7/8/79

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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Office of the Secretary
Docketing & Service
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ATOMIC SAFETY AND LICENSING APPEAL BOARD

Michael C. Farrar, Chairman
Richard S. Salzman
Dr. W. Reed Johnson

In the Matter of	)	
	)	
FLORIDA POWER & LIGHT COMPANY	)	Docket No. 50-389
(St. Lucie Nuclear Power Plant,	)	
Unit No. 2)	)	

INTERVENORS RESPONSE IN OPPOSITION
AND SUGGESTION FOR HEARING

On Friday, June 22, 1979 the Staff filed their motion for a delay until September 21, 1979 to respond to questions posed by the Appeal Board on the issues of the Florida Power and Light Company electrical grid stability (off-site power) and emergency diesel generators (on-site power) due to the unavailability of certain Staff experts who are assigned to the Three Mile Island Nuclear Plant Unit 2 case. (NRC Docket No. 50-320). Intervenors oppose the motion. While the St. Lucie Intervenors, in an attempt to be more than reasonable, did not oppose a previous Staff requests for extentions of time, they now recognize that the Staff responses to

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important questions of serious health and safety significance in the St. Lucie 2 case are long overdue and have accordingly delayed the Appeal Board's hearing process and consideration of these serious safety issues. Intervenors recognize the validity of the Staff responsibilities at the Three Mile Island Nuclear Accident Investigation and do not seek to preempt those efforts. However, it is also the Intervenors view that the priority of consideration of safety concerns at the Hutchinson Island site are on a par with those of Three Mile Island and that the health and safety of residents of Pennsylvania cannot under the law be given a greater priority than that of the citizens of South Florida. Yet, the Staff is allowing construction of the Hutchinson Island Nuclear Plant without full and adequate health and safety considerations while it concentrates its efforts on the aftermath of Three Mile Island. As it presently exists, the record of the St. Lucie 2 case is insufficient with respect to the unanswered questions on the grid stability and emergency power issues. The blame for the delays and insufficiency appears to lie, to a large extent, with the Staff. Therefore, Intervenors suggest that the Appeal Board hold a hearing designed to determine why Staff compliance with Appeal Board requests has not been forthcoming. In ALAB 489, Offshore Power Systems the Appeal Board found:

> "One thing the Board may do is ascertain why the Staff document in question has not been forthcoming."
>
> -- ALAB 489 Offshore Power Systems, (8 NRC 207) Sept. 1978

A hearing would give the Staff the opportunity to establish whether its delays are reasonable:

> "If the Staff can provide adequate assurance that it is acting as quickly and reasonably as the circumstances permit and we emphasize the word reasonably -- then the Board can ask no more and should reschedule the filing date accordingly."
>
> -- ALAB 489, supra.

It may well be that the Staff has good and sufficient reasons for its delays but the reasons cited in the Staff motion of administrative inconvenience is by itself not an adequate ground for granting the Staff's request for additional delay. The Offshore Power case supra, establishes that it is the prerogative of the Board to grant a delay or establish a schedule.

While the Intervenors oppose the Staff's motion for delay on the stated grounds, the Intervenors feel the Staff should be given at hearing every opportunity necessary regarding ascertainment of the nature of the Staff's problems, and the time necessary to resolve them so as to develop a full record. This is necessary because Intervenors further advise the Board that they deem that their participation is appropriate to inquire into the reason and reasonableness of, as well as need for Staff delay, thereby establishing an adequate record in case the Board grants the Staff Motion and Intervenors then based on evidence adduced elect to renew their <u>Motion for Stay</u> previously denied by the Appeal Board in ALAB 537, 9 NRC ____ .

Therefore, in Intervenor's view, the scope of the suggested hearing should include the following considerations:

1. The reasonableness of the cause of the Staff delay.

2. An assessment of adequacy of Staff's review effort, if Staff be required to meet the original filing deadline.

3. Whether valid reasons such as a complete and mature investigation requires an extended time for proper assessment of the issues.

Regarding the Appeal Board's request that Intervenors indicate their degree of participation at the forthcoming hearing, Intervenors intend to participate in the following manner:

1. Begin discovery within 10 days by posing interrogatories to the Florida Power and Light Company based on prepared written testimony recently filed by the Company since that testimony does not adequately answer the questions and concerns of Intervenors.

2. Commence discovery with the Staff after their response to the Appeal Board questions are filed.

3. Attempt to obtain expert witnesses to testify at forthcoming NRC hearing. (Preliminary discussions with Union Concerned Scientist have been held.)

4. Prepare to present Intervenors case, in any event, through cross-examination of FPL and Staff witnesses.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
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FLORIDA POWER & LIGHT COMPANY	)	Docket No. 50-389
	)	
(St. Lucie Nuclear Power Plant,	)	
Unit 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "INTERVENORS RESPONSE IN OPPOSITION AND SUGGESTION FOR HEARING" dated June 29, 1979 in the above-captioned matter, have been served on the following by deposit in the United States mail, first class or air mail, this 29th day of June, 1979:

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June 22, 1979

William D. Paton, Esquire
United States Nuclear
Regulatory Commission
Washington, D. C. 20555

Re: In the Matter of Florida Power & Light Company (St. Lucie Nuclear Power Plant, Unit No. 2) - Docket No. 50-389

Dear Mr. Paton:

Supplementing my letter to you of June 1, 1979, enclosed please find the following:

Testimony of Frederick George Flugger relating to questions A2, B1, B2, B3 and B4 of ALAB-537.

Copies of this testimony have been simultaneously filed with the Board and served on all parties. This testimony, together with the joint testimony of Michel P. Armand, Ernest L. Bivans and Wilfred E. Coe relating to questions A1 and D of ALAB-537, and the testimony of George E. Liebler relating to question C of ALAB-537, served June 1, 1979, constitutes all of the prepared written testimony to be filed by FPL in accordance with ALAB-537.

In a telephone conversation Friday, June 15, 1979, with M. Villar and M. Armand of FPL, Edward J. Fowlkes of FERC requested the following information to allow completion of his review and we are providing that information herewith, with copies filed with the Board and served on all parties:

1. Breaker diagram of FPL electrical system around Midway Substation in 1983.

2. Analysis of the contingency loss of both Midway 240 KV buses.

3. Line outage data for the various lines feeding into Midway Substation.

Very truly yours,

Norman A. Coll

NORMAN A. COLL

NAC/sm
Enclosures

cc: See attached service list.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

In The Matter Of:	)	
FLORIDA POWER & LIGHT COMPANY	)	Docket No. 50-389
(St. Lucie Nuclear Power Plant, Unit 2)	)	
	)	

CERTIFICATE OF SERVICE

I HEREBY CERTIFY that true and correct copies of the foregoing letter dated June 22, 1979, addressed to William D. Paton, Esquire, and the enclosures referred to therein, have been served this 22nd day of June, 1979, on the persons shown on the attached service list by deposit in the United States mail, properly stamped and addressed.

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By Norman A. Coll
NORMAN A. COLL

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

In The Matter Of:)	
FLORIDA POWER & LIGHT COMPANY)	Docket No. 50-389
(St. Lucie Nuclear Power Plant, Unit 2))	

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Testimony of

Frederick George Flugger

Relating to

ASLAB Memorandum and Order of

April 5, 1979, on

Electrical Grid Stability and Emergency Power Systems

(Questions A2, B1, B2, B3, and B4 of ALAB 537)

My name is Frederick George Flugger. I am Supervisor, Plant Licensing, Power Plant Engineering Department for Florida Power and Light Company. My education and professional qualifications appear in the Nuclear Regulatory Commission's record of the St. Lucie Unit 2 (Unit 2) proceeding following Tr. 1310.

The purpose of this testimony is to respond to questions A2, B1, B2, B3, and B4 in Section II of the Appeal Board's Order of April 5, 1979. My affidavit of March 31, 1978 is relevant to the issues raised by the Appeal Board. It is provided as Attachment A and is hereinafter referred to as the Flugger Affidavit.

This testimony demonstrates that the Unit 2 onsite AC power system design is in full compliance with NRC requirements, that the design basis events evaluated in the PSAR provide a proper basis for the design of Unit 2 and that Unit 2, as designed, can acceptably accomodate the postulated loss of all AC event. Before responding to the Appeal Boards's questions, it is appropriate that a few basic considerations be discussed to place the responses in proper perspective.

First, consider the frequency of loss of the electrical grid. FPL nuclear operating history suggests a frequency of outage of about 4×10^{-1} per year for the FPL grid. Although there is little comparative historical data readily available, I believe that the relative difference in reliability between the FPL system, based on its historical data, and other grids associated with the general population of nuclear plants is probably not more than a factor of about 2. It must be noted that as the FPL system evolves during construction of Unit 2 and during its operation, any difference in reliability that may be inferred from FPL's operating history to date will be reduced or eliminated. The testimony in response to Appeal Board questions A1 and D discusses the substantial actions that have been and will be taken to improve the reliability of the FPL grid.

In any event, from a nuclear plant design standpoint, the difference implied by historical data is very small when compared to nuclear plant design reliability levels. The relatively small reliability differences that may be associated with peninsular and nonpeninsular grids will not affect the design of Unit 2 engineered safety features (ESF's).

