 1.0 is the safety margin to criticality, and the subcritical margin is the difference between 1.0 and k-eff of the given system. NRC guidelines for fuel storage racks require that the maximum k-eff, including all known uncertainties, be equal to or less than 0.95, a value which provides a substantial subcritical margin. Turner Affidavit, ¶ 13.

Extensive guidance on the subject of criticality is available in the form of industry standards and NRC regulatory guidance. Turner Affidavit, ¶¶ 18-23. Criticality analyses for spent fuel pools are governed by the requirements stated in General Design Criterion 62 of 10 CFR Part 50, Appendix A (1988) as follows: "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." Turner Affidavit, ¶ 17.

The most definitive clarification of NRC guidance is provided in NRC's "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (April 14, 1978), which sets forth in detail the NRC acceptance criteria for spent fuel storage pools. Section III.1.5 of this guidance emphasizes that the "neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions." (Emphasis in the original.) Turner Affidavit, ¶ 22.

Moreover, section III.1.2 (Postulated Accidents) of the April 14, 1978 guidance invokes the double contingency principle of ANSI N16.1-1975 for fuel pool analyses, stating that:

> The double contingency principle of ANSI N16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident. Realistic initial conditions (e.g., the presence of soluble Boron) may be assumed for the fuel pool and fuel assemblies.

Turner Affidavit, ¶ 22.

The industry standards and NRC guidance on criticality limiting the maximum k-eff to 0.95, including all uncertainties, provide a substantial subcriticality margin as a factor of safety to assure conformance with General Design Criterion 62 and to preclude the possibility of a criticality incident in the storage facilities. The design and criticality safety analyses of the expanded fuel storage facilities for St. Lucie 1 were performed in accordance with and conform to these standards and guidance. Turner Affidavit, ¶ 23.

Two independent methods, the KENO-IV and CASMO-2E codes, were used to calculate the reference k-eff for the St. Lucie 1 spent fuel storage racks. KENO-IV is a multigroup Monte Carlo code used extensively in the nuclear industry for the criticality evaluation of spent fuel racks. CASMO-2E is a multigroup transport theory computer code used for k-eff analysis and depletion calculations for fuel assemblies. Turner Affidavit, ¶ 24.

Both of these calculational methods were benchmarked against critical experimental data for configurations as nearly representative as possible of the actual spent fuel storage geometry, consistent with NRC regulatory guidance. These benchmark calculations established a bias and uncertainty which were incorporated into the analysis of the maximum k-eff of the racks. The methods of criticality analysis used for the St. Lucie 1 storage pool were previously employed in comparable analyses of racks at other nuclear plants, which were reviewed and approved by the NRC. Turner Affidavit, ¶ 25.

Fuel burnup (depletion) calculations were performed using the CASMO-2E two-dimensional transport theory code, which explicitly describes and tracks the compositions of each individual fuel pin in a fuel assembly. Turner Affidavit, ¶ 26. In using the spent fuel compositions from the CASMO-2E calculations

to analyze the k-eff of the storage racks, isotopic compositions at reactor shutdown were used and the concentration of xenon-135 (the fission product with the highest absorption cross section) was set to zero to assure a conservative estimate of the poisoning effect of the fission products. Id., ¶ 27.

Calculations with CASMO-2E and KENO-IV explicitly described each fuel pin, each Boraflex sheet and the stainless steel material used in the Region 1 rack structure. Spacer grids in the fuel assemblies (non-productive absorber material that would reduce the k-eff) were conservatively neglected. Neutron leakage from the racks was also conservatively neglected, which assures that the true reactivity will be less than the value calculated. The nominal k-eff calculated for an infinite array of Region 1 assemblies was 0.9313 without credit for the (redundant) reactivity control provided by the soluble boron in the pool water. Turner Affidavit, ¶ 28. CASMO-2E calculations (as confirmed by KENO-IV) for an infinite array of Region 2 storage cells resulted in a nominal k-eff of 0.9114. Id., ¶ 29.

The NRC acceptance criteria for criticality analyses require consideration and inclusion of all known uncertainties in the calculation of k-eff, as discussed above. These encompass uncertainties in the calculational methods as well as with mechanical tolerances in storage rack manufacture and in fuel assembly fabrication. Given these uncertainties, the actual k-eff may be higher or lower than the k-eff values calculated for the nominal design case. For conservatism, the total uncertainty is added to the nominal k-eff in calculating the maximum k-eff value, thereby assuring that the highest possible k-eff value is used. Uncertainties in the evaluation include boron-10 concentration in the Boraflex, Boraflex thickness, Boraflex width tolerance, storage cell center-to-center spacing tolerances, stainless steel thickness tolerances, fuel pin lattice spacing, fuel enrichment and density tolerances, and possible eccentric positioning of fuel in the cells. The Region 2 analysis also requires inclusion of an allowance for uncertainties in the burnup calculations. With all uncertainties included, the maximum possible k-eff values are 0.9409 for Region 1 and 0.9435 for Region 2. Since all uncertainties are included in the maximum k-eff values, the safety margins specified in the NRC acceptance criteria are conservative. With the required soluble boron of 1720 ppm in the pool water, the maximum reactivities in Regions 1 and 2 are 0.767 and 0.760, respectively. Turner Affidavit, ¶ 30.

