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Docket No. 50-335

DEC 03 1976

Florida Power & Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
Nuclear and General Engineering
Post Office Box 013100
Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. DPR-67 for the St. Lucie Plant Unit No. 1. The amendment consists of a revision to License No. DPR-67 required to authorize resumption of operation under the conditions evaluated in your filings dated October 25, 1976 and November 18, 1976, which analyzed the safety of resumed operation with repaired fuel assemblies.

The amendment 1) authorizes operation with repaired fuel assemblies and 2) requires additional monitoring during power operation if the core power distribution is being determined by use of the excore detector monitoring system when the core burnup is less than 10,000 megawatt days per metric ton of uranium.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

12-3-76 Telecon to FPL (Whittier) advised of issuance of this letter and amendment. EA Reeves ORB#2

Enclosures:
1. Amendment No. 10 to License No. DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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DATE	12/2/76	12/2/76	12/3/76	12/02/76	12/3/76

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cc w/enclosures and cy of FPL's filings dtd. 10/25/76 and 11/18/76:
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FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for authorization to operate with repaired fuel assemblies by Florida Power & Light Company (the licensee) dated October 25, 1976 and November 18, 1976, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended as follows:

A. Revise paragraph 2.B.1. of the license to read:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location on the licensee's site on Hutchinson Island in St. Lucie County, Florida, in accordance with the procedures and limitations set forth in this license with repaired fuel assemblies as described in the licensee's filings dated October 25, 1976 and November 18, 1976.

B. Add a new Section M of Enclosure 1 to the license to read:

M. The following additional monitoring in Mode 1 shall be performed if the excore mode described in paragraph 4.2.1.3 of the Technical Specifications is used when the core burnup is less than 10,000 megavatt days per metric ton of uranium.

1. The axial peaking factor shall be monitored using all operable fixed incore Rh detectors. The surveillance period shall be sufficient to ensure that the axial peaking factor increases less than 3% between measurements based on the average growth rate from the most recent measurements, but not to exceed 7 operating days.

A 3% per day growth rate in the axial peaking factor shall be assumed until measurements are available. All measured growth rates shall be doubled to account for measurement and projection uncertainties.

2. Core power maps shall be taken at least once per 7 days of accumulated operation in Mode 1 and compared with predicted distributions.

3. If an axial peaking factor in excess of the design basis value (1.5) is observed, it shall be reported to the Commission within 24 hours and the "THERMAL POWER" limit of Technical Specification 4.2.1.3.c. shall be multiplied by the factor (1.5/measured axial peaking) whenever the axial peaking factor exceeds 1.5.

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3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION
Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Date of Issuance: DEC 03 1976

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SURNAME ➤						
DATE ➤						

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By letter dated October 25, 1976 and supplement dated November 18, 1976, Florida Power & Light Company (FPL) submitted a safety analysis report providing the basis for returning St. Lucie Plant Unit No. 1 to operation with the repaired fuel assemblies. The submittals were evaluated by the NRC staff as a basis for amending Facility License No. DPR-67 for the St. Lucie Plant Unit No. 1. The amendment would (1) authorize operation with repaired fuel assemblies and (2) require additional monitoring during power operation if the core power distribution is being determined by use of the excore detector monitoring system when the core burnup is less than 10,000 megawatt days per metric ton of uranium.

DISCUSSION

Background

A power distribution anomaly was observed at the St. Lucie Unit 1 reactor beginning June 22, 1976. The reactor was being operated at 80% power. Moderator temperature coefficient tests were in progress when a quadrant tilt (as measured by the incore detectors) increased from essentially zero to between 3 and 4 percent. Simultaneously, the core-averaged axial peaking factor grew from a nominal value of 1.3 to approximately 1.5 over a period of two weeks. In addition, the reactor's critical boron concentration deviated from the expected boron concentration, with the deviation equivalent to the core being about 0.4% more reactive than expected. The axial power shape for rods-out operation remained essentially centered at the midplane of the core during this period. The unit was brought to Hot Zero Power (HZP) on July 10, with approximately 800 MWD/MTU of exposure accumulated by the core. Subsequent rod symmetry checks and rod worth measurements at HZP confirmed the presence of the tilt and of the increased axial peaking. The reactor was then shut down and the vessel head was removed. An investigation of the fuel and the core internals was initiated by Florida Power and Light Company (FP&L) to determine the cause of the observed anomalies. (1)

The NRC staff has closely monitored the investigation as it progressed from the core internals inspection to an identification of the cause of the anomalies. Additionally, the staff has assessed the generic nature of the anomalies and has reviewed the safety considerations of the repaired St. Lucie 1 core configuration.

