ICE-272F/I12173/mls

ATTACHMENT #6

INTERFACE DESIGN REQUIREMENTS

FOR

QSPDS/SAS DATA COMMUNICATIONS

FOR

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNITS NO. 3 AND 4

REQUIREMENT NUMBER 16081-ICE-3111, REVISION 00

Nuclear Power Systems
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PURPOSE

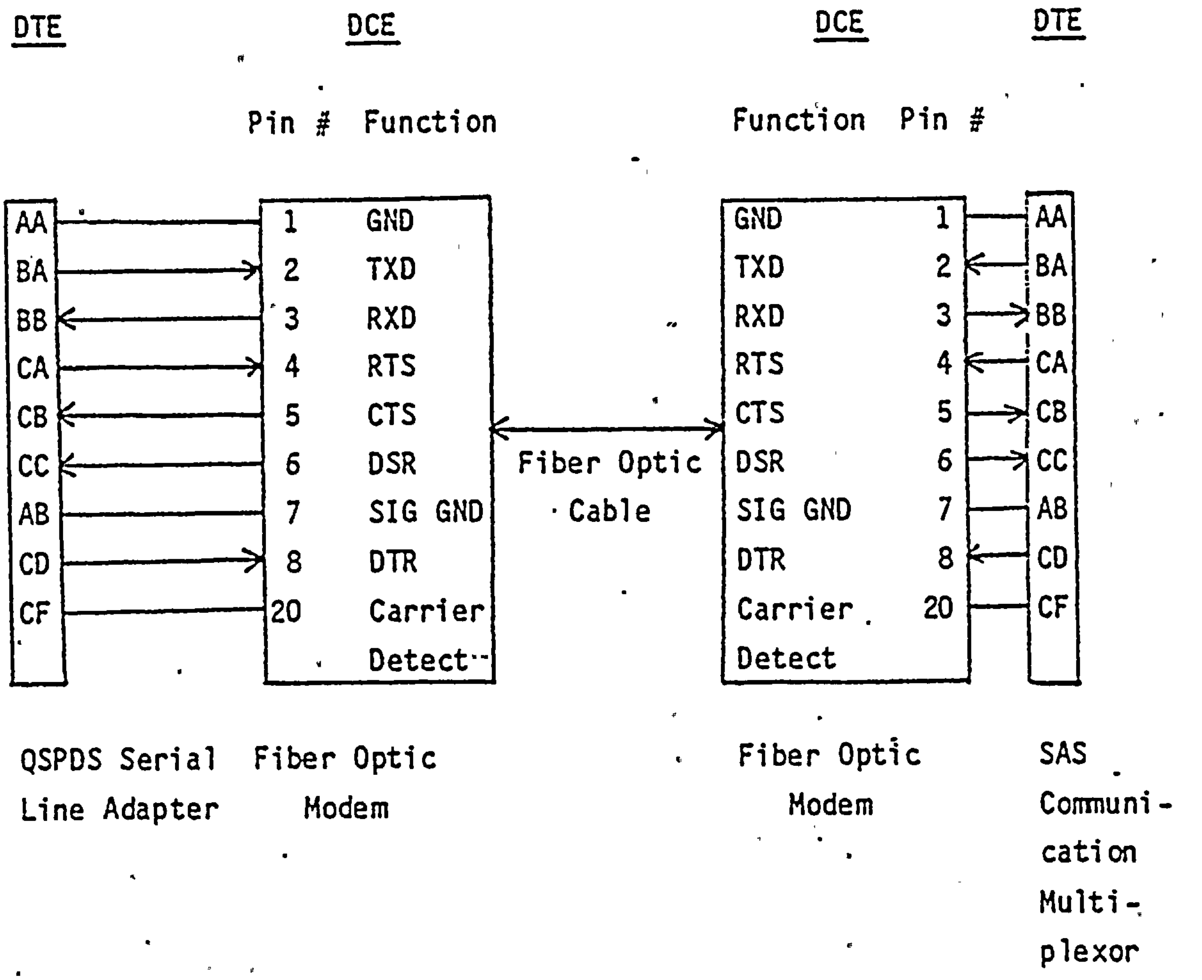
This document provides the criteria governing the digital interfaces between the Qualified Safety Parameter Display System (QSPDS) and the Safety Assessment System (SAS) for Florida Power and Light Company's Turkey Point Units No. 3 and 4.

The interface design requirements presented herein are intended to define both the functional and operational requirements for data communications between QSPDS and SAS. Hardware and software requirements are established to complete the specification and design of the interface.

SCOPE

The QSPDS/SAS interface shall consist of full duplex digital data links between the two QSPDS processors and the SAS processor.

FIGURE 1
DATA LINK INTERCONNECTION



QSPDS Cabinet

The RXD to TXD, CTS to RTS and DSR to DTR interchanges are done by the modems. Therefore from computer to modem no interchange is required, and there is a one-to-one connection as shown above. The above configuration diagram assumes that the QSPDS and SAS computers are configured as Data Terminal Equipment (DTE).

Designation	Definition
AA	Overall Shield (Protective Ground)
AB	Signal Ground
BA	Transmit Data (TXD)
BB	Receive Data (RXD)
CA	Request to Send (RTS)
CB	Clear to Send (CTS)
CC	Data Set Ready (DSR)
CD	Data Terminal Ready (DTR)
CF	Carrier On

The interconnection of these signals is shown in Figure 1. Signal characteristics are defined by the EIA Standard RS-232-C (Reference 3.3.1).

5.0 OPERATIONAL REQUIREMENTS

5.1 INTERFACE CONTROL

5.1.1 Addressing

There shall be two consecutive device addresses for each of the QSPDS/SAS data links; one for receive and one for transmit. Each address shall have separate interrupt control logic associated with it.

5.1.2 Interface Commands

The internal QSPDS data link interface cards shall accept and implement as a minimum the following processor commands:

a. Separate Interrupt Enable/Disable/Disarm Commands for both Transmit and Receive,

b. Data Terminal Ready (CD),

c. Request to Send (CA).

PREFERRED ALARM STATUS BYTE
CONFIGURATION

LSB	1 - HI	(High Limit Alarm)
	2 - LO	(Low Limit Alarm)
	3 - FAIL	(Failed Sensor)
	4 - BAD	(Bad Data - Out of Range)
	5 - SUSPCT	(Suspect Data)
	6 - QSPTRB	(QSPDS trouble)
	7 - SET TO 1	(To Avoid Confusion with GS)
MSB	8 - PARITY	(Odd Parity)

Explanations:

Failed Sensor - Equipment associated with the sensor has failed.

Bad Data - Sensor input is outside the valid range for the sensor.

Suspect Data - Calculated results which were affected/revised due to bad data or failed sensor being present.

The convention "1" = alarm/failed condition and "Ø" = normal/operational condition will be employed.

SIGNAL VALUE

Signal value can be any number represented by 1 to 8 ASCII characters.

Ex: 2000.2 is represented by 6 ASCII characters including the decimal point.

CROSS REFERENCE TABLE
CHANNEL A (Continued)

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
49	Q2HIA	CET Highest Temp Quad-2	32 to 2300	°F
50	Q2HIDA	CET Highest Temp ID (Quad-2)	0 to 10	
51	Q2NHIA	CET Next Highest Temperature Quad-2	32 to 2300	°F
52	Q2NIDA	CET Next Highest Temperature ID (Quad-2)	0 to 10	
53	Q3HIA	CET Highest Temp Quad-3	32 to 2300	°F
54	Q3HIDA	CET Highest Temp ID (Quad-3)	0 to 10	
55	Q3NHIA	CET Next Highest Temperature Quad-3	32 to 2300	°F
56	Q3NIDA	CET Next Highest Temperature ID (Quad-3)	0 to 10	
57	Q4HIA	CET Highest Temp Quad-4	32 to 2300	°F
58	Q4HIDA	CET Highest Temp ID (Quad-4)	0 to 10	
59	Q4NHIA	CET Next Highest Temperature Quad-4	32 to 2300	°F
60	Q4NIDA	CET Next Highest Temperature ID (Quad-4)	0 to 10	
61	CET26A P7	Core Exit Temperature P7	32 to 2300	°F
62(6)	CET3A E7 (N11)	Core Exit Temperature E7 (N11)	32 to 2300	°F
63	CET25A N10	Core Exit Temperature N10	32 to 2300	°F
64	CET24A N8	Core Exit Temperature N8	32 to 2300	°F
65	CET20A L6	Core Exit Temperature L6	32 to 2300	°F
66	CET7A K8	Core Exit Temperature K8	32 to 2300	°F
67	CET23A M3	Core Exit Temperature M3	32 to 2300	°F
68	CET18A H5	Core Exit Temperature H5	32 to 2300	°F
69	CET17A H3	Core Exit Temperature H3	32 to 2300	°F
70	CET14A G2	Core Exit Temperature G2	32 to 2300	°F
71	CET2A E4	Core Exit Temperature E4	32 to 2300	°F

PREFERRED ALARM STATUS BYTE CONFIGURATION

LSB	1 - HI		(High Limit Alarm)
	2 - LO		(Low Limit Alarm)
	3 - FAIL		(Failed Sensor)
	4 - BAD		(Bad Data - Out of Range)
	5 - SUSPCT		(Suspect Data)
	6 - QSPTRB		(QSPDS trouble)
	7 - SET TO 1		(To Avoid Confusion with GS)
MSB	8 - PARITY		(Odd Parity)

Explanations:

Failed Sensor - Equipment associated with the sensor has failed.

Bad Data - Sensor input is outside the valid range for the sensor.

Suspect Data - Calculated results which were affected/revised due to bad data or failed sensor being present.

The convention "1" = alarm/failed condition and "Ø" = normal/operational condition will be employed.

SIGNAL VALUE

Signal value can be any number represented by 1 to 8 ASCII characters.

Ex: 2000.2 is represented by 6 ASCII characters including the decimal point.

GROUP SEPARATOR

Group Separator (GS) is sent to the SAS to indicate the end of message packet. An acknowledge (ACK) or no acknowledge (NAK) ASCII character is sent to QSPDS by the SAS after every message packet. If an ACK is not received by the QSPDS, the message packet is retransmitted up to a maximum of two (2) times before declaring and tagging the data link as failed. The QSPDS will consider parity, framing, and overrun errors as NAKs in that the last data link transmission will be repeated following the above protocol.

5.2.2 Message Block Format

Message block consists of the message packets. Approximately every 1 to 2 seconds, QSPDS transmits the entire Message Block to the SAS. The Message Block has the following format.

STX	Message Packet	......	Message Packet N	ETX	CHK	EOT

The Message Block starts with start of text (STX) character, followed by message packets and ending with End of Text character (ETX), checksum (CHK, which is an Exclusive Or of all the data bytes between ETX and STX excluding the control characters GS) and End of Transmission (EOT) character.

6.0 DIAGNOSTIC TEST REQUIREMENTS

The QSPDS/SAS data link diagnostic checks shall be responsible for detecting serious failure of the data link hardware. This shall be accomplished by checking the status of the data link hardware and checking the number of NAKs (or incorrect responses) received consecutively from the SAS. If more than 3 NAKs (or incorrect responses) are received consecutively the data link between QSPDS and SAS is tagged as failed and the error condition is alarmed on the plasma display unit. When a failed data link is detected the transmission is stopped by the QSPDS for the present scan cycle. The transmission of data from the QSPDS to the SAS is restarted the next scan cycle. If a NAK/ACK is not received within 3 seconds after a message packet is sent, the data link is tagged as failed and alarmed on the plasma display unit. The QSPDS tries to establish communication again with SAS the next scan cycle. The QSPDS continuously searches for the operation of the data link every 3 seconds until the link becomes operational. The QSPDS will consider parity, framing, and overrun errors as NAKs in that the last data link transmission will be repeated following the above protocol.

