

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)

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

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

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

17 First, consider the frequency of loss of the electrical grid. FPL nuclear
18 operating history suggests a frequency of outage of about 4×10^{-1} per year
19 for the FPL grid. Although there is little comparative historical data readily

1 available, I believe that the relative difference in reliability between the
2 FPL system, based on its historical data, and other grids associated with the
3 general population of nuclear plants is probably not more than a factor of about 2.
4 It must be noted that as the FPL system evolves during construction of Unit 2
5 and during its operation, any difference in reliability that may be inferred
6 from FPL's operating history to date will be reduced or eliminated. The
7 testimony in response to Appeal Board questions A1 and D discusses the substan-
8 tial actions that have been and will be taken to improve the reliability of
9 the FPL grid.

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

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

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

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

6 Third, an unprotected loss of coolant accident (LOCA) does not result from the
7 postulated loss of all AC event. There is no failure of the reactor coolant
8 pressure boundary associated with this event. A reactor coolant pump (RCP)
9 seal can only yield very small and acceptable leak rates. (See the response
10 to question B2 infra.) Unit 2 has more than adequate capability to remove
11 decay heat, which is necessary to accommodate the postulated loss of all AC
12 event. There is sufficient condensate to provide steam generator makeup for
13 at least 16 hours, the auxiliary feedwater pump is steam driven, auxiliary
14 feedwater pump control and auxiliary feedwater system valves are DC powered,
15 and the steam generators have sufficient inventory to allow the operator about
16 55 minutes to actuate auxiliary feedwater before steam generator dryout occurs.
17 With these considerations in mind we can procede with the responses to the
18 Appeal Board's questions.

19 Question A2

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

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

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

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

11 Response

12 The Board's question cites a possible literal interpretation of GDC 17 that
13 contravenes the intent of this design criterion. The intent of GDC 17 is
14 provided in a straightforward manner by the language of proposed GDC 24 and
15 39 issued for guidance by the Atomic Energy Commission on July 10, 1967 before
16 GDC 17 was adopted in its present form. Their language states:

17 GDC 24

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

21 GDC 39

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

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

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

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

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

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

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

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

9 Question B1

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

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

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

38 25/ Fitzpatrick Affidavit of June 12, 1978, p. 4. Also see Regulatory Guide
39 1.108, Section B.

1 26/ This conclusion further assumes that the failure of two diesel generators
2 to start would be statistically independent events, an assumption which
3 leads to the lowest likelihood of combined failure, and which might be
4 nonconservative if there exists the potential for common failure modes
5 for the onsite systems.

6 27/ NUREG 75/087, Section 2.2.3, paragraph II.

7 28/ We have accepted the Standard Review Plan guideline values as reasonable
8 in other case. Public Service Electric and Gas Company (Hope Creek
9 Units 1 and 2), ALAB - 429, 6 NRC 229, 234 (1977).

10 Response

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

18 Event Frequency

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

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

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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

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

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

15 ESF Reliability

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

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

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

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

14
$$P(T) = P(A) \cdot P(B) \cdot P(C) \cdot P(D) \cdot P(E) \cdot P(F)$$

15 where: $P(T)$ = probability of not restoring AC power by time "T"

16 $P(A)$ = probability of loss of offsite power

17 $P(B)$ = probability of loss of first diesel

18 $P(C)$ = probability of loss of second diesel

19 $P(D)$ = probability that offsite power is not
20 repaired and returned to service by time "T"

21 $P(E)$ = probability that first diesel is not
22 repaired and returned to service by time "T"

23 $P(F)$ = probability that second diesel is not
24 repaired and returned to service by time "T"

25 The restoration of AC probability terms, $P(D)$, $P(E)$ and $P(F)$ can be developed
26 in a straightforward manner. Let $P(T)$ be the probability that AC is not
27 restored at time "T", $P(T+\Delta T)$ this probability at a finite later time "T+ ΔT ",

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1 and $C \cdot \Delta T$ the repair probability during the time interval ΔT (where C is a
2 constant). Then,

3
$$P(T+\Delta T) = P(T) \cdot (1-C \cdot \Delta T)$$

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

5
$$\frac{d P(T)}{dT} = -C \cdot P(T)$$

6

7 whose solution is:

8
$$P(T) = e^{-CT}$$

9 The equation for $P(T)$ can be used to mathematically represent $P(D)$, $P(E)$ and
10 $P(F)$. Examination of historical data allows determination of the time constant
11 " C " for each of these probability terms. An evaluation of FPL system data from
12 1972 to present indicates that a time constant of 1.6 hr^{-1} is appropriate for
13 $P(D)$. (See the response to question B3 infra.) St. Lucie 1 and Turkey Point
14 diesel generator outage data indicate that a time constant of 0.16 hr^{-1} is
15 appropriate for both $P(E)$ and $P(F)$. (See the response to question B3 infra.)