Second, the probabilities associated with nuclear plant design and operation are not normally precisely quantifiable because of uncertainties that may exist in the data, the depth of experience that comprises the data base, and applicability of the data to a specific design. However, these probabilities can normally be specified fairly accurately within a range of values.

The NUREG-75/087 (reference 1) $10^{-6}/10^{-7}$ guideline value has been and should be associated with events whose consequences are comparable to 10 CFR Part 100 guidelines. The postulated loss of all AC event is not a 10 CFR Part 100 type event. It results in a very slow and tolerable transient that can be accommodated by the existing Unit 2 design.

The time to restore AC power is pivotal to the evaluation of the postulated loss of all AC event. FPL's historical grid data demonstrates that the duration of loss of offsite power is very short-lived. FPL operating experience from January 1972 to present indicates a mean time to restore offsite power to FPL facilities of less than 1/2 hour.

Third, an unprotected loss of coolant accident (LOCA) does not result from the postulated loss of all AC event. There is no failure of the reactor coolant pressure boundary associated with this event. A reactor coolant pump (RCP) seal can only yield very small and acceptable leak rates. (See the response to question B2 *infra*.) Unit 2 has more than adequate capability to remove decay heat, which is necessary to accommodate the postulated loss of all AC event. There is sufficient condensate to provide steam generator makeup for at least 16 hours, the auxiliary feedwater pump is steam driven, auxiliary feedwater pump control and auxiliary feedwater system valves are DC powered, and the steam generators have sufficient inventory to allow the operator about 55 minutes to actuate auxiliary feedwater before steam generator dryout occurs.

With these considerations in mind we can procede with the responses to the Appeal Board's questions.

Question A2

For its part, the first paragraph of GDC-17 appears to establish an unattainable set of conditions for electrical power systems generally. It reads as follows (emphasis added):

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety functions for each system (*assuming the other system is not functioning*) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded *as a result of anticipated operational*

occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

This paragraph requires that an assessment of the sufficiency of the offsite power system start with the assumption that the onsite system is not functioning. That assessment must then consider the effect of "anticipated operational occurrences." But loss of the offsite power system itself may reasonably be considered to be such an occurrence. The parties should, therefore, explain how the St. Lucie Plant can comply with the literal requirements of this paragraph as written. If it cannot, they should attempt to justify the situation in terms of the purpose of the requirement.

Response

The Board's question cites a possible literal interpretation of GDC 17 that contravenes the intent of this design criterion. The intent of GDC 17 is provided in a straightforward manner by the language of proposed GDC 24 and 39 issued for guidance by the Atomic Energy Commission on July 10, 1967 before GDC 17 was adopted in its present form. Their language states:

GDC 24

"In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems."

GDC 39

"Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system."

The intent of these criteria is simply to ensure that an onsite AC source be provided adequate to backup the offsite AC source. This philosophy is embodied in current industry standards. IEEE std. 308-1974 (reference 2) embodies the concept of the "preferred" (offsite) power supply system and the "standby" (onsite) power supply system. The functions of these systems are cited in the standard as follows:

"The preferred power supply shall furnish electric energy for the shutdown of the station and for the operation of emergency systems and engineered safety features."

"The standby power supply shall provide electric energy for the operation of emergency systems and engineered safety features during and following the shutdown of the reactor when the preferred power supply is not available."

RG 1.32 (reference 3) endorses IEEE 308-1974, with a few nonrelevant exceptions, as an adequate basis for complying with GDC 17. In other words, the NRC Staff interprets and requires compliance with GDC 17 in a manner which does not contemplate the literal interpretation suggested by the Appeal Board's question. Unit 2, as designed, complies with the accepted interpretation and intent of GDC 17.

Finally, it is appropriate to note the relationship between 10 CFR 50.36 and Appendix A to 10 CFR Part 50. The latter provides design criteria while the former imposes operational restrictions. The NRC regulations at 10 CFR 50.34 require that a safety analysis be performed to assess the ability of the facility to meet its design objectives. The safety analysis provides the basis for establishing limiting conditions for operation (LCO), which provide the minimum functional capability or performance levels required for safe operation of the facility. 10 CFR 50.36(c)(2). The LCO's become part of the facility's operating license. Therefore, it is pertinent to note that continued Unit 2 operation with both onsite diesel generators inoperable would constitute a violation of Technical Specifications. RG 1.93 (reference 4) states that the limiting condition for operation (LCO) is met "when all the electric power sources required by GDC 17 are available." If both diesels were inoperable the plant's operating license would restrict operation in accordance with RG 1.93 as follows:

"If the available onsite a.c. electric supplies are two less than the LCO, power operation may continue for a period that should not exceed two hours..... If no onsite a.c. supply is restored within the first two hours of continued power operation, the unit should be brought to a cold shutdown state within the next 36 hours."

Thus, the Unit 2 operating license will contain conditions in the form of these Technical Specifications to minimize the risk of exposure to continued plant operation with both diesels inoperable.

Question B1

As we see it, the likelihood of loss of all AC power at St. Lucie may be expressed as the product of two factors: (1) the probability that there will be an offsite power failure involving the FPL network generally or the Midway substation in particular and a resulting loss of station power -- which probability seems based on historical events, to lie in the range 1.0 to 0.1 per year; and (2) the probability that neither of the two onsite AC power systems (diesel generators) will start. The probability that any one diesel generator will fail to start on demand is taken by the staff to be one per hundred demands, i.e., 10^{-2} 25/.

If these figures are accurate, then the combined probability for the "loss of all AC power" scenario is in the range 10^{-4} to 10^{-5} per year. 26/ In this regard, the staff's Standard Review Plan for Nuclear Power Plants sets forth numerical guidelines for determining whether an event "resulting from the presence of hazardous materials or activities in the vicinity of the plant" should be considered in designing the plant (i.e., whether it is a "design basis" event). 27/ Under these guidelines, events with a realistically calculated probability value of at least 10^{-7} per year (or 10^{-6} per year for a conservative calculation) must be so considered.

The "loss of all AC power" sequence is not precisely within the category of events contemplated by the Standard Review Plan. However, its ultimate result -- assuming that power is not timely restored -- is an unprotected loss of coolant accident, the consequences of which are likely to exceed the guidelines of 10 CFR Part 100. We do not understand why this sequence of events (i.e., loss of offsite power combined with failure of diesels to start), which appears to have a probability well above the guideline values, should not be taken into consideration in the design of the plant. 28/ The parties are to address this point, setting forth their reasons for adhering (if they do) to a contrary position.

25/ Fitzpatrick Affidavit of June 12, 1978, p. 4. Also see Regulatory Guide 1.108, Section B.

26/ This conclusion further assumes that the failure of two diesel generators to start would be statistically independent events, an assumption which leads to the lowest likelihood of combined failure, and which might be nonconservative if there exists the potential for common failure modes for the onsite systems.

27/ NUREG 75/087, Section 2.2.3, paragraph II.

28/ We have accepted the Standard Review Plan guideline values as reasonable in another case. Public Service Electric and Gas Company (Hope Creek Units 1 and 2), ALAB - 429, 6 NRC 229, 234 (1977).

Response

The question pertains to two different but complementary nuclear plant design concepts, namely, the frequency of occurrence of an event (events/unit of time) and the reliability of an Engineered Safety Feature (ESF) (failure to function when called upon to do so). Before the question of whether the postulated simultaneous loss of offsite and onsite AC power sources should be included in the design basis can be addressed, it is necessary to discuss the concepts of event frequency and ESF reliability.

Event Frequency

Many types of events have been considered in the design of Unit 2. These may be generally categorized into several major groups as follows:

1. Events of moderate frequency leading to no significant radioactive releases from the facility and no violation of fuel design limits.
2. Infrequent events which have the potential for small radioactive releases from the facility and small amounts of fuel failure.
3. Events of low probability, Design Basis Accidents (DBA), which are required by 10 CFR Part 50 to establish the performance requirements of ESF's and are used in evaluating the ability of the facility to comply with 10 CFR Part 100 guidelines.

Unit 2 design bases are the "specific functions to be performed by a structure, system or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design" (10 CFR 50.2). They are developed by analyzing limiting events, i.e., other events of the type analyzed are less severe. This approach provides reasonable assurance that the facility has adequate capability to accommodate unanalyzed events.

The probability of occurrence of non-design basis initiating events that may produce results more severe than DBA's is considered so small that these events are not incorporated into the plant design. Section 2.2.3 of NUREG-75/087 (reference 1) provides a $10^{-6}/10^{-7}$ guideline for "design basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant." In using this guideline it should be understood that:

1. This guideline is appropriate for events that have a potential for yielding offsite exposures that equal or exceed 10 CFR Part 100 guidelines.
2. There is little experience available to provide a statistical basis for quantifying with precision the probability of occurrence of initiating events which have such low probability. Thus considerable engineering and scientific judgment is involved in determining whether or not a given event should be included in the design basis.
3. If an event which was considered to be outside the design bases did occur, it would not necessarily produce consequences that are catastrophic or exceed 10 CFR Part 100 guidelines. Considerable engineering, design evaluation and operating experience has been accumulated since the first commercial light water reactors went into operation around 1960. This significant experience base has demonstrated that a nuclear

facility has substantial inherent capability to acceptably accomodate a broad spectrum of events.