CROSS REFERENCE TABLE

CHANNEL A (Continued)

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
49	Q2HIA	CET Highest Temp Quad-2	32 to 2300	°F
50	Q2HIDA	CET Highest Temp ID (Quad-2)	0 to 10	
51	Q2NHIA	CET Next Highest Temperature Quad-2	32 to 2300	°F
52	Q2NIDA	CET Next Highest Temperature ID (Quad-2)	0 to 10	
53	Q3HIA	CET Highest Temp Quad-3	32 to 2300	°F
54	Q3HIDA	CET Highest Temp ID (Quad-3)	0 to 10	
55	Q3NHIA	CET Next Highest Temperature Quad-3	32 to 2300	°F
56	Q3NIDA	CET Next Highest Temperature ID (Quad-3)	0 to 10	
57	Q4HIA	CET Highest Temp Quad-4	32 to 2300	°F
58	Q4HIDA	CET Highest Temp ID (Quad-4)	0 to 10	
59	Q4NHIA	CET Next Highest Temperature Quad-4	32 to 2300	°F
60	Q4NIDA	CET Next Highest Temperature ID (Quad-4)	0 to 10	
61	CET26A P7	Core Exit Temperature P7	32 to 2300	°F
62(6)	CET3A E7 (N11)	Core Exit Temperature E7 (N11)	32 to 2300	°F
63	CET25A N10	Core Exit Temperature N10	32 to 2300	°F
64	CET24A N8	Core Exit Temperature N8	32 to 2300	°F
65	CET20A L6	Core Exit Temperature L6	32 to 2300	°F
66	CET7A K8	Core Exit Temperature K8	32 to 2300	°F
67	CET23A M3	Core Exit Temperature M3	32 to 2300	°F
68	CET18A H5	Core Exit Temperature H5	32 to 2300	°F
69	CET17A H3	Core Exit Temperature H3	32 to 2300	°F
70	CET14A G2	Core Exit Temperature G2	32 to 2300	°F
71	CET2A E4	Core Exit Temperature E4	32 to 2300	°F

CROSS REFERENCE TABLE

CHANNEL A (Continued)

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
72	CET10A D3	Core Exit Temperature D3	32 to 2300	°F
73	CET15A G8	Core Exit Temperature G8	32 to 2300	°F
74	CET12A E10	Core Exit Temperature E10	32 to 2300	°F
75	CET11A D5	Core Exit Temperature D5	32 to 2300	°F
76	CET9A C12	Core Exit Temperature C12	32 to 2300	°F
77	CET8A C8	Core Exit Temperature C8	32 to 2300	°F
78	CET1A A8	Core Exit Temperature A8	32 to 2300	°F
79	CET22A L14	Core Exit Temperature L14	32 to 2300	°F
80	CET21A L12	Core Exit Temperature L12	32 to 2300	°F
81	CET6A J12	Core Exit Temperature J12	32 to 2300	°F
82	CET5A J10	Core Exit Temperature J10	32 to 2300	°F
83	CET19A H11	Core Exit Temperature H11	32 to 2300	°F
84	CET16A G15	Core Exit Temperature G15	32 to 2300	°F
85	CET13A F13	Core Exit Temperature F13	32 to 2300	°F
86	CET4A F11	Core Exit Temperature F11	32 to 2300	°F
87	TMARAA	Loop A RCS Temp Sat. Margin	-2100 to 700(1)	°F
88	PMARAA	Loop A RCS Press Sat. Margin	-3000 to 3000(1)	PSI
89	TMARBA	Loop B RCS Temp Sat. Margin	-2100 to 700(1)	°F
90	PMARBA	Loop B RCS Press Sat. Margin	-3000 to 3000(1)	PSI
91	TMARCA	Loop C RCS Temp Sat. Margin	-2100 to 700(1)	°F
92	PMARCA	Loop C RCS Press Sat. Margin	-3000 to 3000(1)	PSI
93	(3)	Reactor Vessel Level 1 through 8 Status Message Packet	0/1(3)	Coolant / No Coolant

NOTES TO CROSS REFERENCE TABLE

(1) + sign indicates subcooling.
- sign indicates superheat.

(2) Message Number is a 2 ASCII character number. It varies from 00 through 89.

(3) Reactor Vessel Level 1 through 8 Status Message

Data Byte

Odd Parity Bit MSB ... LSB

8	7	6	5	4	3	2	1

Byte #1

Bit 1	Reactor Vessel Level 1 (Coolant/No Coolant)
Bit 2	Reactor Vessel Level 2 (Coolant/No Coolant)
Bit 3	Reactor Vessel Level 3 (Coolant/No Coolant)
Bit 4	Reactor Vessel Level 4 (Coolant/No Coolant)
Bit 5	Reactor Vessel Level 5 (Coolant/No Coolant)
Bit 6	Reactor Vessel Level 6 (Coolant/No Coolant)
Bit 7	Set to "1"

NOTES TO CROSS REFERENCE TABLE (Continued)

Byte #2

Bit 1	Reactor Vessel Level 7 (Coolant/No Coolant)
Bit 2	Reactor Vessel Level 8 (Coolant/No Coolant)
Bit 3	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 4	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 5	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 6	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 7	Set to "1"

Ø indicates presence of coolant.

1 indicates absence of coolant or no coolant.

Bit 7 of these data bytes will be set to "1" (as shown) to avoid confusion which may arise by the SAS deciphering this byte as a 'group separator'.

(4) For dummy values, the integer format will be employed. An example is: ƀØ. Integer format is detailed in note 5.a.

(5) Format of Analog Values

a) Integer type: The field width is the size of the maximum range of the value plus 1 for a sign. Positive values have a blank in the sign position, negative values have a minus sign in the sign position. The numeric field is leading zero suppressed, replaced by blanks. If the value is zero, the right most position will contain a zero.

Example: If saturation margin, range +700 to -2100°F, is 50°F the transmitted data is ƀƀƀ50. If it is -10°F, the transmitted data is -ƀƀ10. If it is zero, the transmitted data is ƀƀƀƀ0.

; where ƀ is ASCII space (blank).

NOTES TO CROSS REFERENCE TABLE (Continued)

b) Exponential Format: Above a value of 10 and below a value of 1000, the integer format (described above) will be used. For the other values, the field width is 8 characters as follows:

a.aaa±bb, where a.aaa is the fractional part of the value and ±bb is the exponential part.

(Note: no sign information is transmitted since the data is always positive.)

For example: 1.23% power is transmitted as 0.123 + 01.

6) CET E7 is for Turkey Point Unit No. 3.

CET N11 is for Turkey Point Unit No. 4.

CROSS REFERENCE TABLE
CHANNEL B

MESSAGE NUMBER(2)	POINT ID	DESCRIPTION	VALUE(5) (GIVEN IN RANGE)	UNITS
00	THOT1B	Hot Leg Temp Loop A	0-750	°F
01	THOT2B	Hot leg Temp Loop B	0-750	°F
02	TCOLD1B	Cold Leg Temp Loop A	0-750	°F
03	TCOLD2B	Cold Leg Temp Loop B	0-750	°F
04	PRESSB	Pressurizer Pressure	0-3000	PSIA
05	THOT3B	Hot Leg Temp Loop C	0-750	°F
06	TCOLD3B	Cold Leg Temp Loop C	0-750	°F
07	THEADB	Upper Head Temp	32-2300	°F
08	TRCETB	Representative Core Exit Temperature	32-2300	°F
09	TMARHEADB	Upper Head Temperature Saturation Margin	-2100 to 700(1)	°F
10	PMARHEADB	Upper Head Pressure Saturation Margin	-3000 to 3000(1)	PSI
11	TMARRCSB	Minimum RCS Temperature Saturation Margin	-2100 to 700(1)	°F
12	PMARRCSB	Minimum RCS Pressure Saturation Margin	-3000 to 3000(1)	PSI
13	TMARCETB	Core Exit Temperature (CET) Saturation Margin	-2100 to 700(1)	°F
14	PMARCETB	CET Pressure Saturation Margin	-3000 to 3000(1)	PSI
15	TMARURB	RCS/Upper Head Temp Saturation Margin	-2100 to 700(1)	°F
16	RLEVHB	Reactor Vessel Level - Head	0 to 100	%
17	RLEVPB	Reactor Vessel Level - Outlet Plenum	0 to 100	%
18	DUMMY 1B	Dummy Value	0(4)	
19	TU1B	Unheated HJTC Temperature Level 1	32 to 2300	°F

CROSS REFERENCE TABLE

CHANNEL B

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
20	TU2B	Unheated HJTC Temperature Level 2	32 to 2300	°F
21	TU3B	Unheated HJTC Temperature Level 3	32 to 2300	°F
22	TU4B	Unheated HJTC Temperature Level 4	32 to 2300	°F
23	TU5B	Unheated HJTC Temperature Level 5	32 to 2300	°F
24	TU6B	Unheated HJTC Temperature Level 6	32 to 2300	°F
25	TU7B	Unheated HJTC Temperature Level 7	32 to 2300	°F
26	TU8B	Unheated HJTC Temperature Level 8	32 to 2300	°F
27	TH1B	Heated HJTC Temperature Level 1	32 to 2300	°F
28	TH2B	Heated HJTC Temperature Level 2	32 to 2300	°F
29	TH3B	Heated HJTC Temperature Level 3	32 to 2300	°F
30	TH4B	Heated HJTC Temperature Level 4	32 to 2300	°F
31	TH5B	Heated HJTC Temperature Level 5	32 to 2300	°F
32	TH6B	Heated HJTC Temperature Level 6	32 to 2300	°F
33	TH7B	Heated HJTC Temperature Level 7	32 to 2300	°F
34	TH8B	Heated HJTC Temperature Level 8	32 to 2300	°F

CROSS REFERENCE TABLE

CHANNEL B (Continued)

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
35	DT1B	Differential HJTC Temperature Level 1	-2268 to +2268	°F
36	DT2B	Differential HJTC Temperature Level 2	-2268 to +2268	°F
37	DT3B	Differential HJTC Temperature Level 3	-2268 to +2268	°F
38	DT4B	Differential HJTC Temperature Level 4	-2268 to +2268	°F
39	DT5B	Differential HJTC Temperature Level 5	-2268 to +2268	°F
40	DT6B	Differential HJTC Temperature Level 6	-2268 to +2268	°F
41	DT7B	Differential HJTC Temperature Level 7	-2268 to +2268	°F
42	DT8B	Differential HJTC Temperature Level 8	-2268 to +2268	°F
43	PC1B	Heater Power Control Signal 1	0 to 100	%
44	PC2B	Heater Power Control Signal 2	0 to 100	%
45	Q1HIB	CET Highest Temp Quad-1	32 to 2300	°F
46	Q1HIDB	CET Highest Temp ID (Quad-1)	0 to 10	
47	Q1NHIB	CET Next Highest Temperature Quad-1	32 to 2300	°F
48	Q1NIDB	CET Next Highest Temperature ID (Quad-1)	0 to 10	
49	Q2HIB	CET Highest Temp Quad-2	32 to 2300	°F
50	Q2HIDB	CET Highest Temp ID (Quad-2)	0 to 10	

CROSS REFERENCE TABLE

CHANNEL B (Continued)

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
51	Q2NHIB	CET Next Highest Temperature Quad-2	32 to 2300	°F
52	Q2NIDB	CET Next Highest Temperature ID (Quad-2)	0 to 10	
53	Q3HIB	CET Highest Temp Quad-3	32 to 2300	°F
54	Q3HIDB	CET Highest Temp ID (Quad-3)	0 to 10	
55	Q3NHIB	CET Next Highest Temperature Quad-3	32 to 2300	°F
56	Q3NIDB	CET Next Highest Temperature ID (Quad-3)	0 to 10	
57	Q4HIB	CET Highest Temp Quad-4	32 to 2300	°F
58	Q4HIDB	CET Highest Temp ID (Quad-4)	0 to 10	
59	Q4NHIB	CET Next Highest Temperature Quad-4	32 to 2300	°F
60	Q4NIDA	CET Next Highest Temperature ID (Quad-4)	0 to 10	
61	CET19B R7	Core Exit Temperature R7	32 to 2300	°F
62	CET18B P8	Core Exit Temperature P8	32 to 2300	°F
63	CET17B N6	Core Exit Temperature N6	32 to 2300	°F
64	CET25B N4	Core Exit Temperature N4	32 to 2300	°F
65	CET24B M11	Core Exit Temperature M11	32 to 2300	°F
66	CET16B M9	Core Exit Temperature M9	32 to 2300	°F
67	CET23B L8	Core Exit Temperature L8	32 to 2300	°F
68	CET14B K5	Core Exit Temperature K5	32 to 2300	°F
69	CET13B K3	Core Exit Temperature K3	32 to 2300	°F
70	CET12B J2	Core Exit Temperature J2	32 to 2300	°F
71	CET9B G6	Core Exit Temperature G6	32 to 2300	°F
72	CET8B G1	Core Exit Temperature G1	32 to 2300	°F
73	CET6B F5	Core Exit Temperature F5	32 to 2300	°F