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

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

25
$$P(T) = 10^{-5} \exp (-1.92T)$$

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

3	Duration of loss of AC	Probability of Having a Total Loss
4	<u>"T" (hours)</u>	<u>of AC Power that Lasts "T" Hours, P(T)</u>
5	0	1×10^{-5}
6	1	2×10^{-6}
7	1.2	1×10^{-6}
8	2	2×10^{-7}
9	2.4	1×10^{-7}
10	3	3×10^{-8}
11	4	5×10^{-9}

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

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

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

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1 Question B2

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

9 29/ [References fn 24 reproduced below:]

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

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

18 Response

19 The Flugger Affidavit filed in response to the Appeal Board's order of March 10,
20 1978 concluded that there was a sufficient volume of condensate storage to allow
21 the unit to maintain hot standby conditions for at least 16 hours; the spent
22 fuel storage pool would not require makeup for at least 36 hours; and that power
23 would be restored before any unacceptable consequences would occur. The Fitz-
24 patrick Affidavit, which provided the Staff response, concurred with FPL's
25 response, but went on to suggest that a failure of a reactor coolant pump (RCP)
26 seal could potentially occur after one hour as a result of the loss of all AC
27 power. For the reasons set forth below, the difference can be reconciled and
28 Unit 2 can be safely maintained in a hot shutdown condition until AC power is
29 restored.

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1 At the outset, it is necessary to analyze the actual condition of the reactor
2 coolant pumps during the event. Upon loss of AC power the reactor will trip,
3 the RCP's will coast down and stop, and cooling water flow to the RCP seals
4 will cease. This static (pump not running) condition is much less severe than
5 the dynamic (pump running) condition discussed in the Unit 2 PSAR at section
6 9.2.2.3.1, which provides a basis for concluding that running the pumps for
7 about one hour, without cooling water to the seals, would not result in pump
8 seizure or unacceptable RCP seal failure.

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

22 Pressure breakdown devices are installed parallel to the first three seals. Reactor
23 coolant at a rate of 1 gpm passes through these devices such that reactor coolant
24 system pressure is distributed equally across the first three seals, i.e., they
25 normally operate at about 1/3 of their design pressure. The fourth seal is subjected

1 to a nominal backpressure and acts as a vapor barrier/backup seal during normal
2 operation. The RCP controlled bleedoff flow of 1 gpm/pump is directed to the
3 chemical and volume control system.

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

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

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

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

23 At the system temperature of 550°F, the elastomeric material, Ethylene Propylene or
24 Nitrile, would be subject to extrusion and hardening, i.e., gradual loss of flexi-
25 bility and permanent setting in a deflected position. The reactor coolant pump

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1 manufacturer has demonstrated, however, the sealing characteristics of the
2 elastomeric material under thermal conditions equivalent to those resulting from
3 the postulated loss of all AC event. Confined "O" rings of the material have
4 been used on several flanged joints of a reactor coolant pump hot test loop,
5 where they have been subjected to temperatures of 550°F for in excess of 100
6 hours during routine pump acceptance testing. The "O" rings maintained their
7 sealing capability without any problems, and as would be expected, hardening
8 and permanent setting of the "O" rings occurred. Under static conditions
9 sealing would be maintained since (i) the "O" rings are backed up by lapped
10 seats, (ii) the "U" cups are totally captured, and (iii) most of the harden-
11 ing would occur on cooldown, rather than at the elevated temperature.