4. The Unit 2 design philosophy utilized is specifically directed at providing assurance that the likelihood of events with consequences more severe than DBA's is extremely low. The facility is designed, built and operated so that it will, with a high degree of reliability, minimize the likelihood of an accident. Despite the care taken to prevent accidents, the design provides for reliable protection devices and systems designed to detect and cope with transient and off-normal conditions. ESF's provide protection to the public even in the event of the occurrence of severe accidents of low probability, i.e., DBA's. Finally, throughout the facility's lifetime nuclear plant operating experience is continually monitored and assessed by the NRC to determine whether design or procedural modifications are required.

ESF Reliability

Reliability of an ESF is simply the probability of performing its safety function when called upon to do so. Although increased material and component quality level, testing and maintenance will improve reliability, above certain levels substantial cost and testing commitments result in minimal increases. Because of this, the concept of redundancy is employed to achieve acceptable reliability levels in nuclear plant designs. Enormous increases in system reliability can be achieved through redundancy because the overall reliability becomes the product of the reliabilities of the independent systems. The use of the single failure criterion in nuclear plant design is based on the concept of redundancy. The objective of this criterion is to prevent any single failure from preventing the accomplishment of a safety

function. This criterion is imposed by Appendix A to 10 CFR Part 50, and is a fundamental premise upon which all nuclear safety related designs are based.

Loss of offsite electrical AC by itself is protected against by an onsite AC system that employs, in accordance with GDC 17, redundant and independent diesel-generators. The postulated loss of all AC power following the loss of offsite AC violates the single failure criterion in that it requires the failure of both redundant and independent diesel generators. For this reason the sequence of events postulated by the question is not a design basis event. Nevertheless, as discussed below, the postulated loss of all AC event can be accommodated for some period of time.

The appropriate probability for evaluation of the postulated loss of all AC event is the probability during any one year of having loss of all AC power combined with the probability of not restoring AC by time "T" which is given by:

$$P(T) = P(A) \cdot P(B) \cdot P(C) \cdot P(D) \cdot P(E) \cdot P(F)$$

where: $P(T)$ = probability of not restoring AC power by time "T"
$P(A)$ = probability of loss of offsite power
$P(B)$ = probability of loss of first diesel
$P(C)$ = probability of loss of second diesel
$P(D)$ = probability that offsite power is not repaired and returned to service by time "T"
$P(E)$ = probability that first diesel is not repaired and returned to service by time "T"
$P(F)$ = probability that second diesel is not repaired and returned to service by time "T"

The restoration of AC probability terms, $P(D)$, $P(E)$ and $P(F)$ can be developed in a straightforward manner. Let $P(T)$ be the probability that AC is not restored at time "T", $P(T+\Delta T)$ this probability at a finite later time "$T+\Delta T$", and $C \cdot \Delta T$ the repair probability during the time interval ΔT (where C is a constant). Then,

$$P(T+\Delta T) = P(T) \cdot (1-C \cdot \Delta T)$$

which in the limit as ΔT approaches zero is given by:

$$\frac{d\,P(T)}{dT} = -C \cdot P(T)$$

whose solution is:

$$P(T) = e^{-CT}$$

The equation for P(T) can be used to mathematically represent P(D), P(E) and P(F). Examination of historical data allows determination of the time constant "C" for each of these probability terms. An evaluation of FPL system data from 1972 to present indicates that a time constant of 1.6 hr^{-1} is appropriate for P(D). (See the response to question B3 <u>infra</u>.) St. Lucie 1 and Turkey Point diesel generator outage data indicate that a time constant of 0.16 hr^{-1} is appropriate for both P(E) and P(F). (See the response to question B3 <u>infra</u>.)

The probability of loss of offsite power P(A) is obtained in a similar manner. If λ is the grid failure rate (number of failures in a period of time "t", such as 0.1 failures per year), then $e^{-\lambda t}$ is the probability that offsite power will not be lost and the probability that offsite power will be lost can be expressed as $P(A) = 1-e^{-\lambda t}$.

Application of the exponential representation for the probability of restoration of power, a frequency of loss of offsite power of 0.1 per year, a diesel generator failure per demand of 10^{-2}, and time constants of 1.6 and 0.16 hr^{-1} for offsite and onsite power restoration respectively yields:

$$P(T) = 10^{-5} \exp(-1.92T)$$

which can be used to quantify the probability for not returning AC power by time "T" as a function of "T". The results are:

Duration of loss of AC "T" (hours)	Probability of Having a Total Loss of AC Power that Lasts "T" Hours, P(T)
0	1×10^{-5}
1	2×10^{-6}
1.2	1×10^{-6}
2	2×10^{-7}
2.4	1×10^{-7}
3	3×10^{-8}
4	5×10^{-9}

If a loss of offsite AC power event frequency of 1.0 per year were assumed instead of 0.1, then a value of P(T) of 1×10^{-6} will be reached at 2.4 hours, and 1×10^{-7} at 3.6 hours.

The evaluation of historical FPL onsite and offsite failure data demonstrates that the probability of a continued loss of AC power decreases significantly with the duration of the loss. If, as suggested by the question, the $10^{-6}/10^{-7}$ criterion were to be applied to the postulated loss of all AC event, then evaluation of a period exceeding about 1 to 4 hours (rounding off 1.2 and 3.6 hours) is not required since the probability of not restoring AC power within that time period is acceptably low.

For the reasons described in the response to Question B2 below, Unit 2 can be maintained in a safe shutdown condition without AC power for a time period well in excess of the time likely for restoration of AC power.

Question B2

In line with the above discussion, the testimony is to analyze events that would occur between the "loss of all AC power" and the violation of either the fuel design limits or the design conditions of the reactor coolant pressure boundary (or any portion thereof). In particular, the parties should, if possible, reconcile their differing responses to question B.1(b) of our March 10, 1978 order, 29/ or, if not, point up precisely where the disagreements lie.

29/ [References fn 24 reproduced below:]

Applicant suggests that the first safety related failure encountered would be excessive core heating due to the loss of water from the condensate storage tank, and that this would occur about 16 hours after the loss of AC power (Flugger Affidavit of March 31, 1978, p. 3).

The staff's judgment is that the first failure would be that of a primary pump seal at about one hour after the loss of AC power --- resulting in a small loss of coolant accident. (Fitzpatrick Affidavit of June 12, 1978, p. 11).

Response

The Flugger Affidavit filed in response to the Appeal Board's order of March 10, 1978 concluded that there was a sufficient volume of condensate storage to allow the unit to maintain hot standby conditions for at least 16 hours; the spent fuel storage pool would not require makeup for at least 36 hours; and that power would be restored before any unacceptable consequences would occur. The Fitzpatrick Affidavit, which provided the Staff response, concurred with FPL's response, but went on to suggest that a failure of a reactor coolant pump (RCP) seal could potentially occur after one hour as a result of the loss of all AC power. For the reasons set forth below, the difference can be reconciled and Unit 2 can be safely maintained in a hot shutdown condition until AC power is restored.

At the outset, it is necessary to analyze the actual condition of the reactor coolant pumps during the event. Upon loss of AC power the reactor will trip, the RCP's will coast down and stop, and cooling water flow to the RCP seals will cease. This static (pump not running) condition is much less severe than the dynamic (pump running) condition discussed in the Unit 2 PSAR at section 9.2.2.3.1, which provides a basis for concluding that running the pumps for about one hour, without cooling water to the seals, would not result in pump seizure or unacceptable RCP seal failure.

In order to evaluate the static performance of the RCP's under loss of all AC conditions, it is necessary to briefly discuss the seal design and construction. Each RCP is equipped with a seal cartridge, which contains four separate seals. Each of the four seals within the seal cartridge is designed to provide the sealing function against full system pressure. A seal cartridge test fixture is used to fully test the seal cartridge prior to installation on the RCP, and the tested seal cartridge is installed as a unit. All seal components are captured within the seal cartridge assembly. The carbon rings within the seal are held in place by hydraulic force since the higher pressure is on the ring's outside diameter, and spring force in addition to hydraulic force holds the rotating and stationary sealing faces together. Thus the RCP seal design is such that a mechanism for development of an appreciable leakage path within the seal cartridge under static conditions does not exist.

Pressure breakdown devices are installed parallel to the first three seals. Reactor coolant at a rate of 1 gpm passes through these devices such that reactor coolant system pressure is distributed equally across the first three seals, i.e., they normally operate at about 1/3 of their design pressure. The fourth seal is subjected to a nominal backpressure and acts as a vapor barrier/backup seal during normal operation. The RCP controlled bleedoff flow of 1 gpm/pump is directed to the chemical and volume control system.

Under static conditions associated with loss of all AC the temperature of the fluid in the seal cartridge will attain a level above the normal seal cartridge operating temperature due to the interruption of cooling water. The temperature would rise from about 180^{o}F to about 550^{o}F.