CROSS REFERENCE TABLE

CHANNEL B (Continued)

MESSAGE NUMBER	POINT ID	DESCRIPTION	VALUE (GIVEN IN RANGE)	UNITS
74	CET5B F3	Core Exit Temperature F3	32 to 2300	°F
75	CET10B H8	Core Exit Temperature H8	32 to 2300	°F
76	CET7B F9	Core Exit Temperature F9	32 to 2300	°F
77	CET20B E8	Core Exit Temperature E8	32 to 2300	°F
78	CET2B B10	Core Exit Temperature B10	32 to 2300	°F
79	CET1B B5	Core Exit Temperature B5	32 to 2300	°F
80	CET15B K11	Core Exit Temperature K11	32 to 2300	°F
81	CET11B H15	Core Exit Temperature H15	32 to 2300	°F
82	CET22B H13	Core Exit Temperature H13	32 to 2300	°F
83	CET21B H9	Core Exit Temperature H9	32 to 2300	°F
84	CET4B E14	Core Exit Temperature E14	32 to 2300	°F
85	CET3B E12	Core Exit Temperature E12	32 to 2300	°F
86	DUMMY 2B	Dummy Value		
87	TMARAB	Loop A RCS Temp Sat. Margin	-2100 to 700(1)	°F
88	PMARAB	Loop A RCS Press Sat. Margin	-3000 to 3000(1)	PSI
89	TMARBB	Loop B RCS Temp Sat. Margin	-2100 to 700(1)	°F
90	PMARBB	Loop B RCS Press Sat. Margin	-3000 to 3000(1)	PSI
91	TMARCB	Loop C RCS Temp Sat. Margin	-2100 to 700(1)	°F
92	PMARCB	Loop C RCS Press Sat. Margin	-3000 to 3000(1)	PSI
93	(3)	Reactor Vessel Level 1 through 8 Status Message Packet	0/1(3)	Coolant / No Coolant

NOTES TO CROSS REFERENCE TABLE

(1) + sign indicates subcooling.
- sign indicates superheat.

(2) Message Number is a 2 ASCII character number. It varies from 00 through 89.

(3) Reactor Vessel Level 1 through 8 Status Message

Data Byte

Odd
Parity Bit
MSB | LSB

8	7	6	5	4	3	2	1

Byte #1

Bit 1	Reactor Vessel Level 1 (Coolant/No Coolant)
Bit 2	Reactor Vessel Level 2 (Coolant/No Coolant)
Bit 3	Reactor Vessel Level 3 (Coolant/No Coolant)
Bit 4	Reactor Vessel Level 4 (Coolant/No Coolant)
Bit 5	Reactor Vessel Level 5 (Coolant/No Coolant)
Bit 6	Reactor Vessel Level 6 (Coolant/No Coolant)
Bit 7	Set to "1"

NOTES TO CROSS REFERENCE TABLE (Continued)

Byte #2

Bit 1	Reactor Vessel Level 7 (Coolant/No Coolant)
Bit 2	Reactor Vessel Level 8 (Coolant/No Coolant)
Bit 3	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 4	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 5	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 6	Set to "Ø" by QSPDS - should be ignored by SAS computer.
Bit 7	Set to "1"

Ø indicates presence of coolant.

1 indicates absence of coolant or no coolant.

Bit 7 of these data bytes will be set to "1" (as shown) to avoid confusion which may arise by the SAS deciphering this byte as a 'group separator'.

(4) For dummy values, the integer format will be employed. An example is: ƀØ. Integer format is detailed in note 5.a.

(5) Format of Analog Values

a) Integer type: The field width is the size of the maximum range of the value plus 1 for a sign. Positive values have a blank in the sign position, negative values have a minus sign in the sign position. The numeric field is leading zero suppressed, replaced by blanks. If the value is zero, the right most position will contain a zero.

Example: If saturation margin, range +700 to -2100°F, is 50°F the transmitted data is ƀƀƀ50. If it is -10°F, the transmitted data is -ƀƀ10. If it is zero, the transmitted data is ƀƀƀƀ0.

where ƀ is ASCII space (blank).

b) Exponential Format: Above a value of 10 and below a value of 1000, the integer format (described above) will be used. For the other values, the field width is 8 characters as follows:

a.aaa$\pm$bb, where a.aaa is the fractional part of the value and $\pm$bb is the exponential part.

(Note: no sign information is transmitted since the data is always positive.)

For example: 1.23% power is transmitted as 0.123 + 01.

ATTACHMENT A

PRESENT PLANT STATUS

TURKEY POINT UNITS 3 & 4

VESSEL STATUS JANUARY 1, 1983

- CRITICAL MATERIAL   INTERMEDIATE TO LOWER SHELL WELD
- RT_{NDT} PTP3   263 ^{O}F
  PTP4   264 ^{O}F
- DATE PTP UNITS WILL EXCEED SCREENING CRITERIA (USING 1ST 8 CYCLE AVERAGE)
  PTP3   MID 1989
  PTP4   LATE 1989
- RT_{NDT} RATE OF INCREASE
  7 ^{O}F/EFPY

TURKEY POINT UNITS 3 & 4

BASIS FOR RT_{NDT} CALCULATION

$$RT_{NDT} = RT_O + \Delta RT + 2\sigma \text{ TERM}$$

RT_O	=	0° F
ΔRT	=	GUTHRIE
2σ TERM	=	59° F
% Cu	=	0.32%
% Ni	=	0.57%
CAPACITY FACTOR	=	80%
PTP 3 EFPY	=	6.3
PTP 3 FLUENCE	=	1×10^{19} N/cm^2
PTP 4 EFPY	=	6.35
PTP 4 FLUENCE	=	1.02×10^{19} N/cm^2

ATTACHMENT B

VESSEL FLUX REDUCTION PROGRAM

Table of Contents

1. Purpose and Objective

The Turkey Point nuclear units are the most economical power plants owned by Florida Power & Light. As such, these units are good candidates for extending their operating lifetime beyond current license life.

The present objective of the flux reduction program is to reduce the fast neutron flux at the vessel surface sufficiently to allow operation to at least the licensed lifetime. To achieve this objective, changes to core designs are anticipated to substantially reduce vessel flux. Fuel management analyses are underway and quantitative vessel fluence analyses are planned to determine the best means of reaching the sufficient flux reduction condition.

2. Dimension of Flux Reduction Requirements

The Turkey Point pressure vessels have only circumferential welds with a screening criteria of 300°F RT_{NDT}. This corresponds to a limiting fluence in the most limiting weld of 1.85×10^{19} n/cm^2. The last reviewed submittal (August 31, 1982) quantified the radially dependent flux level in the critical weld as depicted in Figure 2.1 for the "8 Cycle Average."

The time dimension of the flux reduction requirement is defined by the need to reach licensed lifetime (year 2007) and the potential desire to reach a later year such as 2015. This implies 19.2 effective-full-power-years (EFPY) and 25.6 EFPY of further operation, respectively, beyond January 1983. The fluence to date is about 1×10^{19} n/cm^2 for both units after 6.37 EFPY of operation. Table 2.1 provides a summary of the current status of both units.

The axial spatial dependence of needed flux reduction can be seen by referring to the axial cross-section of the vessel illustrated in Fig. 2.2. The limiting weld is about five feet above the bottom of the active core and is about 16" from the nearest assemblies in the core at the N-S and E-W axes. The fluence in the base vessel material will not be limiting compared to the weld because of its considerably lower copper content. These factors lead to the need to reduce the source of fast neutrons from the core to an area extending about 1½ feet above and below the weld elevation.

The radial dimension of the required flux reduction is presented in Fig. 2.3. To reach currently licensed lifetime, some flux reduction must occur over an angle of about $\pm 15^\circ$ about the core axes. Referring to a radial cross-section of the vessel, Fig. 2.4, flux reduction to a length of 40" of the weld about

each of the axes is necessary. Visual inspection of assembly placement reveals that all twelve assemblies on the core "flats" must reduce their source of fast neutrons. The same inspection leads to the observation that no other assemblies are nearly as important to the needed flux reduction. Assemblies near the core diagnonals and at the core edge could even be allowed to increase their source substantially.

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TURKEY POINT

CURRENT STATUS (1/1/83)

	FLUENCE (10^{19} N/CM^2)	RTNDT (°F)
UNIT 3	1.00	263
UNIT 4	1.02	264
SCREENING CRITERIA	1.85	300

TABLE 2.1

FLORIDA POWER & LIGHT CO.
TURKEY POINT UNIT 4
FAST FLUX vs AZIMUTHAL ANGLE

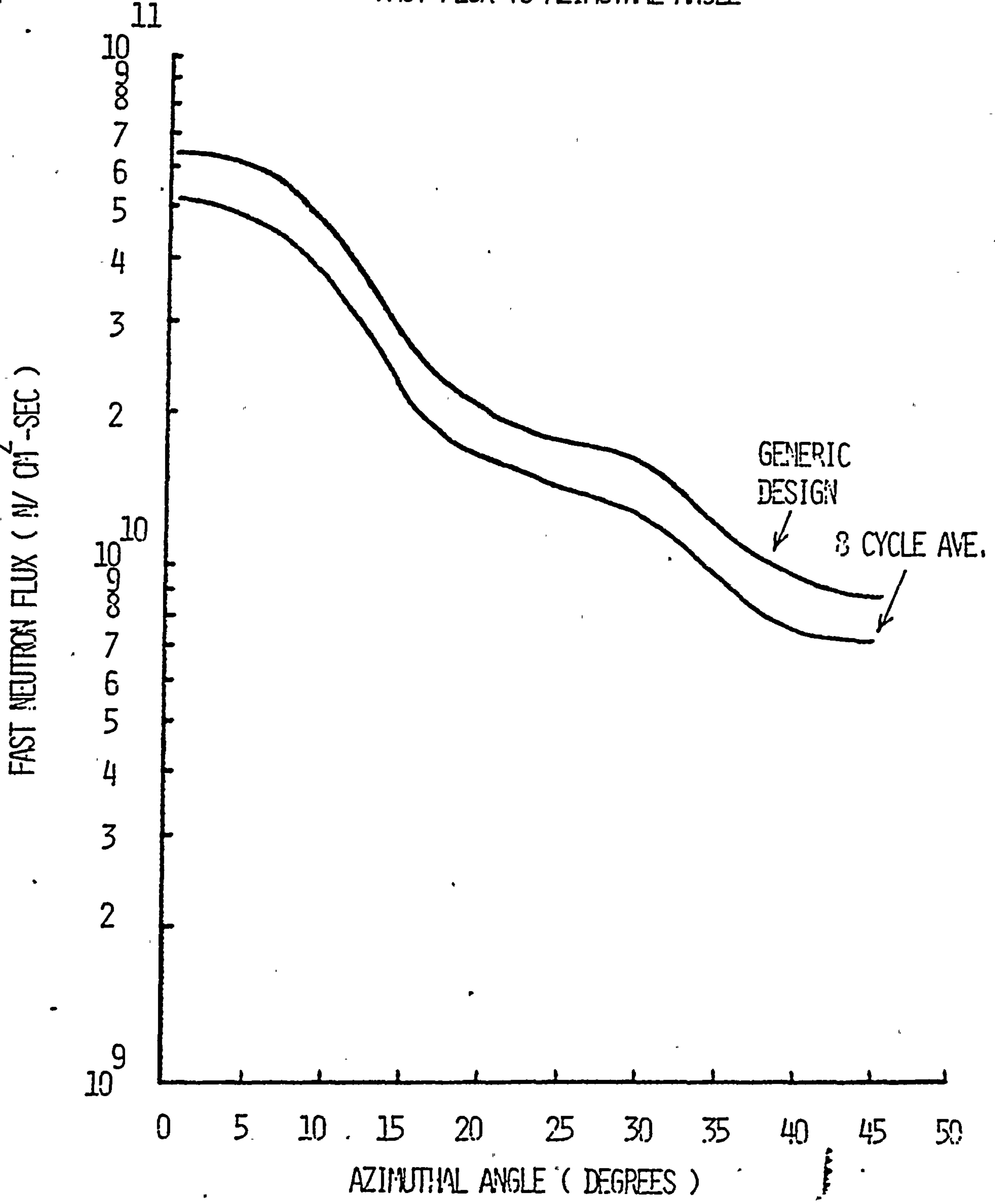

FIGURE 2.1

FLORIDA POWER & LIGHT CO.