The purpose of this safety evaluation report is twofold. First, the bases for the staff's conclusions that the causes of the anomalies have been identified and adequately understood are summarized. This understanding facilitated evaluating the potential for the recurrence of similar anomalies at other reactors. Second, the bases for the staff's conclusions that the repair employed is adequate and returns the core to a safe configuration essentially the same as defined by the approved Final Safety Analysis Report (2) are summarized.

This report also provides the bases for the staff's conclusion that the operation of St. Lucie 1 in the repaired configuration does not constitute a significant hazards consideration. Operation in the repaired configuration neither increases the likelihood of an accident, increases the magnitude of its consequences nor adds the possibility of previously unconsidered accident.

INSPECTION AND TESTS

Inspection at St. Lucie 1

A thorough inspection of the core and vessel internals revealed only one type of indication which could account for the observed anomalies. This indication was the visual observation of blisters on a significant number of the burnable poison rods. The blisters were later confirmed to be caused by hydriding of the poison rod cladding. The visual inspections were limited to that portion of the cladding on those poison rods (4 per bundle) which were positioned in the second row of rods within each bundle. Approximately 15-20% of an individual rod surface could be observed. A total of 92 poison rods from 23 fuel bundles were visually examined in this manner and 26 rods exhibited some stage of hydride blisters. Hydride blisters in general were found in the axial mid-section of the poison rods. Due to the small percentage of cladding surface observable, it was assumed that the hydriding was widespread throughout all those fuel bundles containing poison rods. On this basis, FP&L decided to remove all of the initial-core poison rods from the fuel bundles and devised a technique for performing this rod withdrawal. Selected poison rods which had been removed were visually inspected and examined by eddy current techniques in the spent fuel pool. The results of these examinations aided in the selection of rods for further examination at the Advanced Reactivity Measurement Facility (ARMF) and Battelle Memorial Institute (BMI).

Tests at the A. R. M. F.

At the time of the shutdown of the St. Lucie reactor the only reasonable explanation for all of the observed anomalies in the reactor power distribution and reactivity states appeared to be an unexpected depletion of the boron -10 isotope (B^{10}) from the burnable poison rods as a result of either poor calculations of neutronic processes or a physical loss (or redistribution) of boron from the rods. When the reactor was opened and perforations in the poison rods were observed, the unexpected depletion of B^{10} became even more likely since it was known that the B_4C can react with high temperature water forming a soluble product and be transported. Florida Power and Light Company then decided to measure the B^{10} content and location in some defected pins to estimate quantitatively the effects on power distribution and reactivity for comparison with the observations. The distribution of B^{10} was determined by direct measurement at the Advanced Reactivity Measurement Facility (ARMF) at Idaho.

The ARMF is a small water reactor with a vertical center hole for experiments. In this experiment the poison pins were run through a dry tube which was lined with B^{10} except for a small window providing an opening of a few inches for neutrons. The reactivity of the system with a segment of the poison rod in this window constitutes a measure of the B^{10} content of the rod in that segment. The system is calibrated with known boron content rods. The accuracy is on the order of one or two percent with a spatial resolution of one or two inches.

During the time frame of this review four St. Lucie rods plus an unirradiated rod were examined in the ARMF. From examinations at St. Lucie one of the four burnable poison rods was known to be intact and three had a perforation. The intact rod showed no anomalies in boron content along the length. The B^{10} was depleted by the amount predicted for the reactor burnup. All three perforated rods showed the same general perturbations in their B^{10} content at the same general location. It was determined that the B^{10} content of the perforated rods was lower than the value predicted by burnup calculations. Also, it was determined that redistribution of some B^{10} had occurred within some perforated rods. The integrated boron missing beyond normal depletion was approximately the quantity predicted to produce the observed anomalies.

Although the testing sample was small, the results seem to demonstrate quite conclusively that intact rods show normal expected burnup depletion and perforated rods show extensive boron removal in the region near the perforation.

Tests at Windsor

The poison pellet moisture qualification at the time the St. Lucie 1 poison rods were fabricated apparently allowed moisture levels above that sufficient to produce primary hydriding.

The exact moisture content of poison pellets initially loaded into the St. Lucie 1 core remains unknown since there was no moisture sampling plan and there are no archival poison rods.

Also at Windsor the geometric stability of the poison pellets in the reactor coolant was tested. Irradiated Al_2O_3 - B_4C pellets from St. Lucie 1 (∿10% depleted) were returned to Windsor for testing in an aqueous environment. These pellets maintained their structural integrity although some loss of boron occurred.