If the postulated loss of all AC event occurs, there are two modes of seal operation that may be utilized, namely, secure bleedoff flow or maintain bleedoff flow. If it is assumed that the controlled bleedoff line is closed thereby eliminating the normal 1 gpm flow through the seal cartridge, then only one seal, the fourth, will be functional, sealing against full system pressure. The other seals will see no pressure differential. However, they will automatically take over the sealing function should the fourth seal develop a leak in excess of 1 gpm. The maximum outleakage would not exceed the normal 1 gpm, as flow is restricted to this value by the pressure breakdown devices in parallel with the first three seals. The system pressure would then be distributed equally among the remaining three seals as the pressure breakdown devices become functional. Should the third seal also malfunction, allowing leakage in excess of 1 gpm, the outleakage would increase to 1.2 gpm as only two pressure breakdown devices would remain functional, the third pressure breakdown device being bypassed through the third seal. If the second seal also is assumed to malfunction, the first seal takes over and the leakage increases to 1.7 gpm as only one pressure breakdown device is functional.

If the controlled bleedoff line is not closed off, pressure distribution through the seals is maintained the same as for normal operation and the bleedoff is 1 gpm per pump. Operation in this mode results in a pressure differential across the first three seals of 1/3 of design and only a nominal backpressure across the fourth seal. In case of malfunction of any of the first three seals, pressure is distributed proportionally among the remaining seals with a corresponding increase in bleedoff as stated above. Securing the bleedoff at any time will cause the fourth seal to take over the sealing function.

Even though there are four independent seals per RCP to ensure the maintenance of the sealing function, and each one is designed to seal against full system pressure, there is no reason why any one of these seals would fail in the static condition. The only components affected by the elevated temperature, are the elastomeric gaskets of the seals, namely the "U" cup in the normally rotating part of the seal, and the "O" rings in the stationary seal segment. The "U" cups are totally captured and the "O" rings are backed up by lapped seats which would maintain low leak rates. All other components are metallic or carbon, which are not affected by the elevated temperatures of the system. The elastomeric components are made of Ethylene Propylene or Nitrile, materials which are suitable for long operation at temperatures up to 250^{O}F without change of characteristics. Temperatures above 250^{O}F will affect the physical characteristics of the material, the extent of the effect being a function of temperature, pressure and time. The accepted operating life at 300^{O}F is in excess of 1000 hours.

At the system temperature of 550^{O}F, the elastomeric material, Ethylene Propylene or Nitrile, would be subject to extrusion and hardening, i.e., gradual loss of flexibility and permanent setting in a deflected position. The reactor coolant pump manufacturer has demonstrated, however, the sealing characteristics of the elastomeric material under thermal conditions equivalent to those resulting from the postulated loss of all AC event. Confined "O" rings of the material have been used on several flanged joints of a reactor coolant pump hot test loop, where they have been subjected to temperatures of 550°F for in excess of 100 hours during routine pump acceptance testing. The "O" rings maintained their sealing capability without any problems, and as would be expected, hardening and permanent setting of the "O" rings occurred. Under static conditions sealing would be maintained since (i) the "O" rings are backed up by lapped seats, (ii) the "U" cups are totally captured, and (iii) most of the hardening would occur on cooldown, rather than at the elevated temperature.

In summary, the RCP seal cartridge will maintain its low leakage characteristics for the duration of the static loss of all AC event, and the RCP seals are expected to remain functional for a period of at least 24 hours.

Operation of a reactor coolant pump after restoration of AC power will likely result in higher than normal seal leak rates due to hardening of the elastomeric materials. Thus a natural circulation cooldown to cold shutdown conditions would be preferred since it would not require running of a reactor coolant pump. In this regard, it is important to note that in April 1977 the St. Lucie #1 reactor coolant system was borated and the plant was brought to a cold shutdown without the reactor coolant pumps running, i.e., on natural circulation.

Insofar as maintenance of reactor coolant system temperature and pressure is concerned, following RCP coast down, flow through the reactor coolant system is maintained by natural circulation of a subcooled fluid. Decay heat is rejected to the secondary system through the steam generators. Steam generator safety valves will limit the steam generator secondary side pressure.

The plant operators will start the steam turbine driven auxiliary feedwater pump and will locally open the steam generator atmospheric dump valves to allow the steam generator safety valves to reseat. Steam generator level will be reestablished, at which time the operators will adjust auxiliary feedwater flow to maintain a constant steam generator level. The reactor coolant system will then be stabilized at hot shutdown conditions.

Due to heat loss from the pressurizer, normal reactor coolant system (RCS) leakage e.g., RCP seal bleedoff flow, and secondary side liquid temperature in the steam generators, there will be a gradual and steady decay in RCS pressure and temperature. Since the RCS pressure decays at a higher rate than temperature, the reactor coolant system will eventually, in about 7 hours, reach the saturation condition. Thereafter, decay heat removal will continue by natural circulation of a saturated fluid.

The Flugger Affidavit indicated that about 200,000 gallons of water would be required to maintain hot standby for 16 hours, which is the minimum technical specification limit anticipated for the Unit 2 condensate storage tank. However, the condensate storage tank is normally maintained in excess of the technical specification limit, and has a design capacity of 400,000 gallons. Additionally, there are another 1,800,000 or so gallons (design capacity) of fresh water storage on site at St. Lucie. It is reasonable to conclude that during the

initial 16 hours, when technical specification condensate storage is being consumed, that portable pumps can be made available to replenish Unit 2's condensate storage tank, and that core heat up due to lack of steam generator makeup is not a real-world concern.

There are two safety class DC batteries installed at this facility. Each is an 1800 ampere-hour, on an 8 hour basis, battery, i.e., each can supply 225 amperes continuously for 8 hours. The DC load required for remote manual auxiliary feedwater operation, control room lighting and one channel of instrumentation from the instrument busses is about 100 amperes. Thus, if in say 1/2 hour or so, one battery is secured and parasitic loads are stripped from the on-line battery, there is more than sufficient battery capacity to accommodate the postulated transient. One battery could sustain the DC load for about 13 hours, the second battery could be reconnected to supply the DC load for another 12 hours or so.

It is also pertinent to point out that the Unit 1 diesels can be aligned to supply Unit 2. The PSAR at Figure 8.3-1 shows a tie between the Unit 1 and 2 startup transformers. The tie allows the 4.16 kV busses to be tied between Unit 1 and 2 so that Unit 1 diesels can be aligned to supply AC power to Unit 2 via this tie. There are three breaker cubicles and two breakers. The breakers are normally installed so that the startup transformers supply their respective unit's AC busses, and the tie between units is physically open, i.e., the breaker is not installed. Loads would have to be stripped from the Unit 1 and Unit 2 busses and the breaker in the cubicle from the Unit 2 startup transformer must be removed and installed in the 4.16 kV switchgear tie. The sequence of events has been reviewed, and it has been determined that it would take two men about one hour to align a Unit 1 diesel to Unit 2.

In summary, under the postulated loss of all AC event, fuel and reactor coolant pressure boundary limits will not be exceeded during the probable time necessary to restore AC power. There is no basis for assuming that all four redundant seals on a RCP will lose function during the postulated loss of all AC event, and no LOCA would result from this postulated event. Thus, 10 CFR Part 100 guidelines are not applicable to this event.

Question B3

The testimony should contain a discussion, supported by such data as is available, related to the time that might be required to start a diesel generator assuming it failed to respond to the initial, auto-start signal.

Response

Should a diesel generator at a nuclear plant fail to start the unit's technical specifications would require that the second diesel and offsite AC circuits be verified operable and that power operation may continue for a period not to exceed 72 hours (reference 4). Thus, if the remaining AC power sources are operable, there is no undue time-pressure constraint to return the diesel to service, which would exist if all AC power were lost. Accordingly, any evaluation of the time to return a diesel to service based on historical data would likely yield a conservative estimate of the time to return a diesel generator to service.

The concept of diesel reliability should also be placed in proper perspective. A 10^{-2} probability of demand with a confidence level of 95% was demonstrated by a 300 start shop test program for a Unit 1 diesel (see Unit 1 FSAR section 8.3.1.3). A successful attempt occurred if the diesel performed the sequence: fast start, automatically bringing the set to full speed and voltage, immediately loading the generator to 60% of continuous rating, and maintaining the 60% load for 5 minutes. A failure in any portion of the sequence was considered a failure per demand. This sequence is based on LOCA generated ESF requirements, which place exacting quick start design requirements upon the diesel generators.

Diesel generator experience at St. Lucie Unit No. 1 has been reflected in the Unit 2 design. There have been seven failure to start incidents at St. Lucie of which only two could be categorized major maintenance items. These two events were associated with turbocharger malfunctions, which involved repair durations of about 60 hours and 173 hours. Four of the remaining five events were corrected in less than two hours. The fifth event involved a sticky solenoid and pluggage of an air starting line for which restoration time was 7-2/3 hours.