TURKEY POINT

REACTOR VESSEL AXIAL CROSS SECTION

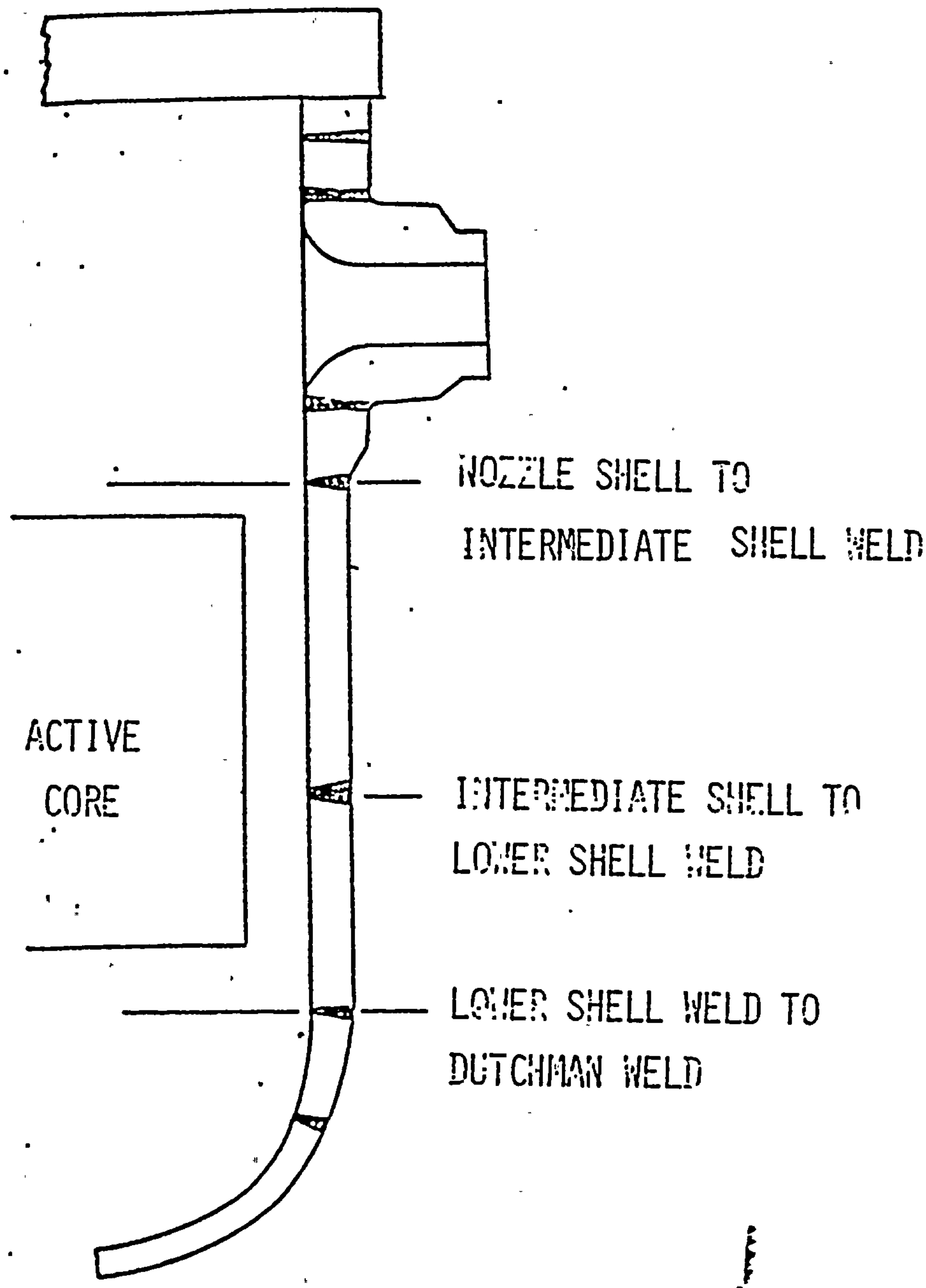

FIGURE 2.2

FLORIDA POWER & LIGHT CO.
TURKEY POINT UNIT 4
REDUCTION FACTOR VS. ANGLE

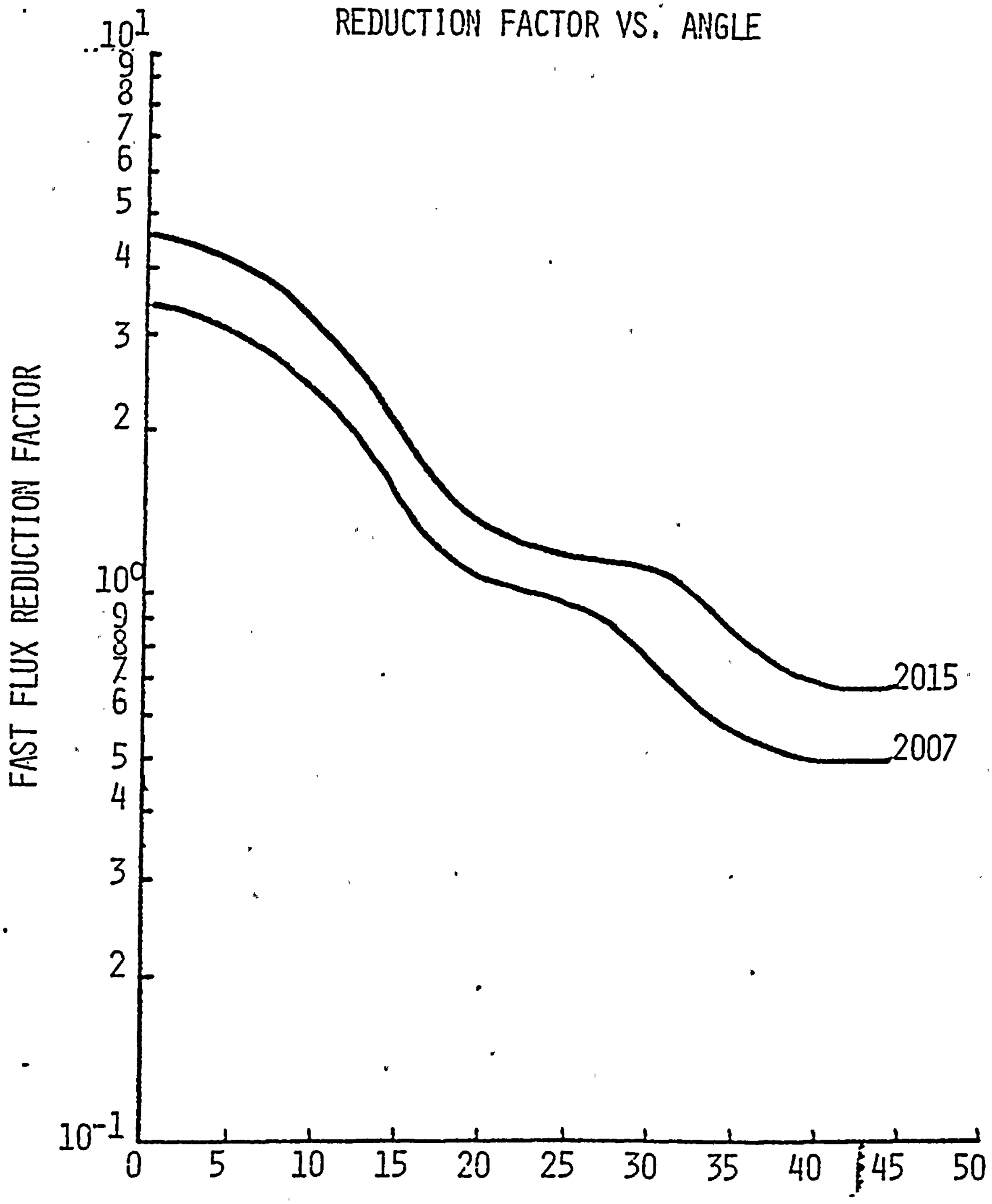

FIGURE 2.3

FLORIDA POWER & LIGHT CO.
TURKEY POINT
REACTOR CORE CROSS SECTION

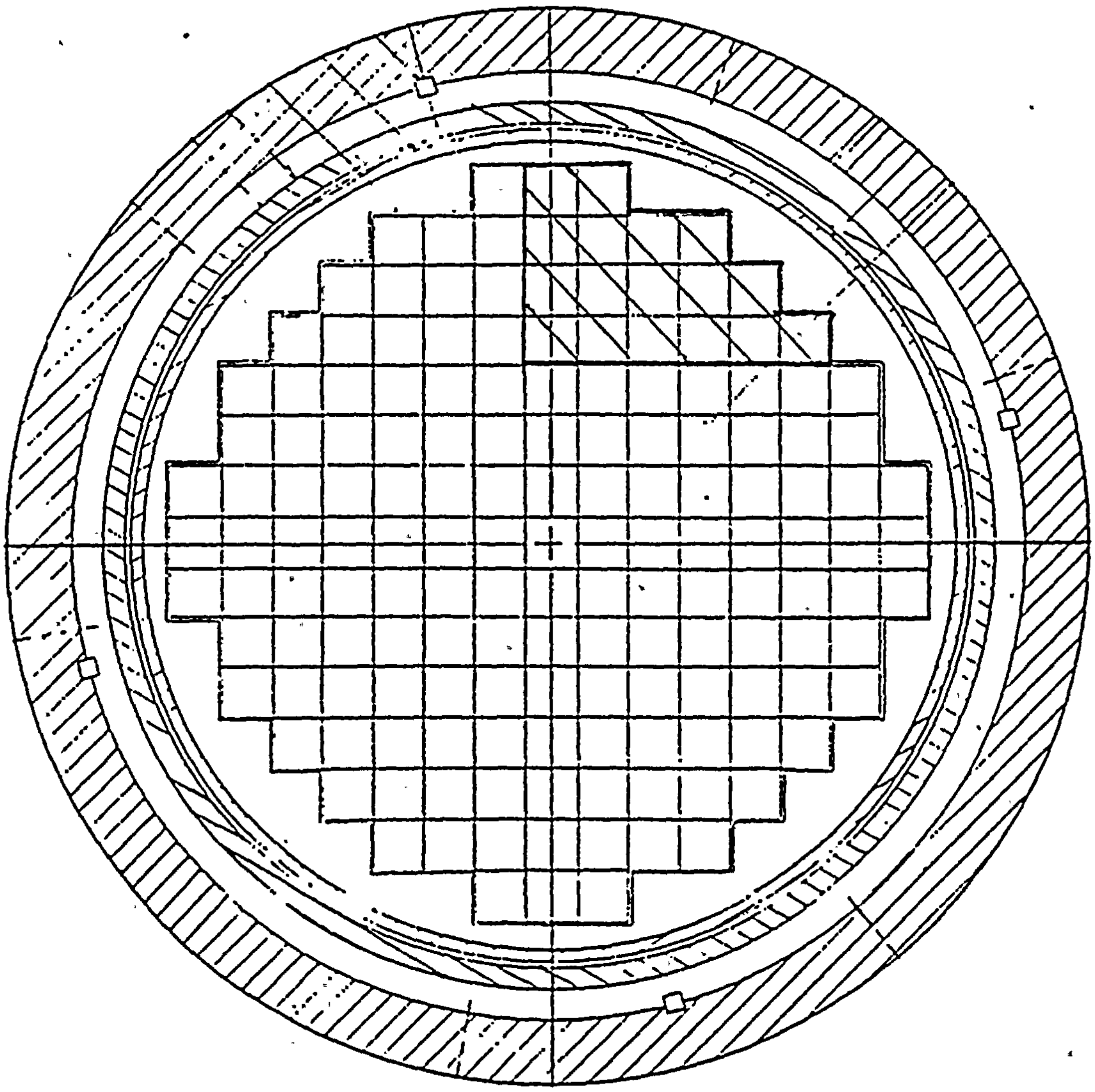

FIGURE 2.4

3. Turkey Point Operating History and Plans

Since startup (Unit 3 = 1972, Unit 4 = 1973), both units have operated on annual cycles (with one exception for each plant). Half of these cycles used conventional fuel management (fresh fuel on the periphery) and the other half used "standard" low-leakage fuel management.