Inspection & Tests at the Battelle Memorial Institute (BMI)

Several of the removed poison rods from the St. Lucie 1 assemblies were examined in the Hot Cell at BMI. Some of these rods were considered sound (non-leaker) and others were perforated (leakers). The integrity of the unfailed rods was verified by puncturing and collecting the internal gas content. In these rods, suspect areas for further destructive examination were identified by profilometry and eddy current techniques. The cross-section metallography at these locations clearly showed the presence of internal hydride formation. This evidence along with the other observations at the BMI hot-cells led to the conclusion that the cladding failure mechanism was unequivocally primary hydriding due to hydrogenous impurities, presumably moisture.(3)

In a leaker rod, the metallographic details at the location of the leak showed a through-wall hydride penetration and a absence of B_4C particles that were initially dispersed in the Al_2O_3 matrix. The reaction of B_4C at this location with subsequent B_{10} removal from the rod is consistent with the observed reactivity change.

Tests at KWO

Poison pellets contained in a test rod irradiated for KWO (a German facility) which were fabricated by the same vendor as the St. Lucie 1 pellets and which had been fully irradiated (100% depletion) were also tested in an autoclave. The results from these tests after greater than 300 hours were similar to those observed at Windsor, i.e. the structural integrity of the pellets remained yet there was some boron removal. The structural integrity of these irradiated pellets and those tested at Windsor differ from those reported by BMI in 1963.(4) However, the physical characteristics of the BMI material were sufficiently different from the St. Lucie 1 pellets that the difference in the corrosion behavior is expected.

Subsequently, the staff has met with a burnable-poison-pellet manufacturer to discuss the expected behavior of the poison pellets in reactors. A survey of the available literature gives evidence that normally fabricated poison pellets analogous to those fabricated for St. Lucie 1 are geometrically stable when exposed to the reactor coolant.(5)

INTERPRETATION OF ANOMALIES AND CAUSES

Three sets of relevant observations were made on the characteristics of the St. Lucie 1 core and internals: (1) in the course of the normal startup tests, (2) the period during which the anomalies were noted, and (3) during subsequent investigations of the causes. The identified cause was found to provide a consistent explanation for the observed anomalies.

The following observations were made both in the course of the normal startup tests and during the period in which the anomalies were noted.

1. The initial startup characteristics of the reactor (as determined in the low-power and early medium-power physics tests) were all normal and remained within the ranges expected by comparison with both calculated values and similar reactors.

2. As the startup program progressed beyond a burnup of about 400 MWD/MTU and in the 50 to 80% power range the anomalies increased in magnitude with burnup until the reactor was shutdown with a burnup of 800 MWD/MTU. It should be noted that three other C. E. reactors of similar design, physical characteristics, and operations had passed through the same burnup stages and beyond without showing these anomalies.

3. The power distribution azimuthal tilt increased from initial values less than 2% (normal)to about 4%. This tilt distribution was somewhat diffused, i.e., not indicative of a localized disturbance.

4. There was a reactivity gain (as noted from boron levels) of about 0.4%Δk above that seen in other reactors after accounting for axial peaking.
5. The axial power distribution changed. In the normal unrodded case, the average peak increased from an expected value of about 1.3 to about 1.5. The distribution was axially symmetric, i.e., peaked at the core midplane. Although the increase occurred in fuel rods located throughout the entire core, there was some variation in magnitude. That region of the core showing the higher azimuthal tilt also showed the greater axial peaking.
6. There was some indication of "stiffness" in the axial distribution since insertion of the control rod bank did not increase the peaking as much as anticipated.

7. Several tests were made at zero power prior to opening the reactor. The results are that the control rod symmetry check indicated that azimuthal reactivity variations were present in the core which were not there initially (the variation in control rod worth of symmetrical rods increased by several cents). These changes followed the tilt pattern, i.e., reactivity worths shifted, indicating a shift of reactivity toward the core midplane. The axial power distribution is not as accurately measured at zero power, but the indication was that the axial peak was even larger than at power and with a shape indicative of a midplane reactivity gain.

In summary the anomalies at the time of shutdown appeared to result from changes in the system occurring over an extended time frame during initial reactor operations. These changes caused a reactivity increase over the core, but preferentially toward the axial midplane, and somewhat random azimuthal XY variations. The changes existed at zero power showing that thermal feedback was not an ingredient.

A number of mechanisms for the anomalies were postulated and investigated. They included:

- Flow maldistribution or blockage
- Temperature maldistribution
- Burnable poison or fuel initial misloading
- Fuel, poison rod or bundle bowing
- Crud deposits and/or removal
- Fuel temperature effects
- Dropped control rods
- Burnable poison loss from either
  - poor predictive calculations or
  - physical redistribution

While such mechanisms served as a basis for examinations after opening the reactor, except for the burnable poison possibility, none appeared to offer a suitable explanation, either singly or in combinations. For example, those involving feedback such as flow or temperature could not explain zero power effects. Misloading or initial boron crud deposits (which were subsequently removed) could not explain initial freedom from the anomalies.