The turbocharger failures were due to a momentary deficiency in lube oil pressure during the switchover from an electric to engine driven oil pump while the engine was coming to speed. Provisions have been incorporated in the Unit 2 design to preclude this. Specifically, the AC driven lube oil pump will run continuously and it will be backed up by a DC driven lube oil pump. Additionally, an idle start capability will be provided for the Unit 2 diesels, which will ensure proper engine lubrication during diesel testing. To avoid corrosion related problems, such as the sticky solenoid/plugged air line incident, the Unit 2 diesels will have a stainless steel air start system. Since the turbocharger failures resulted from a design feature that has been modified in the Unit 2 design, these two data points have been omitted from the FPL data base. A recent Turkey Point Diesel Generator Voltage Regulator Transformer problem was resolved by disconnecting a neutral lead, resulting in the elimination of third harmonic current heating effects. Since this problem was unique to the Turkey Point design and does not apply to the St. Lucie diesel generators, this data point was also omitted from the data base. The repair time frequency distribution based on St. Lucie and Turkey Point experience to date is as follows:

Repair Time (minutes)	Frequency of Occurrence	Repair Time (minutes)	Frequency of Occurrence
10	1	111	1
15	1	180	1
21	1	217	1
30	1	240	1
37	1	258	1
65	1	275	1
70	1	390	1
76	1	460	1
77	1	503	1
91	1	708	1
94	1	1435	1
		3563	1

The median diesel repair time is 111 minutes and the mean is 388 minutes. If each event is assumed to have equal probability of occurrence, then the probability of restoration of a safety-related diesel at an FPL nuclear facility can be expressed mathematically by $1-e^{-CT}$, where C is 0.16 hr^{-1}. 1/

1/ Although it is inappropriate to include the turbocharger and voltage regulator data, inclusion of these data does not alter the conclusions reached in question B1 supra, i.e., evaluation of a period exceeding about 1 to 4 hours is not required since the probability of not restoring AC power within that time period is acceptably low. Inclusion of these data yields a median of 217 minutes, a mean of 1434 minutes and a C of 0.04 hr^{-1}. This results in an expression for P(T) of 10^{-5} exp (-1.68T) as compared to 10^{-5} exp (-1.92T), which is used in the response to question B1 supra.

To respond to question B1 <u>supra</u> the equally important issue of how long it would take to restore offsite power was reviewed: FPL's history of system disturbances from January 1972 to present indicates that loss of offsite power to plants on the Florida Power & Light system was distributed as follows:

Duration (min.)	Frequency of Occurrence	Duration (min.)	Frequency of Occurrence
1	1	30	2
8	1	31	1
9	1	32	1
13	1	40	1
15	1	43	2
17	4	53	1
20	2	77	1
22	1		
23	1		

The median restoration time of offsite AC power is 21 minutes and the mean is 26 minutes. If each event is assumed to have equal probability of occurrence, then the probability of restoration of AC power to any FPL facility can be expressed mathematically by $1-e^{-CT}$, where $C = 1.6\ hr^{-1}$. In all system disturbances affecting FPL's nuclear plants the diesel generators started and supplied AC power for the duration of the incident.

In summary, FPL operating experience indicates that the duration of offsite power loss is short-lived. Thus, the probability of restoring offsite or onsite AC power within an hour is very high, which is reflected in the probability assessments provided in response to question B1 <u>supra</u>.

Question B4

Finally, in the light of the discussion of points 2 and 3 above, the parties are to review possible measures for decreasing the likelihood of exceeding design limits on the reactor fuel and pressure boundary under the assumption that there is some time available to activate an auxiliary power source subsequent to a total loss of AC power.

Response

As demonstrated by the responses to questions B1 and B2 supra, the potential for exceeding design limits on the reactor fuel and pressure boundary prior to restoration of AC power is acceptably low. The Unit 2 design as proposed is considered acceptable and in compliance with NRC requirements.

Since the ability to accommodate this loss of AC event is dependent on operator action during a non-design basis event, we have briefly reviewed the design with regard to areas that relate to their ability to cope with the postulated event. Loss of all AC power will be immediately evident to the operators since the unit will trip, the low 4 kV bus voltage and diesel failure to start alarms will annunciate, and the control room lighting will dim to the DC lighting system level. The DC system will provide power for requisite monitoring instrumentation and for auxiliary feedwater system operation. The detailed actions to stabilize the unit in this mode will be reviewed prior to issuance of an operating license to ensure that the operators have the capability to achieve and maintain hot shutdown conditions for the duration of the loss of all AC event.

References:

1. "NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," U. S. Nuclear Regulatory Commission, September 1975.
2. IEEE Std. 308-1974, "IEEE Standard Criteria for Class IE Power Systems for Nuclear Power Generating Stations."
3. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants."
4. Regulatory Guide 1.93, "Availability of Electric Power Sources."

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of: FLORIDA POWER AND LIGHT COMPANY (St. Lucie Nuclear Power Plant, Unit No. 2)	DOCKET NO. 50-389

AFFIDAVIT OF FREDERICK G. FLUGGER

I am Frederick G. Flugger, Supervisor, Plant Licensing, Power Plant Engineering Department for Florida Power and Light Company. My education and professional qualifications appear in the Nuclear Regulatory Commission's record of the St. Lucie 2 proceeding following Tr. 1310 .

The purpose of this affidavit is to respond to question B.1(b) concerning loss of all AC power from the Appeal Board's Order of March 10, 1978 in this proceeding.

Question B.1(b)

As a function of the delay time involved, what are the consequences of a loss of offsite power at St. Lucie 2 combined with failure of onsite power sources to start on demand (i.e., delayed start). No other failure of the system (e.g., LOCA) need be considered in this analysis.

RESPONSE

Loss of all AC power is not a design basis for St. Lucie. Like all other plants, St. Lucie has been designed to the single failure criterion, in accordance with applicable NRC regulations.

In order for a loss of all AC power to occur after a loss of offsite power, a double failure, i.e., the failure of two independent diesels to start and supply onsite power, is required. Consequently, a detailed analysis of such an event has not been performed.

However, assuming the hypothesis in the Board's question, there are two predominant safety functions to be performed following loss of offsite power and failure of onsite power to start; (1) removal of decay heat from the reactor coolant system and; (2) removal of decay heat from the spent fuel storage pool.

(1) Heat from the reactor core will be transferred to the steam generator by natural circulation of reactor coolant. Heat removal can then be accomplished by the feedwater provided by the auxiliary feedwater system and exhausted to the atmosphere by

the atmospheric steam dump valves. This process is totally independent of AC powered equipment and components. The feedwater will be supplied by a steam turbine driven auxiliary feedwater pump, operated with steam from the steam generators. The auxiliary feedwater pump takes suction from the condensate storage tank (CST). The CST contains a sufficient volume of condensate so that as so operated it would allow the unit to remain at hot standby for at least 16 hours.

(2) The loss of offsite power and the failure of onsite power to start will cause the spent fuel pool cooling system to stop operation. The decay heat from the stored spent fuel will cause the water temperature to rise and eventually boil.

The water level in the pool will not require make-up for at least 36 hours.

In view of the foregoing, FPL believes that either offsite power would be restored, or onsite power supplied, before any safety-related consequences would occur.

Frederick G. Flugger

FREDERICK G. FLUGGER

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Subscribed and sworn to before me this 31st day of March, 1978.

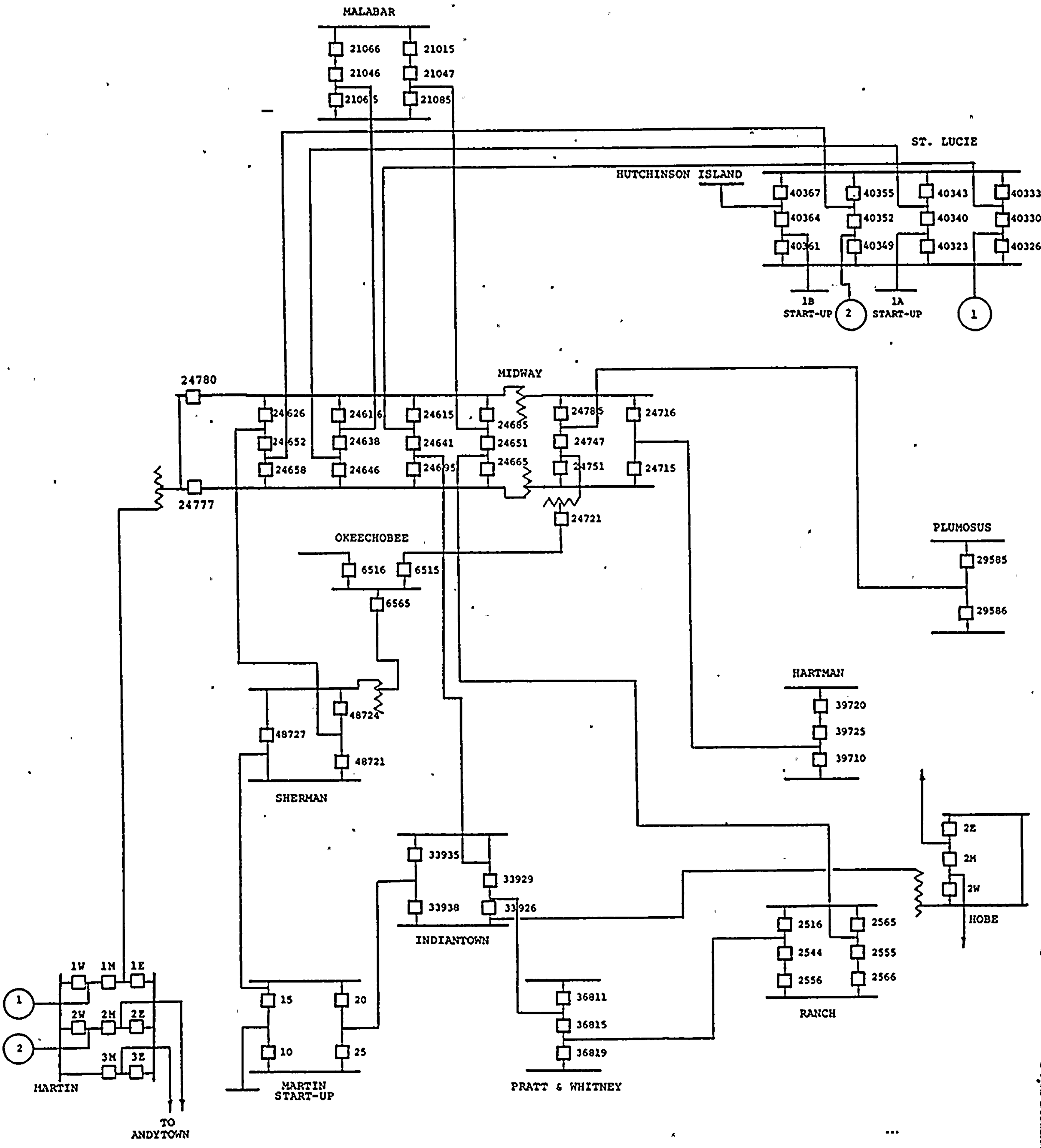
My commission expires: NOTARY PUBLIC STATE OF FLORIDA at LARGE MY COMMISSION EXPIRES AUGUST 24, 1981 BONDED THRU MAYNARD BONDING AGENCY