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

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

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

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

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

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

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

5 There are two safety class DC batteries installed at this facility. Each is an
6 1800 ampere-hour, on an 8 hour basis, battery, i.e., each can supply 225 amperes
7 continuously for 8 hours. The DC load required for remote manual auxiliary
8 feedwater operation, control room lighting and one channel of instrumentation
9 from the instrument busses is about 100 amperes. Thus, if in say 1/2 hour or
10 so, one battery is secured and parasitic loads are stripped from the on-line
11 battery, there is more than sufficient battery capacity to accommodate the
12 postulated transient. One battery could sustain the DC load for about 13 hours,
13 the second battery could be reconnected to supply the DC load for another 12 hours
14 or so.

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

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

7 Question B3

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

11 Response

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

21 The concept of diesel reliability should also be placed in proper perspective.
22 A 10^{-2} probability of demand with a confidence level of 95% was demonstrated by
23 a 300 start shop test program for a Unit 1 diesel (see Unit 1 FSAR section
24 8.3.1.3). A successful attempt occurred if the diesel performed the sequence:
25 fast start, automatically bringing the set to full speed and voltage, immediately

1 loading the generator to 60% of continuous rating, and maintaining the 60%
2 load for 5 minutes. A failure in any portion of the sequence was considered
3 a failure per demand. This sequence is based on LOCA generated ESF require-
4 ments, which place exacting quick start design requirements upon the diesel
5 generators.

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

14 The turbocharger failures were due to a momentary deficiency in lube oil
15 pressure during the switchover from an electric to engine driven oil pump while
16 the engine was coming to speed. Provisions have been incorporated in the Unit 2
17 design to preclude this. Specifically, the AC driven lube oil pump will run
18 continuously and it will be backed up by a DC driven lube oil pump. Additionally,
19 an idle start capability will be provided for the Unit 2 diesels, which will
20 ensure proper engine lubrication during diesel testing. To avoid corrosion
21 related problems, such as the sticky solenoid/plugged air line incident, the
22 Unit 2 diesels will have a stainless steel air start system. Since the turbo-
23 charger failures resulted from a design feature that has been modified in the
24 Unit 2 design, these two data points have been omitted from the FPL data base.
25 A recent Turkey Point Diesel Generator Voltage Regulator Transformer problem

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1 was resolved by disconnecting a neutral lead, resulting in the elimination
 2 of third harmonic current heating effects. Since this problem was unique to
 3 the Turkey Point design and does not apply to the St. Lucie diesel generators,
 4 this data point was also omitted from the data base. The repair time frequency
 5 distribution based on St. Lucie and Turkey Point experience to date is as
 6 follows:

7	Repair Time	Frequency of	Repair Time	Frequency of
8	(minutes)	Occurrence	(minutes)	Occurrence
9	10	1	111	1
10	15	1	180	1
11	21	1	217	1
12	30	1	240	1
13	37	1	258	1
14	65	1	275	1
15	70	1	390	1
16	76	1	460	1
17	77	1	503	1
18	91	1	708	1
19	94	1	1435	1
			3563	1

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

24 ^{1/} Although it is inappropriate to include the turbocharger and voltage
 25 regulator data, inclusion of these data does not alter the conclusions reached
 26 in question B1 supra, i.e., evaluation of a period exceeding about 1 to 4
 27 hours is not required since the probability of not restoring AC power within
 28 that time period is acceptably low. Inclusion of these data yields a median
 29 of 217 minutes, a mean of 1434 minutes and a C of 0.04 hr^{-1} . This results
 30 in an expression for P(T) of $10^{-5} \exp(-1.68T)$ as compared to $10^{-5} \exp$
 31 $(-1.92T)$, which is used in the response to question B1 supra.

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

5	Duration	Frequency of	Duration	Frequency of
6	(min.)	Occurrence	(min.)	Occurrence
7	1	1	30	2
8	8	1	31	1
9	9	1	32	1
10	13	1	40	1
11	15	1	43	2
12	17	4	53	1
13	20	2	77	1
14	22	1		
15	23	1		

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

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

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1 Question B4

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

7 Response

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

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

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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."

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AFFIDAVIT OF FREDERICK G. FLUGGER

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

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

11

12 Question B.1(b)

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

1 in this analysis.