Most of the postulated causes could not have produced the observed axial peaking.

During the subsequent investigation of the cause, the reactor was opened and many observations were made. No evidence was seen of blockage in the system. There was no sign of misloading or misaligned control rods or fingers, no rod or bundle bowing or growth anomaly and no sign of crud deposits. These observations further confirmed that such physical changes which had been postulated as possibilities were not primarily responsible for the observed anomalies, nor were any new possibilities added as a result of the observations.

The one primary change observed was the perforations in the burnable poison rod cladding. The design function of the burnable poison rods is to control reactivity in the initial core to assure a more negative moderator temperature coefficient. The observed perforations led to the extensive examinations described above. The observations of these perforations and the concommitant distribution of boron are briefly summarized as follows:

1. A large fraction of the rods, on the order of two-thirds, have perforations. This is true for both the observations from outside the fuel bundle and among the individual poison rods removed from the bundles. About one half of the fuel bundles contain twelve burnable poison rods in each bundle.
2. The perforations were located preferentially toward the center of the rod which corresponds to the core midplane.

3. There are more perforated rods per bundle in bundles associated with high axial peaking factors (e.g. 1.64) than in bundles with a lower axial peaking (e.g. 1.48), i.e. a correlation exists between the number of perforated poison rods power bundle and axial peaking in that bundle.
4. The A.R.M.F. results show that non-perforated rods have the content and distribution of boron expected (predicted by CE calculations for the 800 MWD/MTU burnup).
5. Perforated rods have boron missing and redistributed. The most boron was missing near the perforation and from that segment of the rod immediately below the perforation. There was more boron toward the bottom of the rods than was initially there at fabrication.

Some of these observations are based on small sample of data. For example, only three perforated rods were examined in the ARMF at the time of this review. Three rods would not provide adequate basis for quantitative predictive calculations for future operations. However, that is not their purpose. The observations provide a plausible base and consistent explanation of the observed reactor anomalies.

In order to relate these observations on the poison rod perforations to the reactor anomalies Combustion Engineering (CE) has provided the following estimates:

1. Calculation of the expected boron depletion in poison rods: Comparison of the result with the boron content of the intact rod gives good agreement and indicates that the calculations of poison burnup are satisfactory.
2. Integration of the boron content of perforated rods: This indicated the approximate amount of boron that is missing in the perforated rods.
3. Calculation of the reactivity effects of removing and redistributing boron: These calculations indicated that the amount of removal required to match the observed reactivity gain in the reactor is consistent with the observed missing boron when considering the reactivity effects of the increased central axial power peaking.
4. Calculations of axial distributions both at power and at zero power conditions with various axial distributions of boron, including the average of those distributions found in the perforated ARMF test rods: These calculations indicated that the boron distribution in the St. Lucie 1 core likely produced the observed axial power distributions. That is, a large removal of boron near the center of the reactor, even with some axial asymmetry in the resulting boron loading, produces an axially symmetric power distribution. The amount of boron removal estimated from the number of perforated rods observed is consistent with the observed peaking both at part power and at zero power.

5. Estimation of the correlation between axial peaking in a bundle and the number of perforated rods per bundle, and an estimation of the distribution of perforated poison rods over the core: These estimates are based on the observed variations in axial peaking and the number of perforated rods in the four bundles from which the poison rods were removed and eddy current tested. These estimates indicated that about ten percent more rods were perforated in the quadrant having increased power production, i.e., positive tilt.
6. Calculation of XY power distribution with boron missing from some pins: These calculations have indicated that azimuthal power tilts of the magnitudes observed can be produced with XY boron variations approximately equivalent to the additional observed rod perforations with the observed amount of missing boron existing in one quadrant.

The conclusions from the observations and calculations on perforated poison rods may be summarized as follows:

1. There are a large number of perforated poison rods in each quadrant.
2. There are variations in perforated rod density over the core. It appears that there are on the order of ten percent more perforated rods in one quadrant than on a core average.
3. Perforations are mostly toward the core axial midplane.
4. Boron is missing predominately near the core midplane.
5. The total boron measured to be missing was included in physics calculations which gave results consistent with the observations.