Lois T. Marino

NOTARY PUBLIC

1983

SYSTEM



MALABAR
21066
21015
21046
21047
21065
21085
ST. LUCIE
HUTCHINSON ISLAND
40367
40355
40343
40333
40364
40352
40340
40330
40361
40349
40323
40326
1B
START-UP
2
1A
START-UP
1
24780
MIDWAY
24626
24616
24615
24685
24785
24716
24652
24638
24641
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OKEECHOBEE
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PLUMOSUS
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HARTMAN
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SHERMAN
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INDIANTOWN
2E
2M
2W
HOBE
2516
2565
2544
2555
2556
2566
RANCH
1W
1M
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2W
2M
2E
3M
3E
1
2
MARTIN
TO
ANDYTOWN
15
20
10
25
MARTIN
START-UP
36811
36815
36819
PRATT & WHITNEY

ATTACHMENT #5 (ADDENDUM)

An analysis was performed on the contingency of the loss of both Midway 240 kV busses. The end result of the loss of both busses with a breaker and a half scheme is that the breakers connected to the busses are open and the lines coming into the substation only connect to the mid-breaker and continue on out again. Specifically at Midway, after the loss, there would be four lines that would pass through the Midway mid breakers:

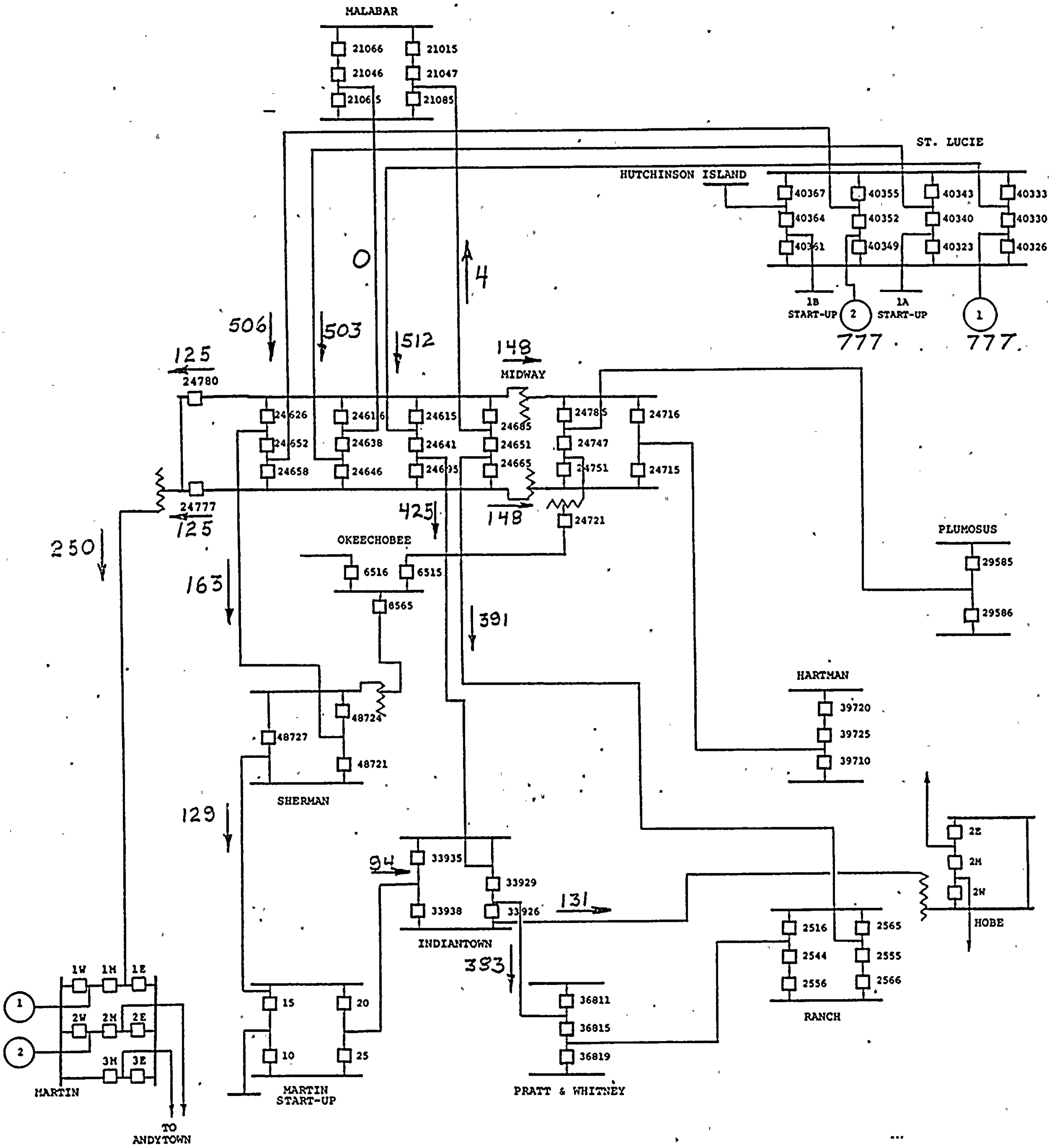
1. St. Lucie-Midway Sherman 230 kV
2. Malabar-Midway-St. Lucie 230 kV
3. St. Lucie-Midway-Indiantown 230 kV
4. Malabar-Midway-Ranch 230 kV

Of these four lines, one connects St. Lucie to the north, two connect St. Lucie to the south, and a fourth passes by with no connection to St. Lucie.

A loadflow study was performed to test what distribution of power flow would result if the loss of both busses occurred at the time of peak summer 1983 load with both St. Lucie units in service.

Two loadflows were run, (normal and with the loss of both busses) and the pertinent flows were plotted on the attached maps. These plots show that no line overloads would be expected and the St. Lucie 240 kV bus is still connected to both the north and south.