As of January 1, 1983, Unit 4 has accumulated 6.37 (Figure 3.1) EFPY and Unit 3 slightly less. Subsequent to steam generator replacement at both units, 18 month operating cycles are planned. Annual cycles will be used only when contingencies necessitate it. Figure 3.2 illustrates this schedule. Unit 3 Cycle 8 started up in April 1982 and is an eighteen month cycle. As of this date (February 8, 1983), an annual Cycle 9 for Unit 4 is intended because of schedular constraints.

Use of 18 month cycles and a planning basis 91% capacity factor between refueling results in approximately an 80% total capacity factor. This factor is used in any discussions of EFPY and calendar dates.

This historical operation of the Turkey Point nuclear units along with "generic" core radial and axial power distributions were previously used to quantify the vessel fluence. The generic power distribution places the axial power peak at the critical weld. The generic radial power distribution is given in Fig. 3.3.

The "8 Cycle Average" Turkey Point specific calculation of fluence used a revised radial power distribution, also provided for Unit 4 in Fig. 3.3. Shown in Fig. 3.4 is the "8 Cycle Average" radial power distribution illustrating the

equivalence of Units 3 and 4.

The axial power specific to Turkey Point has not been accounted for, however. Inspection of the actual axial powers on the core flats leads to the estimate of a 4% lower accumulated fluence at the critical weld than the results provided in August 1982. This reduction in fluence to date is referred to in subsequent discussions of needed flux reductions.

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TURKEY POINT UNIT 4

ACCUMULATED BURNUP VS. YEAR

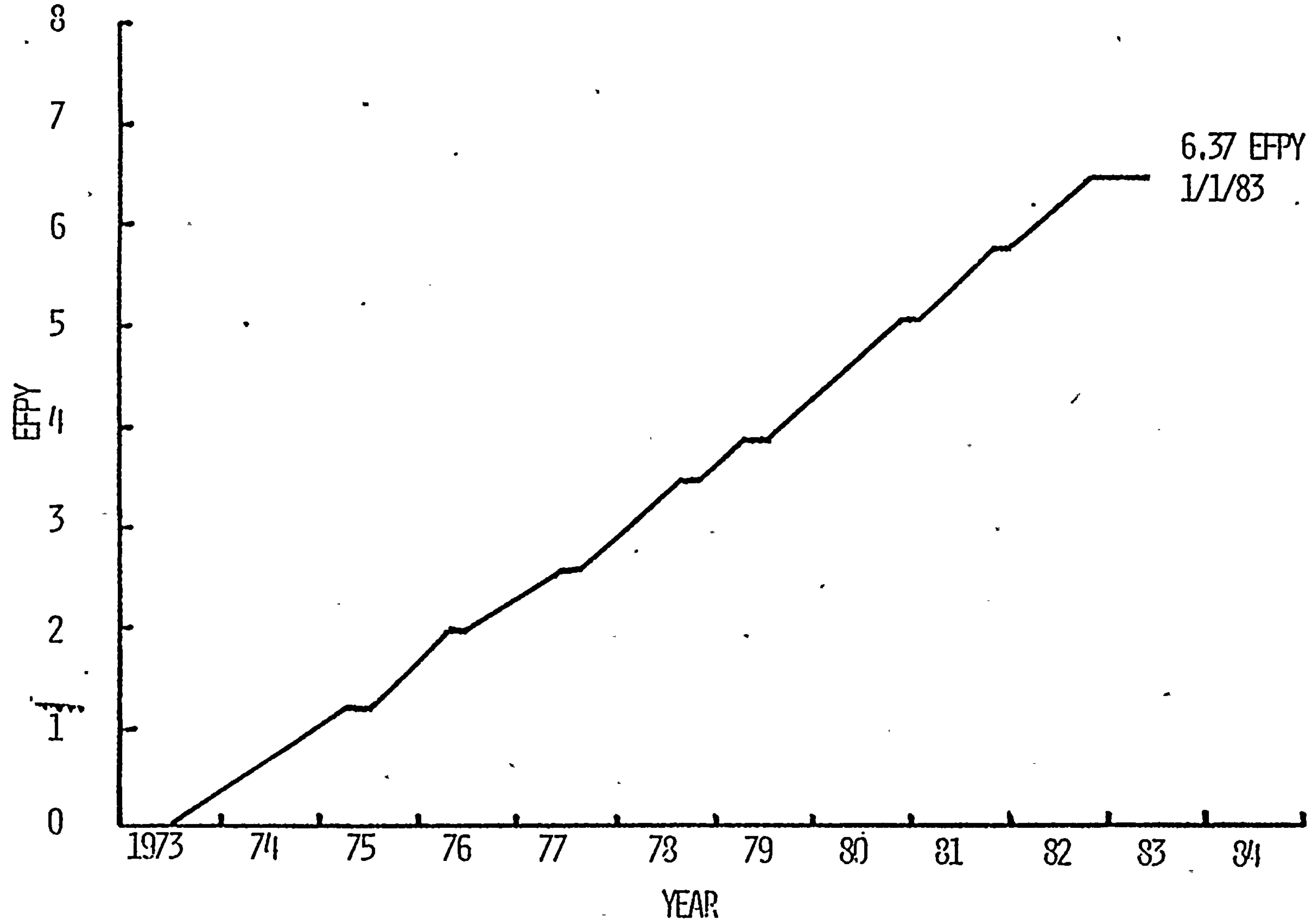

FIGURE 3.1

FLORIDA POWER & LIGHT CO.
TURKEY POINT OPERATING SCHEDULE
(AS OF JANUARY 1, 1983)

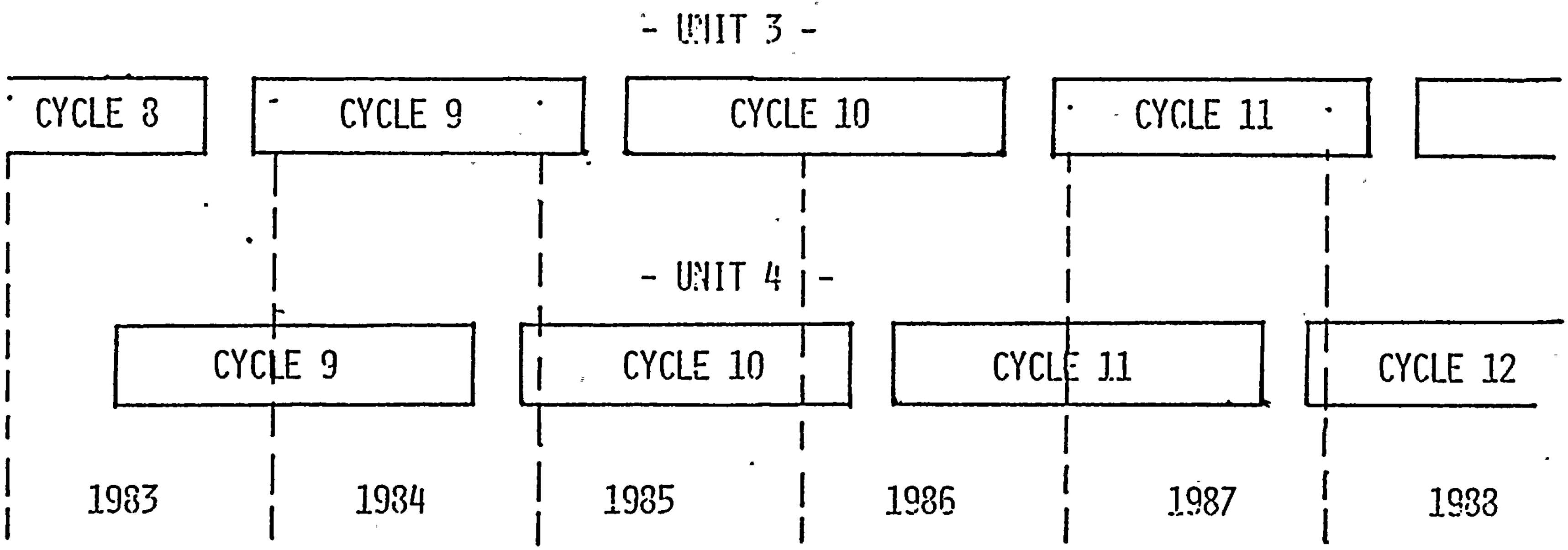

FIGURE 3.2

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TURKEY POINT

DESIGN BASIS PERIPHERAL POWER DISTRIBUTION

CORE FLAT →

.93	.77				
	1.07	1.12	.80		
			1.04	.85	
				.92	

TURKEY POINT UNIT 4

8 CYCLE AVERAGE PERIPHERAL POWER DISTRIBUTION

.73	.62				
	1.17	.95	.56		
			1.10	.64	
				1.02	

FIGURE 3.3

FLORIDA POWER & LIGHT CO.

TURKEY POINT UNIT 3

8 CYCLE AVERAGE PERIPHERAL POWER DISTRIBUTION

.75	.60				
	1.16	.98	.52		
			1.10	.60	
				.98	

TURKEY POINT UNIT 4

8 CYCLE AVERAGE PERIPHERAL POWER DISTRIBUTION

.73	.62				
	1.17	.95	.56		
			1.10	.64	
				1.02	

FIGURE 3.4

4. Flux Reductions Achieved To Date

In late 1981, the Pressurized Thermal Shock (PTS) issue became a serious concern with respect to fuel management because of the vessel flux limitations which would be a part of PTS. Flux reduction for the next reloads for each Unit were given attention even though quantitative flux targets were not yet known. Attempting to err on the side of prudence, the Unit 4 Cycle 9 reload was specified in March 1982 with a "modified low-leakage" loading pattern. At that time, the planned startup of Cycle 9 was June 1983. The annual Cycle 9 "backup" design was also set with the same approach and achieves greater flux reduction than the eighteen month cycle presented in this section. Similarly, in July 1982 the Unit 3 Cycle 9 design used "modified low-leakage" (planned startup December 1983).

The "modified low-leakage" is feasible within existing operating margins. The predicted radial power distributions (cycle average) for Cycle 9 of both Units is provided in Fig. 4.1. It is anticipated that these designs provide almost a factor of two reduction over the "8 Cycle Average." The impact of this reduction in light of the now known target fluence, is illustrated in Fig. 4.2.

If the Turkey Point Units had operated since initial criticality with conventional fuel management, stainless steel dummy assemblies would need to be implemented now in order to stay below the RT_{NDT} screening criteria at licensed lifetime. The drop-dead date for dummy assemblies based on the "8 Cycle Average" flux level is 1986. With modified low-leakage, dummy assemblies would not be necessary until 1990. These projections assume that use of stainless steel dummy assemblies in all twelve core flat positions

achieve a factor eight flux reduction relative to the generic power distributions.