6. The missing (or redistributed) boron supplies a spatially varying reactivity gain which causes:
   (a) a total reactivity gain
   (b) a relative axial reactivity gain near the midplane
   (c) an axial power peaking increase both at part power and at zero power
   (d) an axially symmetric power peak (peak at midplane)
   (e) a "stiff" axial peak not easily perturbed by axial asymmetries, e.g. control banks.
   (f) an azimuthal reactivity variation at zero power.
7. Qualitatively these changes explain each and every observed anomaly in the core.
8. Quantitatively they are consistent with the observations, though the statistical samples are small.

The speculation that the abnormal boron loss might be explained by poor calculation is contrary to the ARMF results on intact rods. Furthermore, a recent review of CE methods in this area has indicated no evidence for such a cause.[6] The staff considers the CE calculations methods to be sufficiently consistent with other methods employed within the industry and concludes that the CE methods are not the significant contributor to the source of the St. Lucie 1 anomalies.

It appears to be reasonable to conclude that anomalies can be predominantly explained by the poison rod perforations and that no other changes are significant in explaining the phenomena or were found in examinations of the reactor or its operating history, and therefore the replacement of the poison rods serves as a basis for the repair of the St. Lucie anomalous behavior.

EVALUATION OF THE REPAIR

Description of the Repair

The repair of those fuel bundles which contain poison rods was accomplished by removing portions of the holddown plate and upper end fitting flow plate which were immediately above the poison rods, installing new poison rods and installing a retention assembly to provide positive rod retention capability where portions of the flow plate were removed to gain access to the poison rod locations. This repair resulted in fuel bundles which are not significantly changed from the original design.[1,2]

The retention assembly provides an upper stop to limit rod axial motion in the unlikely event of a LOCA. The retention assembly is a single component made from a set of four rectangular cross section, austenitic stainless steel bars which are shaped to replace the four sections of the original flow plate web which were removed during the first step of the repair. The four bars are assembled in a square structure and connected at the corners by welding to stainless steel tubes. The assembled structure is positioned on the flow plate by

inserting the four corner tubes into the four drilled corner holes in the flow plate; the tubes are deformed against the flow plate, and the underside of each tube is flared to secure the retention assembly into position.

In order to ensure that the retention assembly is held securely against flow forces, the underside of the retention assembly bars are relieved slightly toward the corners so that the act of pressing down and securing the corners effectively preloads the entire retention assembly.

The replacement burnable poison rods are of essentially the same design as the rods described in the FSAR except for minor changes which minimize the probability for recurrence of the mechanism which caused failures among the original rods. The description of the poison rod changes is as follows:

1. The nominal outside diameter of the replacement cladding is slightly larger than the nominal OD of the original rods.

2. The lower end cap of the burnable poison rod, which engages the retention grid at the lower end fitting of the fuel bundle, has been slightly modified by increasing the width of the lead-in taper and the diameter of the retention surface. These changes have been made to facilitate insertion of the new rods and to assure that the rod is held down in the retention grid allowing growth to occur at the upper end of the rod.

Tests of this design have been performed and show that the modified end cap design enters and positively engages with the retention grid. Testing also showed that forces far in excess of the predicted hydraulic uplift force are required to remove the poison rod from the retention grid. Therefore, the positive prevention of upward movement of the replacement rods is afforded by the retention grid and modified lower end cap.

3. The active poison loadings in the replacement rods, specified in terms of grams of Boron-10 per inch of pellet stack, is set equivalent to the as-built loading in the original rods.

In summary, the reworked fuel bundles incorporate the following modifications relative to the original design:

1. The original upper end fitting flow plate has been modified by removal of those portions of the web which were immediately above the poison rods.
2. The poison rods have been replaced by a design incorporating a lower allowable moisture content, a slightly larger clad O.D. to ensure adequate initial support from spacer grids, and a slightly modified lower end cap to ensure that the replacement rods properly engage the retention grid.
3. A retention assembly has been installed on the upper flow plate to provide a positive means of limiting axial movement of the poison rods.

Mechanical and Thermal Hydraulic Effects

The retention assembly has the following design features.

1. The retention assembly design does not have a significant effect on the flow resistance of the upper end fitting flow plate. It does not give rise to flow patterns which adversely effect other reactor or fuel components.
2. The retention assembly was designed such that the forces associated with installation do not produce distortion of the existing upper end fitting.
3. The retention assembly will be compatible with the fuel handling equipment and does not obscure the existing fuel bundle identification features located on the upper surface of the existing flow plate.
4. The retention assembly was designed to have sufficient preload to prevent unacceptable vibratory motion under the maximum predicted vertical and lateral flow forces.
5. The presence of the retention assembly has been estimated analytically to result in only an insignificantly small increase (0.8%) in the overall pressure drop loss coefficient for the assembly.
6. Since both the existing upper end fitting and the retention assembly are made from austenitic stainless steel, no adverse thermal distortions during thermal cycling are expected. Nevertheless, thermal cycling tests have been conducted to substantiate the acceptability of the retention assembly to upper fitting joints.