2

3 RESPONSE

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

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

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

21 (1) Heat from the reactor core will be
22 transferred to the steam generator by natural circu-
23 lation of reactor coolant. Heat removal can then be
24 accomplished by the feedwater provided by the auxiliary
25 feedwater system and exhausted to the atmosphere by

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

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

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

17 In view of the foregoing, FPL believes that
18 either offsite power would be restored, or onsite power
19 supplied, before any safety-related consequences would
20 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: _____

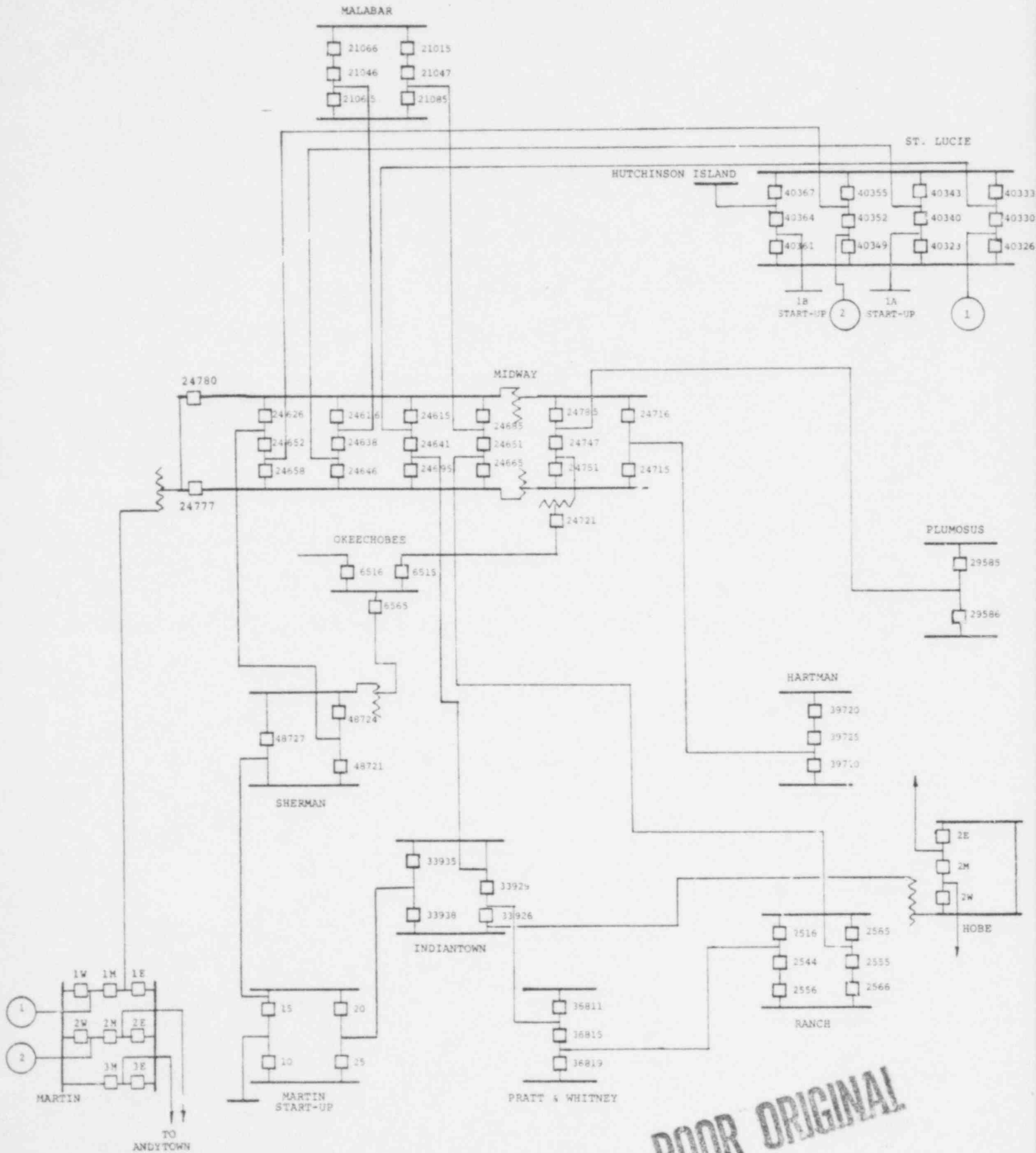
NOTARY PUBLIC STATE OF FLORIDA IN LARGE
MY COMMISSION EXPIRES AUGUST 24, 1981
BONDED THRU MAYNARD BONDING AGENCY