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TURKEY POINT UNIT 3

CYCLE 9 PERIPHERAL POWER DISTRIBUTION

.50	.42				
	1.16	.98	.48		
			1.15	.46	
				1.00	

TURKEY POINT UNIT 4

CYCLE 9 PERIPHERAL POWER DISTRIBUTION

.41	.42				
	1.12	.92	.42		
			1.04	.40	
				.83	

FIGURE 4.1

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TURKEY POINT UNIT 4

VESSEL FLUENCE VS. VESSEL LIFE

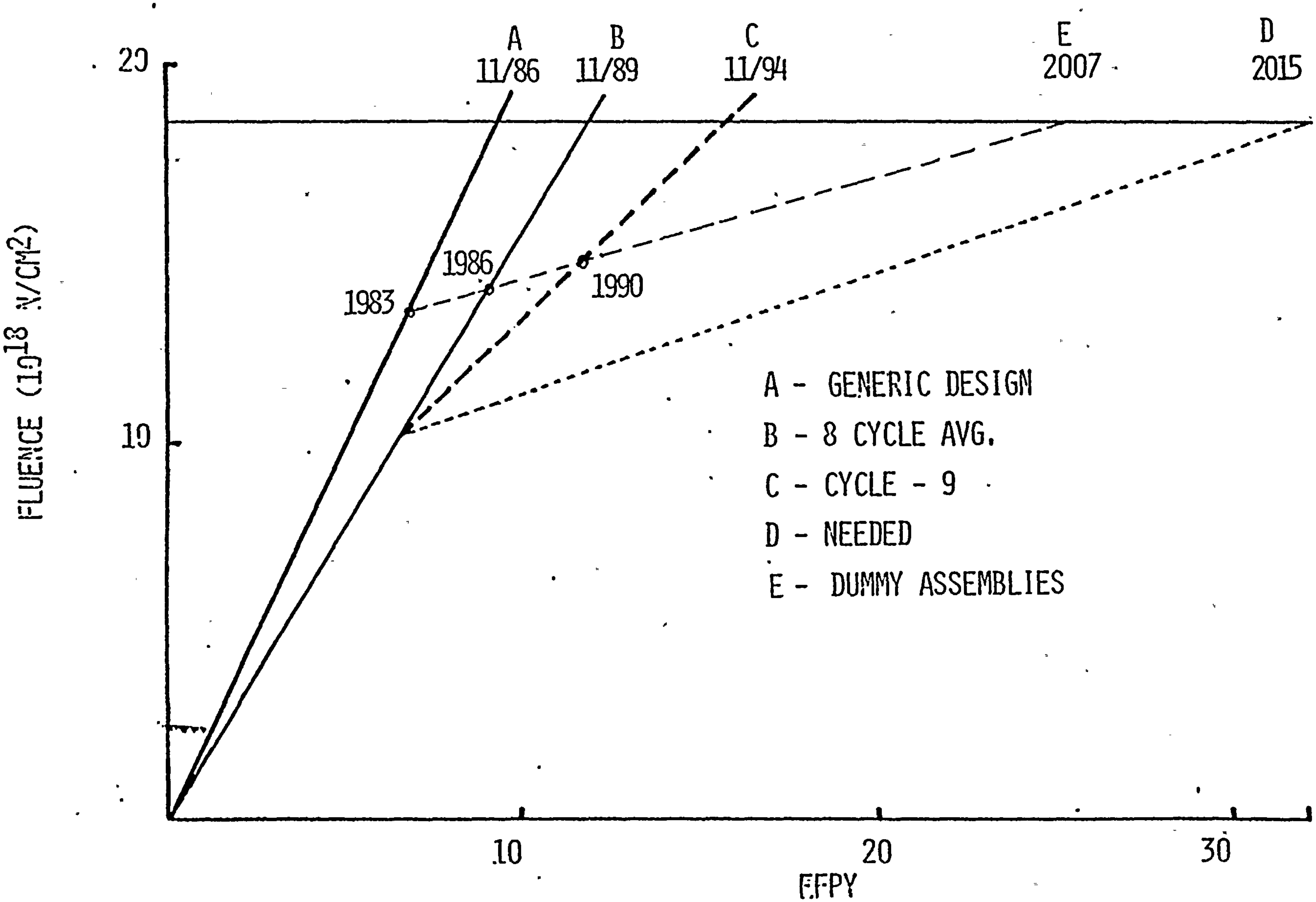

FIGURE 4.2

5. Near Term Flux Reduction Plans

In the second-half of 1982, with the establishment of the screening criteria, the limiting fluence became known and flux reduction became more urgent. Because materials were already in process for the next reloads, further modifications to the Cycle 9 designs were evaluated which did not entail change to the fuel loading. Time constraints limited changes to the Unit 4 Cycle 9 design to those which fell within existing operating margins.

As will be seen in subsequent sections of this report, increases in operating margin are required for Unit 3 in time to allow more extensive changes in its Cycle 9 design. The annual Cycle 9 Unit 4 design now has no time to be changed but has a radial power of 0.32 on the core flats which is about the same as modifications to the 18 month cycle could have achieved. As a general point, annual cycles can achieve lower vessel flux levels because of the greater inherent operating margin to LOCA and DNB limits. The lower number of feed assemblies increases the designers flexibility in shifting power away from the core flats.

The switch to the annual Unit 4 Cycle 9 has caused the Cycle 10 reload to start the design process now. This design assumes increased operating margins and will implement flux reduction features described in this section. Cycle 10 is now planned to start in May 1984 and will be an 18 month cycle.

A portion of the design flexibility associated with annual cycles can be obtained by moving to higher assembly discharge burnups (fewer feed assemblies). Achievement of high burnups and NRC approval of the high burnup topicals submitted by the fuel vendors in 1982 is seen as a high

priority with respect to flux reduction.

The Unit 4 18 month Cycle 9 design was used for the near-term flux reduction fuel management studies. Conclusions resulting from these studies are generally applicable to any 18 month Turkey Point cycle.

Figure 5.1 summarizes the anticipated current magnitude of flux reduction. The previous Cycle 9 design, and using equivalent core designs in the future, would cause the screening criteria to be reached in August 1995. Switching to dummy assemblies would be needed eight years from now if no other actions were to be taken. Translating these limitations to flux, Fig. 5.2 illustrates the flux levels versus azimuthal angle which cannot be exceeded (on the average) to avoid reaching the screening criterion. These flux limits assume the 4% reduction in historical flux level due to the corrected axial shape.

Even with increases in operating margin, the time required to implement exotic assembly designs or materials constrain the near term solutions to "off-the-shelf" materials and standard assembly designs. The options considered for near term implementation on the core flats were spent fuel (lowest reactivity), fresh full or part length burnable absorbers, part length control rods installed on burnable poison spiders, and assemblies containing natural or depleted uranium.

The radial power impact of the two most simple changes compared to the previous Ccyle 9 design are provided in Figs. 5.3 and 5.4. The case of low reactivity fuel and burnable poisons is anticipated to achieve the majority of

needed flux reductions. The burnable poisons (Fig. 5.4) used in the study were full length. The small axial extent of needed flux reduction, however, indicates that part length poison rods can be just as effective with a lesser decrease in overall radial power. Part length BPs would, therefore, assist in mitigating the loss in operating margin for a given level of flux reduction.

The impact of the near term design changes on the axial power shapes is illustrated in Fig. 5.5. The use of spent fuel on the core flats has a large advantage compared to the generic power shape by shifting the powers upwards, away from the critical weld in addition to the expected reduction in axial peaking. This factor results in about a 10% decrease in critical weld flux in addition to the decrease in radial power.

Combining the radial powers and the axial shapes results in the powers plotted in Fig. 5.6. The expected impact of implementing these changes is given in Fig. 5.7. The design changes planned for Cycle 9 of Unit 3 and Cycle 10 of Unit 4 correlate with Curve C on Fig. 5.7 which indicates that the screening criterion would be reached in August 2004. Assuming no further changes, dummy assemblies could be used beginning in 2001 to reach licensed lifetime.

These changes, however, are not without penalty. Increases in hot spot peaking (<u>FQ</u>) and radial channel peaking (<u>F∆H</u>) are expected. In addition, compared to designs without these changes, core reactivity is lost. In future cycles, this will be recovered by increasing the amount of U-235 loaded in the core. These penalties are summarized in Table 5.1. Table 5.2 lists the expected RT_{NDT} values associated with the near term design changes.

Florida Power & Light intends to implement the most effective of these design changes. Near-term approvals, however, of topicals, technical specification changes and licensing analyses are required by third quarter 1983 for the following items.

- High-burnup topical
- Enrichment limit on fuel storage
- Analyses for higher $F_{\Delta}H$ operating limit
- Analyses for higher LOCA (F_Q) operating limit.