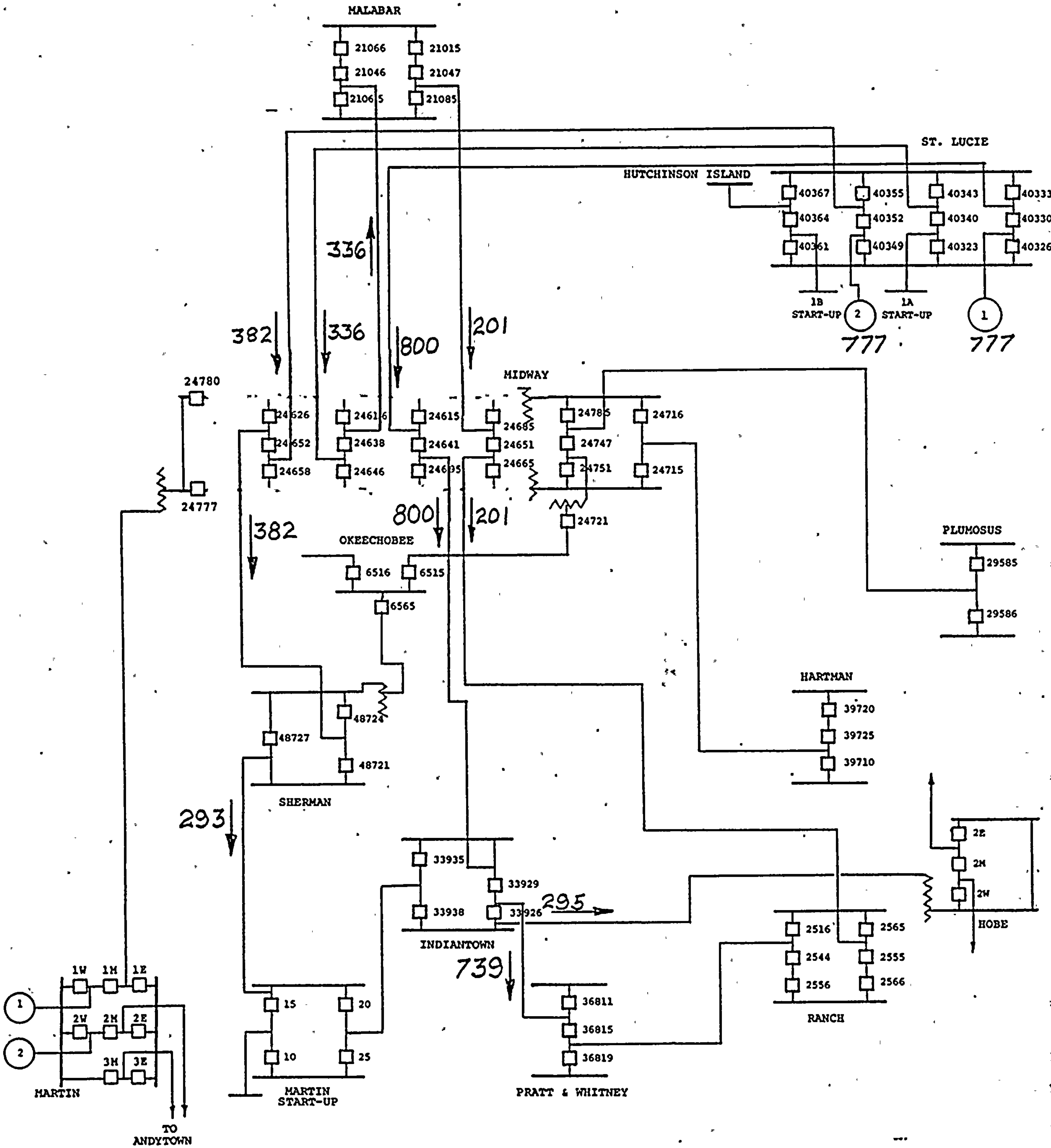
1983
NORMAL



MALABAR
21066
21015
21046
21047
21065
21085
ST. LUCIE
HUTCHINSON ISLAND
40367
40355
40343
40333
40364
40352
40340
40330
40361
40349
40323
40326
1B START-UP
1A START-UP
777
777
0
4
506
503
512
148
125
24780
MIDWAY
24626
24616
24615
24685
24785
24716
24652
24638
24641
24651
24747
24658
24646
24695
24665
24751
24715
24777
125
425
148
24721
250
OKEECHOBEE
6516
6515
6565
PLUMOSUS
29585
29586
163
391
HARTMAN
39720
39725
39710
48724
48727
48721
SHERMAN
129
94
33935
33929
33938
33926
131
INDIANTOWN
2E
2M
2W
HOBE
2516
2565
2544
2555
2556
2566
RANCH
393
36811
36815
36819
PRATT & WHITNEY
1W
1M
1E
2W
2M
2E
3M
3E
MARTIN
TO ANDYTOWN
15
20
10
25
MARTIN START-UP

1983

(LOSS OF BOTH BUSSES)



LE COLE 62079

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MIDWAY - TO - ST LUCIE PLANT #1 240 KV 11.62 mi
1975-1978

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
1/27/76	15:15: 0	15:15:30	00:00:30	NONE	FPL CREW
5/14/78	7:45: 0	7:55: 0	00:10:00	NONE	LIGHTNING ARRESTE

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
LIGHTNING ARRESTER	1	0
FPL CREW	1	0
TOTAL	2	0

LE COLE 62079

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MIDWAY -- TO -- ST LUCIE PLANT #2 240 KV
1975-1978

11.62 mi

DATE	OFF	PART ON	ALL ON	DMG ITEM	CAUSE
7/ 5/76	23:34: 0	23:34:30	00:00:30	NONE	UNKNOWN
7/11/76	23:30: 0	MOMENTARY			UNKNOWN
5/14/78	7:45: 0	7:55: 0	00:10:00	NONE	LIGHTNING ARRESTE

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
LIGHTNING ARRESTER	1	0
UNKNOWN	1	1
TOTAL	2	1

LE COLE 62079

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MIDWAY - TO - ST LUCIE PLANT #3 240 KV
1975-1978

11.62 mi

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
5/14/78	7:45: 0	7:55: 0	00:10:00	NONE	LIGHTNING ARRESTE
7/18/78	5:53:30	5:53:45	00:00:15	NONE	UNKNOWN

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
LIGHTNING ARRESTER	1	0
UNKNOWN	1	0
TOTAL	2	0

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MALABAR - TO - MIDWAY #1 240 KV 50.39 mi
1975-1978

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
6/ 5/75	0:31: 0	MOMENTARY			UNKNOWN
6/24/75	6: 0: 0	MOMENTARY			NON-DIS. LIGHT.
7/ 2/75	22: 0: 0	MOMENTARY			UNKNOWN
7/ 3/75	22:43: 0	MOMENTARY			UNKNOWN
7/16/75	5:31: 0	MOMENTARY			UNKNOWN
11/13/75	1:55: 0	MOMENTARY			UNKNOWN
7/13/76	15:25:30	22: 4: 0	6:38:30	INSULATOR	LIGHTNING
8/ 4/76	3:18: 0	MOMENTARY			UNKNOWN
9/13/76	22:37:30	MOMENTARY			UNKNOWN
11/12/76	0:34: 0	MOMENTARY			UNKNOWN
11/12/76	6: 9: 0	6:10: 0	0:01:00	NONE	UNKNOWN
6/14/77	1:48: 0	MOMENTARY			UNKNOWN
7/ 5/77	20:51: 0	4:51: 0	8:00:00	NONE	CONDUCTOR
7/23/77	3:53: 0	MOMENTARY			UNKNOWN
8/21/77	6: 5: 0	MOMENTARY			UNKNOWN
11/ 7/77	22: 7:30	MOMENTARY			UNKNOWN
12/ 4/77	2:34:45	MOMENTARY			UNKNOWN
2/16/78	6:37: 0	6:37:30	0:00:30	NONE	UNKNOWN
5/14/78	7:45: 0	7:47: 0	0:02:00	NONE	LIGHTNING ARRESTE
6/ 4/78	5:46: 0	MOMENTARY			WEATHER IN AREA
8/21/78	23:49: 0	MOMENTARY			UNKNOWN

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
CONDUCTOR	1	0
LIGHTNING ARRESTER	1	0
LIGHTNING	1	0
UNKNOWN	2	14
WEATHER IN AREA	0	1
NON-DIS. LIGHT.	0	1
TOTAL	5	16

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MALABAR - TO - MIDWAY #2 240 KV 53.74 mi
1975-1978

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
7/15/78	1:25:30	MOMENTARY			UNKNOWN

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
UNKNOWN	0	1
TOTAL	0	1

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MIDWAY - TO - RANCH #1 240 KV 53.31 mi
1975-1978

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
6/19/77	15:42:15	2:50: 0	12:32:15	X-ARM	VANDLALISM
7/ 3/77	2:37:30	MOMENTARY			INSULATOR
7/ 7/77	5:52: 0	MOMENTARY			UNKNOWN
7/22/77	5:47: 0	MOMENTARY			UNKNOWN
8/10/77	2:57: 0	2:57:15	00:00:15	NONE	NON-DIS. LIGHT.
8/18/77	14:10: 0	17:15: 0	3:05:00	NONE	X-ARM
8/27/77	22: 8:45	MOMENTARY			UNKNOWN
10/ 1/77	3:52: 0	MOMENTARY			UNKNOWN
10/17/77	4:54:30	11:49: 0	6:54:30	CONDUCTOR	INSULATOR
11/11/77	23:19: 0	MOMENTARY			UNKNOWN
12/29/77	21:59:45	6:27: 0	8:27:15	CONDUCTOR	INSULATOR
5/14/78	7:45: 0	7:48:15	00:03:15	NONE	LIGHTNING ARRESTE
6/17/78	15:10:15	9:16: 0	18:05:45	NONE	X-ARM
8/ 5/78	7:24:15	MOMENTARY			WEATHER IN AREA
8/12/78	2:57: 0	MOMENTARY			UNKNOWN
10/15/78	6:54: 0	MOMENTARY			UNKNOWN
10/20/78	1: 0: 0	MOMENTARY			UNKNOWN
10/30/78	5:43:30	MOMENTARY			UNKNOWN
12/14/78	19:35:30	MOMENTARY			UNKNOWN

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
X-ARM	2	0
INSULATOR	2	1
LIGHTNING ARRESTER	1	0
UNKNOWN	0	10
WEATHER IN AREA	0	1
NON-DIS. LIGHT.	1	0
VANDLALISM	1	0
TOTAL	7	12

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
INDIANTOWN - TO - MIDWAY 240 KV 24.12 mi
1975-1978

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
4/12/76	16:21:45	MOMENTARY			UNKNOWN
9/23/77	5:50:30	MOMENTARY			NON-DIS. LIGHT.