Louis J. Merino
NOTARY PUBLIC

POOR ORIGINAL

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1983 SYSTEM



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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:

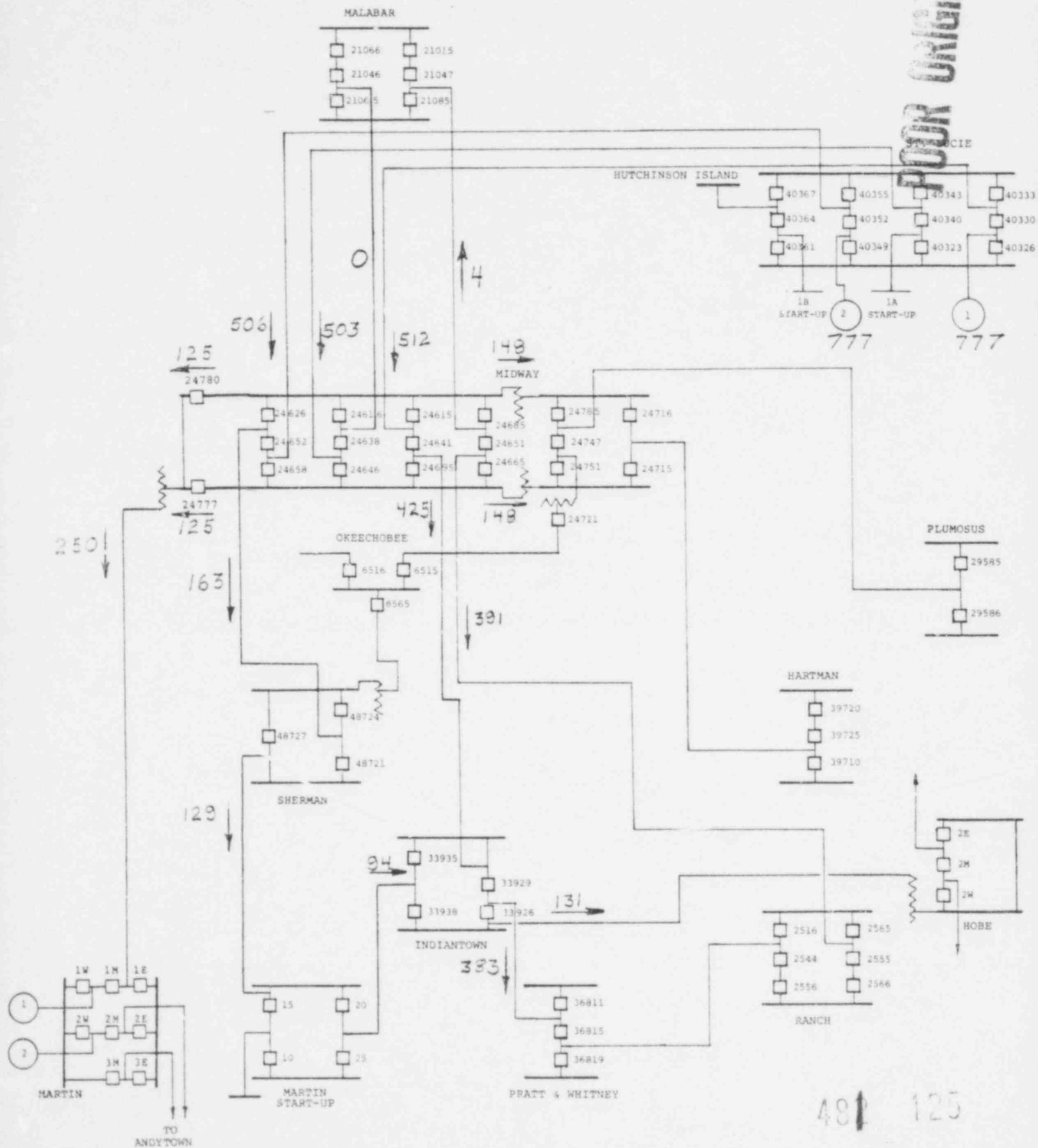
1. St. Lucie-Midway Sherman 230 kV
2. Malabar-Midway-St. Lucie 230 kV
3. St. Lucie-Midway-Indiantown 230 kV
4. Malabar-Midway-Ranch 230 kV

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.