Potential accident conditions were also evaluated in the criticality safety analysis of the St. Lucie 1 storage racks. The accidents considered included the following: increased temperature, boiling, dropped assembly, and abnormal assembly location. Turner Affidavit, ¶ 32. The largest positive reactivity effect would occur if a fresh fuel assembly of 4.5% enrichment were to be accidentally installed in a Region 2 storage cell with the surrounding cells assumed to be fully loaded with fuel of the highest permissible reactivity. The soluble poison normally present in the pool water would maintain the k-eff substantially below the limiting k-eff value of 0.95 and would assure that criticality could not be obtained even if Region 2 were fully loaded with fresh fuel. The St. Lucie 1 criticality calculations for the high density storage racks considered the double contingency principle of ANSI N16.1-1975. In invoking the double contingency principle, NRC specifically permits credit for soluble boron to be taken under accident conditions. Therefore, the soluble boron in the St. Lucie 1 pool water is adequate to protect against the potential accident of an abnormally located fuel assembly and assure that 0.95 is not exceeded under all credible accident conditions. Turner Affidavit, ¶ 33.

In addition to the accident conditions discussed above, the criticality consequences of a dropped cask accident were also considered. Fuel failures (ruptures) do not directly affect criticality. In general, reduced fuel spacing (such as might be the consequence of a cask accident) results in lower k-eff values because of the reduced moderation and the higher concentration of non-productive neutron absorption. Furthermore, the presence of the soluble poison provides assurance that a cask drop accident cannot cause a criticality accident. Turner Affidavit, ¶ 34. In addition, assuming that, somehow, the pool was allowed to drain, criticality would not occur. This is the case since, without water, there would be no neutron moderator in the pool -- a necessary condition for criticality. Weber Affidavit 1, ¶ 6.

As documented in the SE for the St. Lucie 1 spent fuel pool expansion, the NRC staff reviewed the Licensee's criticality analysis methods and results, and found the criticality aspects of the design of the high density St. Lucie 1 spent fuel storage racks to be acceptable. SE, § 2.3, March 11, 1988.

In summary, the criticality safety analyses for the St. Lucie 1 storage racks were performed in accordance with accepted industry practice and in conformance with all applicable regulations and guidelines, using calculational methods that have been in common use and have been previously reviewed and found acceptable by the NRC Staff. In particular, the design of the racks conforms to the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 62; guidance provided by NRC in its April 14, 1978, guidance letter; ANSI N16.1-1975; and other related guides and standards. Turner Affidavit, ¶ 42.

These analyses demonstrate that fuel assemblies of authorized initial enrichments and burnup when stored in Region 1 and Region 2 racks, have a k-eff less than 0.95, including all uncertainties, under both normal or accident conditions. The increased capacity for fuel storage and the high enrichments do not effect the pre-existing k-eff limit of 0.95 for the fuel storage pool at St. Lucie 1. Therefore, the minimum criticality safety margin associated with a k-eff limit of 0.95 is not changed and there is no increase in the probability of a criticality accident. Turner Affidavit, ¶ 42.

Furthermore, the presence of soluble boron assures a very large subcriticality margin under normal operating conditions and provides additional assurance that k-eff will be maintained less than 0.95 under all credible postulated accident conditions. Consequently, it can be concluded that the design of the St. Lucie 1 storage racks conforms to safe and conventional practices in the industry, conforms to all applicable regulations and guidelines, and provides assurance that a criticality accident can not occur under any credible postulated conditions. Turner Affidavit, ¶ 42.

III. Conclusions

Based upon the foregoing, the attached affidavits, "Licensee's Statement," and "Licensee's Memorandum," there is no genuine issue of material fact pertinent to any of Intervenor's Contentions. Licensee's Motion for Summary Disposition of Intervenor's Contentions should be granted *in toto*; and FPL respectfully requests that the Board do so, and issue a decision in Licensee's favor. If the Board identifies any issues within any contention which must be tried, however, Licensee requests that the Board grant summary disposition as to the other issues and contentions. See, e.g., Licensee's Memorandum at 2.

Respectfully submitted,

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