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
UNKNOWN	0	1
NON-DIS. LIGHT.	0	1
TOTAL	0	2

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FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
INDIANTOWN - TO - PRATT WHITNEY 240 KV 8.45 mi
1975-1978

DATE	TIME OFF	PART ON	ALL ON	DMG ITEM	CAUSE
6/17/78	15:10:45	MOMENTARY			UNKNOWN

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
UNKNOWN	0	1
TOTAL	0	1

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
PRATT WHITNEY - TO - RANCH #2 240 KV 20.74 mi
1975-1978

DATE	TIME: OFF	TIME: PART ON	TIME: ALL ON	DMG ITEM	CAUSE
4/ 6/76	10:56: 0	MOMENTARY			RELAYED WHEN CLOS
10/12/77	1:15:30	2:26:15	16:10:45	NONE	RELAY

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
RELAY	1	0
RELAYED WHEN CLOSED	0	1
TOTAL	1	1

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FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MIDWAY - TO - PLUMOSUS 138 KV 27.88 mi
1975-1978

DATE	TIME: OFF	TIME: PART ON	TIME: ALL ON	DMG ITEM	CAUSE
1/ 9/75	9:36: 0	9:39: 0	00:03:00	GUY WIRE	FPL CONT. CREW
3/14/75	9:28: 0	MOMENTARY			SWITCH
10/25/75	6:13: 0	17:19: 0	11:06:00	POLE	VEHICLE
5/15/76	16:21: 0	MOMENTARY			NON-DIS. LIGHT.
6/29/76	23:19:30	23:20: 0	00:00:30	NONE	TRANSFORMER
8/20/76	14:11:45	14:14:45	00:03:00	CONDUCTOR	VEHICLE
9/12/76	15:15:30	15:16: 0	00:00:30	INSULATOR	VANDLALISM
9/17/76	13:25: 0	MOMENTARY			NON-DIS. LIGHT.
12/13/76	1:16:15	MOMENTARY			UNKNOWN
12/14/76	14:25:30	MOMENTARY			UNKNOWN
12/16/76	14:21: 0	MOMENTARY			UNKNOWN
1/17/77	9:27:30	9:28: 0	00:00:30	NONE	FPL CREW
2/17/77	11:18: 0	11:18:15	00:00:15	NONE	RELAYED WHEN CLOS
4/21/77	9:35: 0	MOMENTARY			X-ARM
6/ 4/77	23:29: 0	9: 7: 0	9:38:00	NONE	X-ARM
8/26/77	15:21:45	MOMENTARY			UNKNOWN
8/26/77	15:22:30	MOMENTARY			UNKNOWN
8/26/77	15:22:45	MOMENTARY			UNKNOWN
9/ 1/77	7:37:45	MOMENTARY			SWITCH
9/20/77	19:53: 0	MOMENTARY			UNKNOWN
9/22/77	5:50:30	MOMENTARY			UNKNOWN
10/12/77	8:27:15	MOMENTARY			UNKNOWN
11/ 5/77	18: 2: 0	MOMENTARY			RELAYED WHEN CLOS
11/ 7/77	13: 0: 0	MOMENTARY			FPL CREW
11/16/77	7:51: 0	MOMENTARY			RELAYED WHEN CLOS
12/13/77	8:54:45	MOMENTARY			RELAYED WHEN CLOS
12/23/77	9:13: 0	MOMENTARY			RELAYED WHEN CLOS
1/ 3/78	8:58: 0	MOMENTARY			RELAYED WHEN CLOS
2/18/78	15: 8:30	MOMENTARY			WEATHER IN AREA
3/ 3/78	13:26:30	MOMENTARY			WEATHER IN AREA
3/18/78	7: 1: 0	MOMENTARY			SWITCH
3/18/78	7:41:30	MOMENTARY			SWITCH
5/14/78	7:45: 0	7:59:30	00:14:30	NONE	LIGHTNING ARRESTE
6/ 9/78	15:18:45	MOMENTARY			UNKNOWN
8/11/78	7:45:30	MOMENTARY			SWITCH
9/15/78	20: 2: 0	MOMENTARY			NON-DIS. LIGHT.
9/24/78	17:24: 0	MOMENTARY			WEATHER IN AREA
10/11/78	7:42: 0	MOMENTARY			SWITCH
11/29/78	10:58: 0	10:59: 0	00:01:00	NONE	SWITCH
12/ 1/78	14:19:45	17:33: 0	3:13:15	NONE	FPL CREW

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
MIDWAY - TO - PLUMOSUS 138 KV
1975-1978

- DATE --	---------- TIME -------- -- OFF --	- PART - -- ON --	-- ALL - -- ON --	--- DMG ITEM ---	---- CAUSE ----
12/ 8/78	7:50:30	MOMENTARY			SWITCH

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
SWITCH	1	7
RELAYED WHEN CLOSED	1	5
X-ARM	1	1
LIGHTNING ARRESTER	1	0
VEHICLE	2	0
FPL CREW	2	1
FPL CONT. CREW	1	0
UNKNOWN	0	10
WEATHER IN AREA	0	3
NON-DIS. LIGHT.	0	3
VANDLALISM	1	0
TRANSFORMER	1	0
TOTAL	11	30

LE COLE 60179

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
PLUMOSUS - TO - RIVIERA #1 138 KV
1975-1978

13.3 mi

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
5/13/75	18:23: 0	MOMENTARY			LIGHTNING
6/20/75	16:55: 0	16:58:30	00:03:30	NONE	LIGHTNING
6/28/75	16:40: 0	MOMENTARY			NON-DIS. LIGHT.
5/ 7/76	6:41: 0	MOMENTARY			GUY WIRE
7/26/76	5: 0: 0	MOMENTARY			UNKNOWN
10/27/76	5:43: 0	MOMENTARY			UNKNOWN
6/ 7/77	9:24: 0	MOMENTARY			UNKNOWN
8/11/77	10: 9: 0	MOMENTARY			NON-DIS. LIGHT.
9/21/77	15:29: 0	MOMENTARY			NON-DIS. LIGHT.
9/22/77	6:17:15	MOMENTARY			NON-DIS. LIGHT.
3/ 3/78	14:19:15	14:22:30	00:03:15	POLE	WIND
4/20/78	11:33:15	MOMENTARY			UNKNOWN
5/ 6/78	14:28:45	14:31:15	00:02:30	NONE	NON-FPL CONT.
6/21/78	5: 9: 0	MOMENTARY			INSULATOR
6/21/78	5:21:30	MOMENTARY			INSULATOR
6/21/78	19:16:30	MOMENTARY			NON-DIS. LIGHT.
10/21/78	18:29:30	22:39: 0	04:09:30	NONE	GUY WIRE

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
INSULATOR	0	2
GUY WIRE	1	1
NON-FPL CONT.	1	0
LIGHTNING	1	1
WIND	1	0
UNKNOWN	0	4
NON-DIS. LIGHT.	0	5
TOTAL	4	13

LE COLE 60179

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
PLUMOSUS - TO - RIVIERA #2 138 KV
1975-1978

14.86 mi

DATE	TIME OFF	TIME PART ON	TIME ALL ON	DMG ITEM	CAUSE
4/11/75	21:36: 0	MOMENTARY			NON-DIS. LIGHT.
11/ 3/75	16:45: 0	MOMENTARY			INSULATOR
11/ 3/75	16:47: 0	MOMENTARY			INSULATOR
11/ 3/75	17:19: 0	0: 4: 0	6:45:00	NONE	INSULATOR
11/ 4/75	5:46: 0	15:52: 0	10:06:00	INSULATOR	SALT SPRAY
11/ 4/75	3:48: 0	MOMENTARY			UNKNOWN
11/ 4/75	5:29: 0	MOMENTARY			UNKNOWN
3/28/76	1:16: 0	MOMENTARY			INSULATOR
3/28/76	1:59: 0	MOMENTARY			INSULATOR
6/17/76	9: 3: 0	9: 4: 0	00:01:00	NONE	NON-FPL CONT.
7/16/76	16:26:30	MOMENTARY			UNKNOWN
9/14/76	14:18: 0	14:19: 0	00:01:00	NONE	TRANSFORMER
5/23/77	6:11:45	MOMENTARY			UNKNOWN
6/29/77	15:35: 0	MOMENTARY			NON-DIS. LIGHT.
1/ 8/78	20:40:45	20:41:15	00:00:35	NONE	WIND
3/22/78	6:56:15	6:57: 0	00:00:45	NONE	TRANSFORMER
3/23/78	7:22: 0	7:26: 0	00:04:00	NONE	TRANSFORMER
4/13/78	7: 0: 0	MOMENTARY			RELAYED WHEN CLOS
4/20/78	5:54:15	5:54:45	00:00:30	NONE	TRANSFORMER

TOTAL OUTAGES BY CAUSE

CAUSE	SUSTAINED	MOMENTARY
RELAYED WHEN CLOSED	0	1
INSULATOR	1	4
NON-FPL CONT.	1	0
SALT SPRAY	1	0
WIND	1	0
UNKNOWN	0	4
NON-DIS. LIGHT.	0	2
TRANSFORMER	4	0
TOTAL	8	11