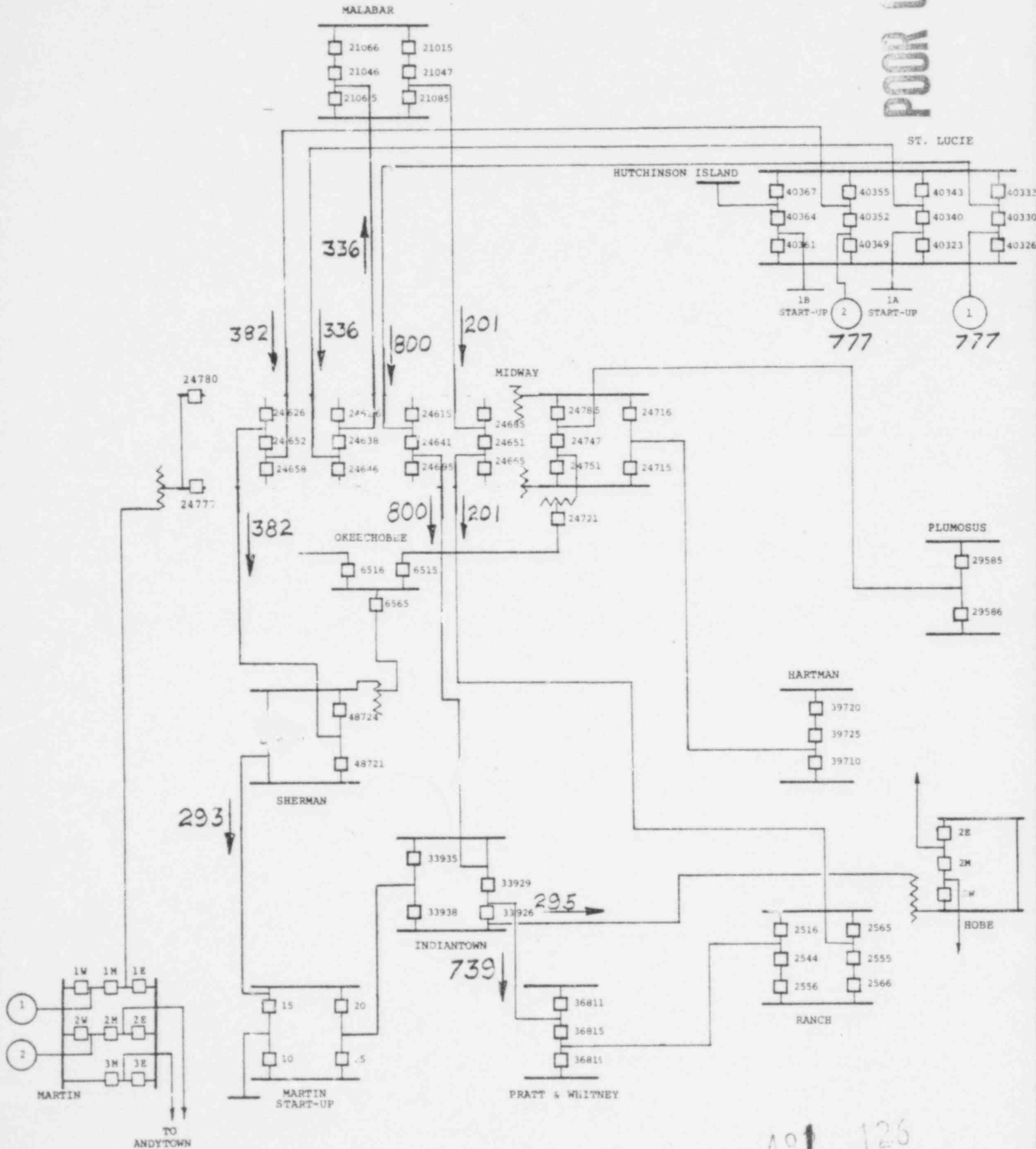
1983
NORMAL



1983

(LOSS OF BOTH BUSES)

POOR ORIGINAL



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

11.62 mi

----- TIME -----		- PART - - ALL -		- DATE - - OFF - - ON - - ON - -		DHG 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 ARPESTE	

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

POOR ORIGINAL

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

----- TIME -----		- PART - -- ALL -		- DATE - -- OFF - -- ON - -- ON - --		DWG 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