The modifications of the upper end fittings discussed above do not affect the performance characteristics of the original design. This conclusion is based on the following considerations:

1. The modifications produce only a small change in the flow area of the upper end fitting. Because the upper end fitting is downstream of the core and only contributes approximately seven percent of the total flow resistance of the fuel assembly, the effect on overall core flow distribution, and on the flow and hydraulic forces acting on any specific fuel assembly, will be insignificant. Flow testing of fuel assemblies with upper end fittings having flow areas which bound the modified upper end fitting have been conducted. These tests confirm the acceptability of the hydraulic forces acting on the fuel assembly.
2. The load path through which the spring preload is transferred from the fuel assembly to the upper guide structure does not rely on that portion of the upper flow plate which will be removed. Therefore, there is no effect on the function of this component.
3. The removal of one of the ligaments in each quadrant of the flow plate will have a negligible reduction in the overall rigidity of the upper flow plate. Since both the axial and lateral deflection characteristics of the fuel assembly are almost entirely dependent on the combination of guide tubes and spacer grids, the modifications to the upper end fitting do not produce any discernible alterations in these characteristics.

The acceptability of the 0.4445 inch O.D. (0.0045 inches greater than the original rods) has been demonstrated by calculations which show that an increase of 9% in the clad wall thickness has no significant effect on the operating temperatures in the poison pellets or on flow in the adjacent coolant channel. Also, tests have been conducted which show that the slightly larger diameter poison rods do not give rise to any unusually high installation forces when inserted in spacer grid cells whose preset interference is comparable to that expected in the partially burned fuel assemblies.

The acceptability of the modified lower end cap has been demonstrated by tests which show that the end cap, once engaged, is restrained by the retention grid to a force exceeding the maximum predicted upward force which could act on a poison rod.

Combustion Engineering has performed tests and analyses to demonstrate that the retention assembly provides a positive means of limiting the axial relocation of fuel rods or poison rods. They have tested the capability to withstand impact loads on both the flared tubes and ligament portions of the retention assemblies. The flared tubes were tested by measuring the load applied to push out the flared tubes from the bottom. The applied load greatly exceeded the accident loads which a lifted corner shim rod could apply to the flared tube during postulated accidents. The ligament portions were tested by measuring their load-deflection properties by pushing up on the ligament from the fuel side of the flow plate.

Two conservatisms for the effects of irradiation and temperature were then applied to the results before being compared with the calculated loads.

The tested load bearing capability was reduced by the effect of temperature while the available ductility was reduced by the effect of the end-of-service-life total fluence. The conservatively adjusted results were integrated to determine the energy absorption properties as a function of deflection. The energy absorption properties of the ligament portions of the retention assembly adequately exceed the maximum impacting energy calculated to occur during the worst postulated LOCA.

Combustion Engineering has also performed vibration tests to demonstrate the ability of the preloading during final assembly to prevent unacceptable vibration. The tests include the application of strain gages at sensitive locations on the ligaments of the retention assembly. The instrumented flow plate was then subjected to flow rates in excess of service flows. The results demonstrated the absence of unacceptable vibration of the installed retention assembly.

The hardware and procedures for implementing the repair of the fuel assemblies at St. Lucie 1 has been the subject of NRC review primarily through the Office of Inspection and

Enforcement. They have, almost continuously, monitored the out-reactor development tests and the on-site repairs. They have coordinated their efforts with the progress of the NRR review to assure that the critical elements are being verified in the field. The Office of Inspection and Enforcement has concluded that the repair was accomplished in a manner which was free from adverse effects on the repaired fuel bundles.[7]

Physics Effects

Replacement of the poison rods will return the St. Lucie fuel to approximately the original design. Behavior of the repaired fuel will be somewhat different from that predicted for the original loading because unburned poison rods will be present in fuel which has accumulated approximately 800 MWD/MTU of exposure. The primary effects of these undepleted poison rods are a lower boron concentration in the cooling water and a modification of the power distribution. In addition, some perturbations of power distribution (tilt, axial and radial peaking factors, and local inhomogeneities) due to the off-design exposure of the fuel rods during the initial plant operation will be experienced. These "burned in" perturbations will be low in magnitude because of the low exposure (800 MWD/MTU) involved.

The replacement rods are approximately 12% overloaded compared with the B^{10} content resulting from normal depletion. Therefore, less boron will be necessary in the coolant and the moderator temperature coefficient of reactivity will be more

negative. Local peaking within fuel bundles will also be affected. However, the licensee has automatically taken these local effects into account by representing the poison rods explicitly in his reanalysis.