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TURKEY POINT UNIT 4

VESSEL FLUENCE VS. VESSEL LIFE

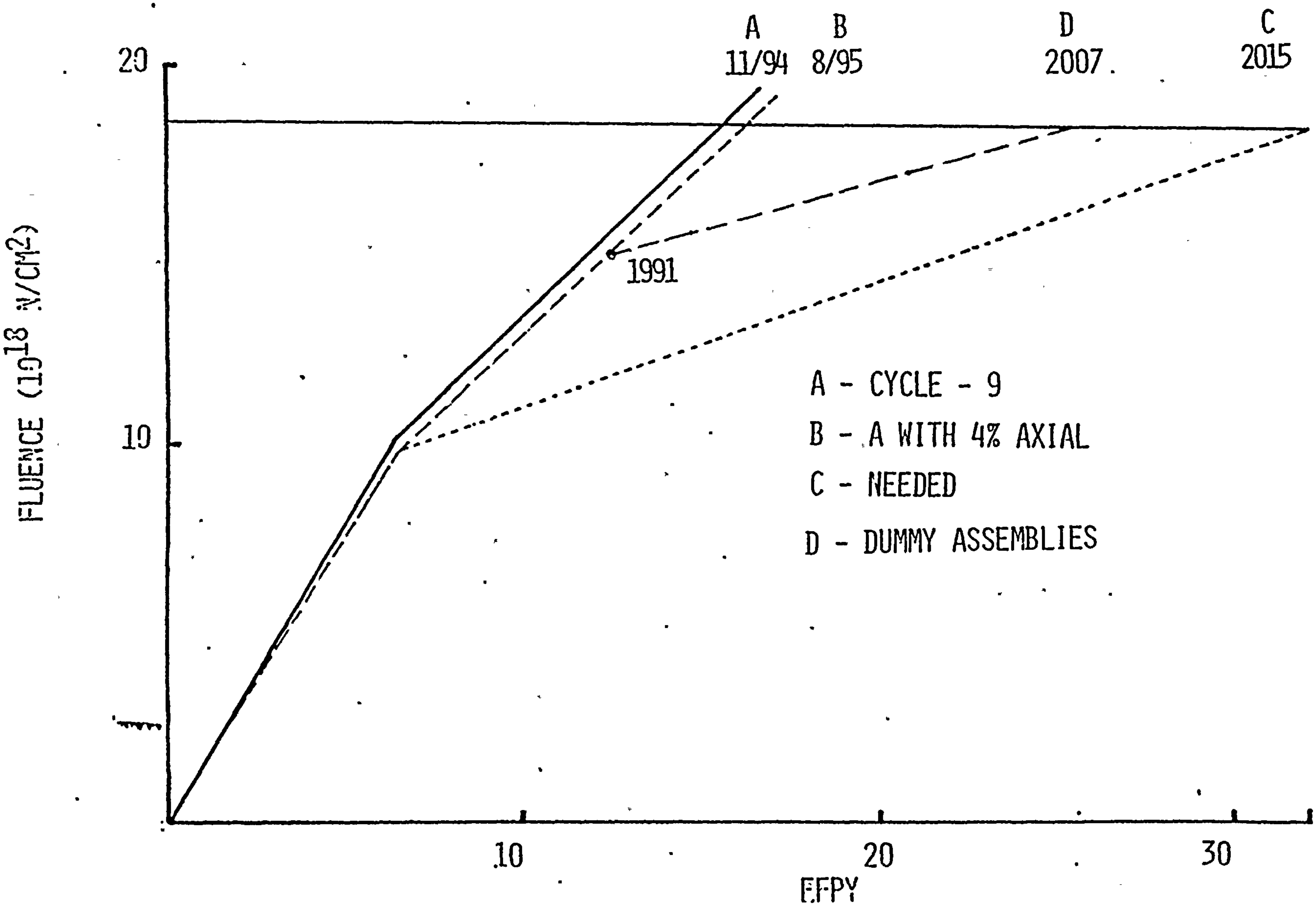

FIGURE 5.1

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TURKEY POINT UNIT 4

FAST FLUX vs AZIMUTHAL ANGLE

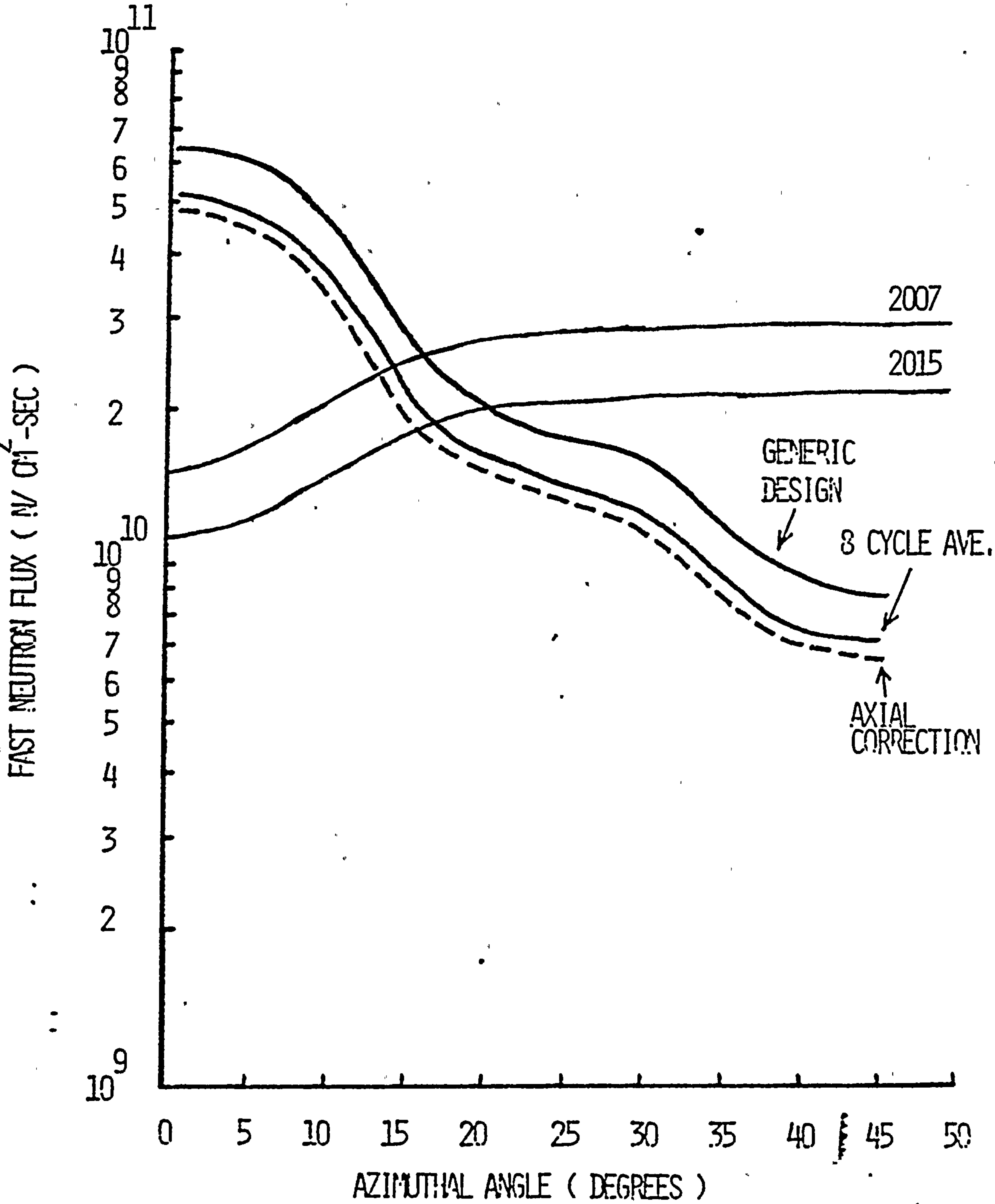

FIGURE 5.2

TURKEY POINT UNIT 4

CYCLE 9 PERIPHERAL POWER DISTRIBUTION

.41	.42				
	1.12	.92	.42		
			1.04	.40	
				.83	

CASE A

HIGHLY BURNT ASSEMBLIES

.29	.27				
	1.12	.90	.42		
			1.10	.42	
				.88	

FIGURE 5.3

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TURKEY POINT UNIT 4

CYCLE 9 PERIPHERAL POWER DISTRIBUTION

.41	.42				
	1.12	.92	.42		
			1.04	.40	
				.83	

CASE B

HIGHLY BURNT ASSEMBLIES + BPS

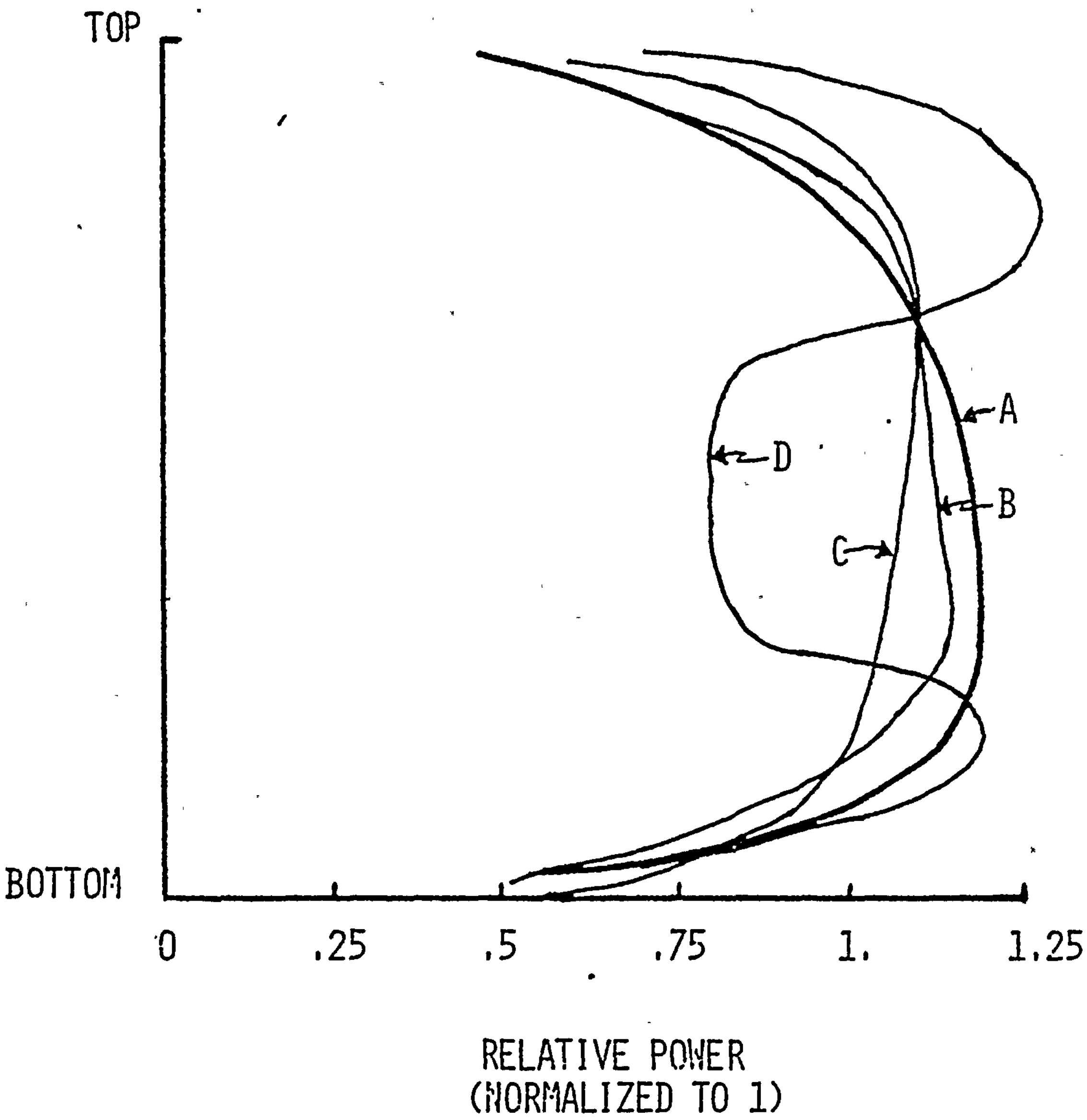
.23	.21				
	1.10	.90	.42		
			1.11	.42	
				.89	

FIGURE 5.4

FLORIDA POWER & LIGHT CO.

TURKEY POINT UNIT 4

PERIPHERAL AXIAL POWER SHAPE



A - GENERIC

B - ACTUAL 8 CYCLE AVERAGE

C - SPENT FUEL

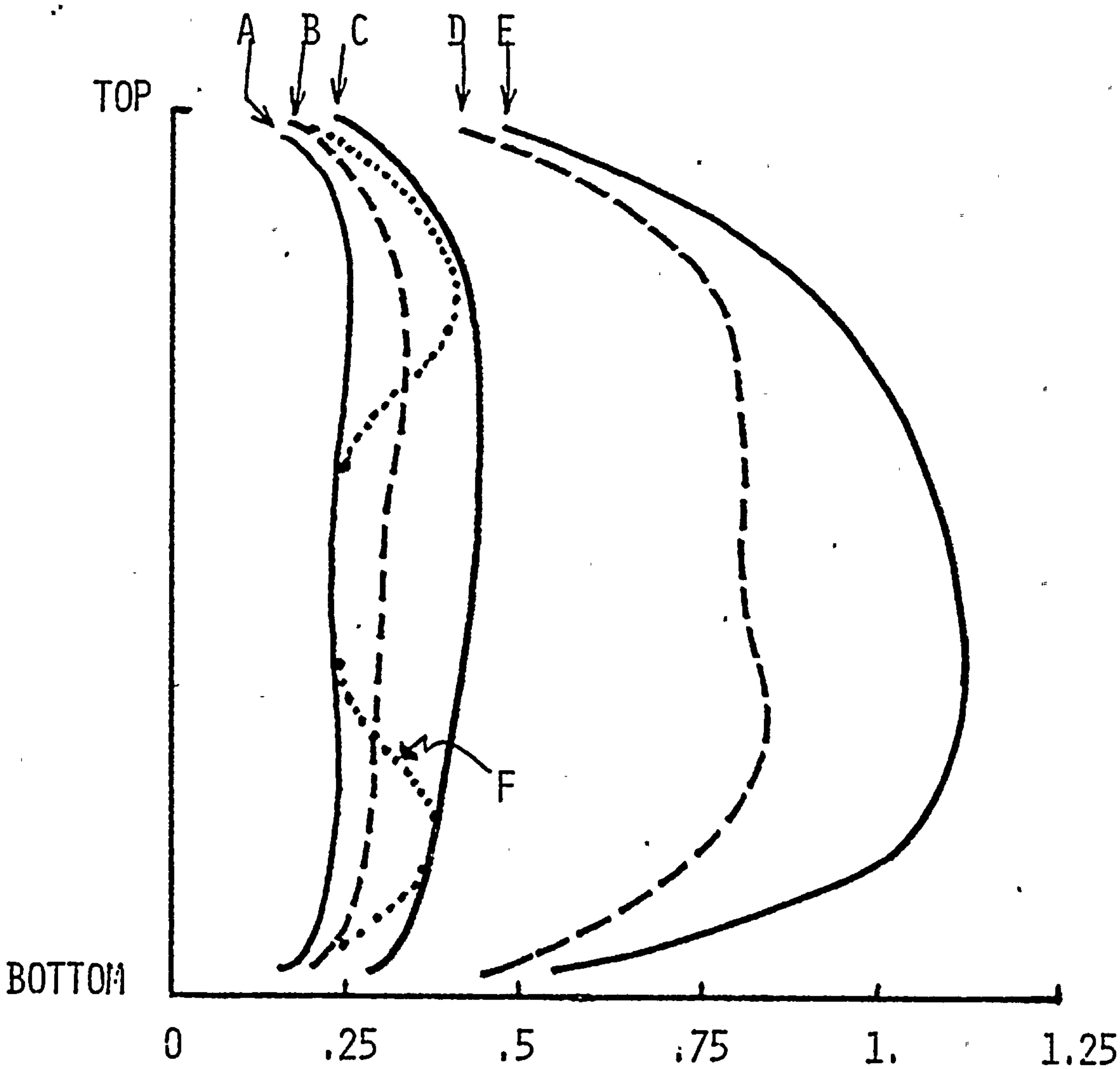
D - SPENT FUEL + PLBP

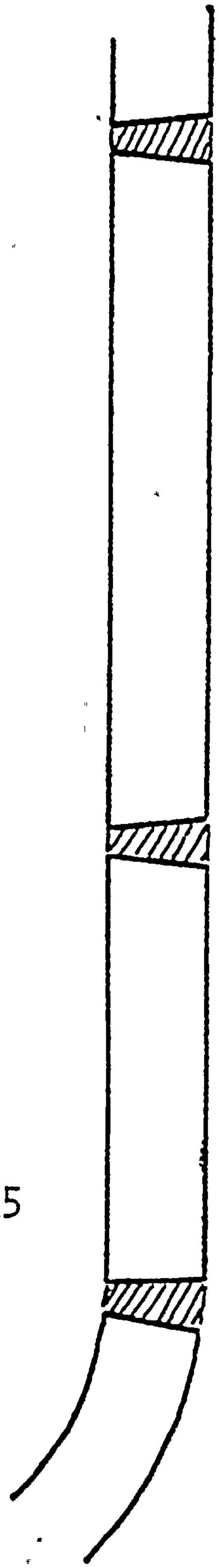
FIGURE 5.5

FLORIDA POWER & LIGHT CO.