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POOR ORIGINAL

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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

----- TIME -----		- PART - - - ALL -			
- DATE -	OFF	ON	ON	- - - - -	CAUSE - - - -
5/14/78	7:45:0	7:55:0	00:10:00	NONE	LIGHTNING ARRESTER
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

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POOR ORIGINAL

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED

LINE SECTION
MALABAR - TO - MIDWAY #1 240 KV
1975-1978

50.39 mi

DATE	TIME OFF	TIME ON	PART	ALL	DHG 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	23:45: 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

POOR ORIGINAL

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

53.74 mi

----- TIME -----
- PART - - ALL -
- DATE -- -- OFF -- -- ON -- -- 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

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POOR ORIGINAL

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FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED

LINE SECTION
MIDWAY - TO - RANCH #1 240 KV
1975-1978

53.31 mi

DATE	OFF	ON	TIME	PART	ALL	DMG ITEM	CAUSE
6/19/77	15:42:15	2:50:0	12:32:15				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

POOR ORIGINAL

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FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED

LINE SECTION
INDIANTOWN - TO - MIDWAY 240 KV 24.12 mi
1975-1978

----- TIME -----			
- PART -		- ALL -	
- DATE -	- OFF -	- ON -	- 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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POOR ORIGINAL

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

8.45 mi

----- TIME -----
- PART - -- ALL -
- DATE -- -- OFF -- -- ON -- -- 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

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POOR ORIGINAL

LE COLE 52479

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

20.74 mi

- DATE -	----- TIME -----		DMG ITEM	CAUSE
	OFF	ON		
4/ 6/76	10:56:0	MOMENTARY		RELAYED WHEN CLOS
10/12/77	1:15:30	2:26:15 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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POOR ORIGINAL

LE COLE 52479

FLOKIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED

LINE SECTION
MIDWAY - TO - PLUMOSUS 138 KV
1975-1978

27.88 mi

DATE	TIME		ALL	DMG ITEM	CAUSE
	OFF	ON			
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	VANDALISM
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

POOR ORIGINAL

481 136

LE COLE 52479

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED

LINE SECTION
MIDWAY - TO - PLUMOSUS 138 KV
1975-1978

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

TOTAL OUTAGES BY CAUSE
CAUSE SUSAINED 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
VANDALISM	1	0
TRANSFORMER	1	0
TOTAL	11	30

POOR ORIGINAL

481

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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

TIME				CAUSE			
DATE		OFF	ON	PART	ALL	DMG ITEM	CAUSE
5/13/75	10:23:	0					LIGHTNING
6/20/75	16:55:	0	16:50:30			00:03:30	LIGHTNING
6/20/75	16:40:	0					NON-DIS. LIGHT.
5/ 7/76	6:41:	0					GUY WIRE
7/26/76	5: 0:	0					UNKNOWN
10/27/76	5:43:	0					UNKNOWN
6/ 7/77	9:24:	0					UNKNOWN
8/11/77	10: 9:	0					NON-DIS. LIGHT.
9/21/77	15:29:	0					NON-DIS. LIGHT.
9/22/77	6:17:15						NON-DIS. LIGHT.
3/ 3/78	14:19:15		14:22:30			00:03:15	WIND
4/20/78	11:33:15						UNKNOWN
5/ 6/78	14:20:45		14:31:15			00:02:30	NON-FPL CONT.
6/21/78	5: 9:	0					INSULATOR
6/21/78	5:21:30						INSULATOR
6/21/78	19:16:30						NON-DIS. LIGHT.
10/21/78	18:29:30		22:39: 0			04:09:30	GUY WIRE

TOTAL OUTAGES BY CAUSE
CAUSE

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

POOR ORIGINAL

LE COLE 60179

FLORIDA POWER & LIGHT COMPANY
TRANSMISSION INTERRUPTION SUMMARY
SYSTEM OUTAGES EXCLUDED
LINE SECTION
PLUMOSUS - TO - RIVIERA 62 138 KV
1975-1978

14.86 mi

TIME		PART		ALL		DMG ITEM	CAUSE
DATE	OFF	ON	ON	ON	ON		
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
3/20/76	1:16: 0	MOMENTARY					INSULATOR
3/20/76	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

POOR ORIGINAL

482 139