The radial peaking factor will be reduced at the restart of the plant because of the greater fuel depletion of the central bundles. However, radial peaking increases with burnup and the maximum value of the radial peaking factor (which occurs at about 6000 MWD/MTU) is 1.325 (hottest rod to average rod) vs. 1.316 for the original design depletion. This increase is still bounded by the value of 1.33 used in the safety analysis. Technical Specifications require the radial peaking factor to be measured at least once every 31 days, which provides a method to confirm that F_r will remain within its limit.

Axial peaking will also be reduced by poison rod replacement. The axial peaking factor for the repaired fuel is initially 1.24 and decreases with exposure. It is always bounded by the maximum value (1.28) calculated for the original design depletion. The calculational methods used by the licensee and Combustion Engineering have been examined by the staff[6], and found acceptable.

The licensee has examined the input parameters to the FSAR accident and transient analyses. All values calculated for the repaired core remain bounded by the FSAR values. Therefore, the FSAR analyses continue to be applicable.

POST-REPAIR SURVEILLANCE

Physics Startup Tests

Power distribution and reactivity anomalies at St. Lucie 1 interrupted the initial power ascension and startup testing program. The program is intended to confirm the design bases, to demonstrate that the reactor is capable of withstanding anticipated transients and postulated accidents, and to provide general quality assurance on the core configuration. Since the St. Lucie 1 initial core was dismantled, repaired and reconstituted, and since the repaired core will behave slightly differently from the initial core, certain parts of the power ascension and startup testing program will be repeated.

A startup testing program for the repaired core was proposed by the licensee and is similar to that normally employed for a reload core. The Control Element Assembly (CEA) group withdrawal sequence is by design A-B-1-2-3-4-5-6-7 where CEA group 7 is the last group withdrawn and the first group inserted. Individual CEA group worths will be measured for CEA groups 7, 6, 5, 4, 3, and 2. The measurement of the reactivity worth for regulating CEA group 1 and shutdown CEA group B will be made only if the previously measured worths of CEA groups 7, 6, 5, 4, 3 and 2 vary from the design calculations by more than the acceptance criteria. The acceptance criteria are that the measured individual group worth varies from the calculated worth by less than either 10% or 0.1 $\Delta k/k$ (whichever limit is larger) <u>and</u> that the measured cumulative worth of CEA groups 7 through 2 is within 10% of the calculated cumulative worth.

The measurements of worths of the shutdown CEA group A, the stuck CEA, the ejected CEA and the dropped CEA will not be made. All of these measurements were performed during the initial low power startup testing program at St. Lucie 1.

For the remainder of fuel cycle 1 at St. Lucie, the calculated shutdown worth is considerably greater than the required shutdown worth. Measuring the worth of CEA groups 7, 6, 5, 4, 3 and 2 empirically verifies over 80 percent of the required shutdown worth while these six CEA groups represent less than half of the total shutdown capability in the core. Since the measured CEA group worths must remain within 10% of the calculated design worths, the staff concludes that the startup testing will confirm the design bases and demonstrate that the St. Lucie 1 core is capable of withstanding anticipated transients and postulated accidents.

At-power Monitoring

Our evaluation of the burnable poison rod problem indicated that prior to substantial burnout of the boron, power distribution anomalies are possible which would not be readily and rapidly detected if the core power distribution was being determined by the excore detector monitoring system. The Technical Specifications do allow operations with power distribution determined by use of the excore monitoring system. However, the excore mode of power distribution monitoring does not measure the axial peaking factor directly, but instead measures imbalance between power in the top and bottom halves of the core. The type of anomaly experienced was an increase

in the axial peaking without moving the centroid of the axial power distribution, therefore the excore mode of power distribution monitoring would not detect this anomaly. Such an increase in the axial peaking factor might not be detected for up to 31 days, when an incore power map is made as required by the Technical Specifications. Such a long period could result in peaking factor increases which are not acceptable. To preclude undetected peaking, the licensee has committed to the following additional monitoring if the excore mode (Technical Specification 4.2.1.3) is used and the core burnup is less than 10,000 MWD/MTU. After 10,000 MWD/MTU, the burnable poison will be sufficiently depleted that redistribution and loss of the remaining boron will result in only minor axial and radial power peaking which are within the original design criteria.

1. If the excore mode (Technical Specification 4.2.1.3) is used then the axial peaking factor will be monitored manually in the maximum possible number of fuel assemblies via the fixed incore detectors. The surveillance period will be selected such that the axial peak will increase less than 3% between measurements based on the average rate from the most recent measurements but is not to exceed 7 operating days. A 3% per day growth rate will be assumed until measurements are available. All measured growth rates will be doubled to account for measurement and projection uncertainties.
2. If the excore mode is used, then core power maps will be taken and compared with predicted distributions every seven operating days, rather than every 31 days as previously required.

The staff has reviewed the proposed additional monitoring detailed above and finds it acceptable assurance that any recurrence of the anomalies will be readily detected providing the action to be taken, if axial peaking were to exceed the design basis value, is explicitly identified. The staff has therefore incorporated the proposed additional monitoring into the license and added a requirement to assure that the thermal power would be appropriately reduced if axial peaking in excess of the design basis value is observed and that such peaking would be reported to the Commission as a reportable occurrence requiring prompt notification. These license conditions have been discussed with FPL and agreed to by their staff.

Subsequent Inspections

The addition of the retention assembly to the flow plate is a slight design modification but the primary features of the fuel bundles remain essentially the same as the original fuel bundles. Although Combustion Engineering has analyzed and tested the retention assembly outreactor, the Florida Power and Light Company has committed[1] to visually inspect a representative number of repaired fuel bundles at the first refueling. The visual inspection will provide the final confirmation that the retention assemblies are performing as expected.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact.

Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

The staff has reviewed the investigation of the cause of the power distribution and reactivity anomalies in St. Lucie 1. All crucial ingredients for an explanation of the anomalies have been pursued. The investigation has identified the most probable cause that led to perforation in a large fraction of the poison rods at St. Lucie 1. The individual effects of these poison rod perforations have been measured to characterize the resulting boron loss and redistribution. The measured effects on individual rods have been evaluated in conjunction with other crucial observations from the investigative program to predict the observed anomalies. Based upon the fact that the combination of observations explains the power distribution and reactivity anomalies, we have concluded that the predominant cause of the anomalies has been identified and is adequately understood. This understanding has led us to conclude that the predominant cause of the anomalies is unique to the St. Lucie 1 initial core. Although the cause of the anomalies is possible at other reactors the power monitoring techniques generally employed by licensees can readily detect the beginnings of such anomalies.

We have reviewed the development of the repair configuration and the implementation at St. Lucie 1. We have considered the effects of the repaired configuration upon the mechanical, thermal-hydraulic and physics design functions initially approved.(2) We conclude that the repair returns the core to a configuration well within the configuration approved initially and that the additional monitoring required would provide acceptable assurance of early detection of anomalies in the unlikely event that anomalies were to reoccur while power distribution was being monitored by the excore detector monitoring system.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Date: December 3, 1976

REFERENCES

1. Letter from R. E. Uhrig, Florida Power & Light, to D. L. Ziemann, NRR, subject: St. Lucie 1- Reference L-76-365, dated October 20, 1976 with attached report CEN-38(F)-P "Repair Report."

2. Safety Evaluation Report of the St. Lucie Plant Unit No. 1, Docket No. 50-335, dated November 8, 1974 and Supplement No. 2, dated March 1, 1976.

3. M. D. Houston to P. S. Check,"Visit to Battelle Hot Cell", dated October 12, 1976.

4. Burian, Fromm & Gates, "Effects of High Boron Burnups on B_4C and Zr B_2 dispersion in Al_2O_3 and Zircaloy-2", BMI-1627, April 24, 1963.

5. M. Tokar to P. S. Check, "Meeting Summary; Irradiation Behavior of Al_2O_3 - B_4C", dated October 27, 1976.

6. D. Fieno to P. S. Check, "Summary of a meeting held with Combustion Engineering to Discuss Lumped Burnable Poison Physics Calculations," dated October 7, 1976.

7. Letter to K. V. Seyfrit, Reactor Technical Assistance Branch, from D. C. Kirkpatrick, subj: "REPORT OF TRIP TO WINDSOR, CONNECTICUT TO WITNESS COMBUSTION ENGINEERING DEMONSTRATION OF ST. LUCIE FUEL MODIFICATION PROCEDURE", October 13, 1976.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 10 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the license for operation of the St. Lucie Plant Unit No. 1 (the facility) located in St. Lucie County, Florida. The amendment is effective as of its date of issuance.

The amendment (1) authorized operation with repaired fuel assemblies and (2) required additional monitoring during power operation if the core power distribution is being determined by use of the excore detector monitoring system when the core burnup is less than 10,000 megawatt days per metric ton of uranium.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the licensee's filings dated October 25, 1976 and November 18, 1976, (2) Amendment No. 10 to License No. DPR-67, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida 33450.

A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 3rd day of December, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

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