TURKEY POINT UNIT 4

PERIPHERAL ASSEMBLY POWERS





RELATIVE POWER

A - SPENT FUEL + BP

B - SPENT FUEL

C - CYCLE 9 DESIGN

D - 8 CYCLE AVG.

E - GENERIC

F - SPENT FUEL + PLBP

FIGURE 5.6

FLORIDA POWER & LIGHT CO.
TURKEY POINT UNIT 4
VESSEL FLUENCE VS. VESSEL LIFE

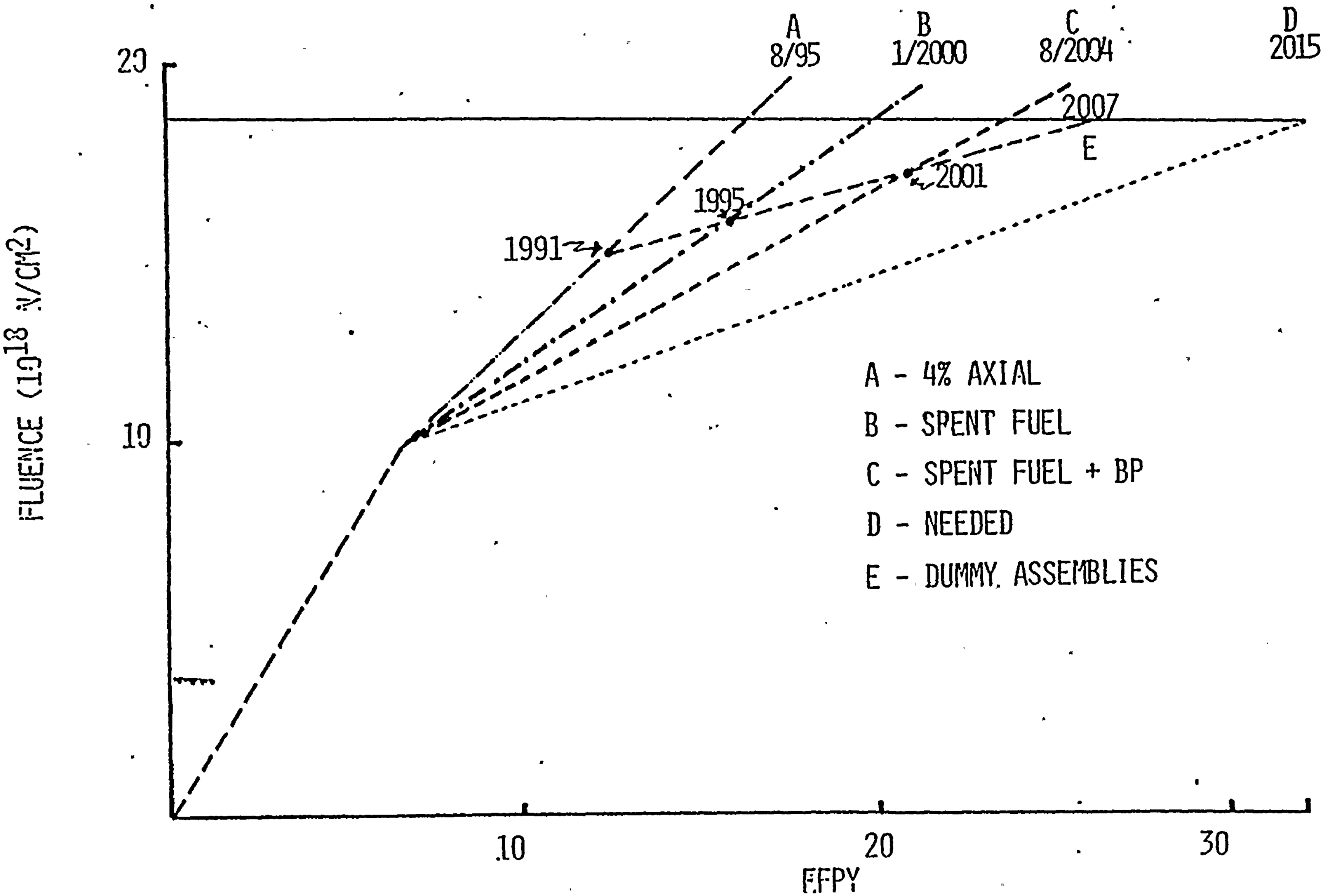


FIGURE 5.7

NEAR-TERM FLUX REDUCTION

OPTION	PERIPHERAL POWER*	REDUCTION FACTOR	CYCLE LENGTH	PEAKING FACTOR
GENERIC	1.0	.76	-	
8 CYCLE AVG.	.76	1.0	-	
NEEDED (2015)	.17	4.5	-	
(2007)	.21	3.4		
PTP 4 CYCLE 9	.45	1.7	0	
SPENT FUEL	.30	2.5	- 6 DAYS	+2%
BURNABLE POISONS	.27	2.8	- 6 DAYS	+2%
SPENT & BP's	.23	3.3	- 12 DAYS	+4%

* AT CRITICAL WELD AXIAL PLANE

TABLE 5.1

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FLUX REDUCTION OPTIONS

(CU = .32, NI = .57)

OPTION	RT_{NDT} @ 2007	RT_{NDT} @2015	DATE RT_{NDT} = 300 °F
GENERIC	376	396	11/86
8 CYCLE AVE.	356	374	11/89 - 7/90*
CYCLE 9 DESIGN	325	339	8/95
SPENT FUEL	312	322	1/2000
SPENT FUEL + BP	304	313	8/2004
STAINLESS STEEL	286	293	9/2025

*INCLUDES AXIAL CORRECTION

TABLE 5.2

6. Long Term Flux Reduction Plans

The long term flux reduction actions have several purposes. These are

- o Reduce vessel flux further than the near term actions
- o Increase the flexibility in means to accomplish flux reduction at the lowest cost
- o Quantify for NRC review all flux reductions

The long term options currently envisioned are summarized in Table 6.1. The most flexibility and lowest cost is expected to come from concentrating on axial zoning of fuel although the manufacturing problems associated with this have not yet been identified.

Quantification of flux reduction is expected to proceed in several steps using the DOT 4.3 computer code.

- o Historical cycle specific flux levels using actual radial and axial powers for both units through Cycle 8.
- o Near term cycle flux levels to establish expected date of reaching screening criteria.
- o Axial and radial adjoint calculations using various materials in the long term options to establish guidelines to be used for future reload design.

Though not yet filled in for other long term options, Table 6.2 does provide the expected peaking factor impact of the dummy assembly option. The expected increase in fuel cycle cost of dummy assemblies is very large as is the original cost of implementation. Therefore, very high motivation exists to avoid dummy assemblies in view of the high confidence that they will not be necessary.

FLUX REDUCTION OPTIONS

- PERIPHERAL POISONS
  - BURNABLE ABSORBERS
  - PART LENGTH BURNABLE ABSORBERS
  - HAFNIUM
- HIGH BURNUP ASSEMBLIES
- NATURAL OR DEPLETED URANIUM
- PARTIAL FUEL ASSEMBLIES
- NON-FUEL ASSEMBLIES
- AXIAL OR RADIALLY ZONED ASSEMBLIES

TABLE 6.1

LONG-TERM FLUX REDUCTION
(2015)

OPTION	PERIPHERAL POWER *	REDUCTION FACTOR	CYCLE LENGTH	PEAKING FACTOR
8 CYCLE AVG.	.76	1.0	-	-
NEEDED	0.16	4.8		
STAINLESS STEEL	"0.12"	6.3		+10%
NATURAL U				
DEPLETED U				
NATURAL + BP				
PARTIAL ASM.				

* AT CRITICAL WELD ELEVATION

TABLE 6.2

7. Schedule

The following time table provides the currently envisioned actions for the FPL flux reduction program for the Turkey Point nuclear units.

Date	Milestone
1978	Implement low-leakage core designs
March 1982	Set modified low-leakage designs
Fall 1982	Near term design change fuel management evaluation
Spring 1983	Finalize Unit 3 Cycle 9 and Unit 4 Cycle 10 Design changes.
	Obtain DOT 4.3 Code at FPL
	Load modified low-leakage core in Unit 4 Cycle 9 (annual).
Fall 1983	Perform long range flux reduction fuel management studies.
	Submit FPL lattice physics topical
	Establish DOT model for Turkey Point
Winter 1983-4	Evaluate fluence using DOT
	Submit PDQ model topical
	Load Unit 3 Cycle 9 with near-term flux reduction changes.
	Have fuel vendor assess fuel assembly designs needed for long-term flux reductions.
Spring 1984	Set Unit 3 Cycle 10 design
	Load Unit 4 Cycle 10
	Submit historical fluence calculations

ATTACHMENT C

ASSESSMENT OF SAFETY MARGINS

Assessment of Safety Margins

Introduction

The core configurations aimed at reducing fluence described previously involve a reduction in the power of the peripheral assemblies. This leads to an increase in peak heat flux in other regions of the core which translates into an increase in the radial nuclear peaking factor and a commensurate increase in the hot spot total peaking factor.

This discussion will focus on how the higher peaking factors can be accommodated without exceeding the core design safety limits, and without reducing reactor power from the current level of 2200 MWth.

Table 1, Assessment of Safety Margins at Turkey Point

There are four basic safety limits associated with the design and operation of a reactor core. The total peaking factor, Fq, has to be maintained below the Fq limit, which is determined by the requirement that during a LOCA, the peak clad temperature must be maintained below 2200°F.

The enthalpy rise factor, $F_{\Delta H}$, which is closely related to the radial peaking factor has to be maintained below its limit which is set so that during anticipated transient of low and moderate frequency there will be no departure from nucleate boiling (DNB) in the core and therefore no fuel damage.

For low probability accidents DNB is permitted, but the extent of fuel damage must be limited so as to assure maintenance of a coolable core geometry and radiation dose rates within limits specified in 10CFR100.

Maximum reactor coolant system pressure during transients must be limited so that the stresses in the pressure vessel and piping stay below the ASME code limits.

An assessment of the available operating and design margin for each one of these parameters shows that there is substantial margin to fuel damage at Turkey Point so as not to present a concern when the nuclear peaking factors are increased. The effect of higher peaking factors on coolant pressure is negligible so that pressure need not be considered further. The concern therefore need to be focused on the availability of $F_{\Delta H}$ and Fq margin when low fluence core configurations are implemented.

Figure 1, Design Margin and Safety Limit

Here are depicted factors which must be considered in evaluating the operating and design margins available. It is possible that the current Technical Specification limit for the peaking factors could be substantially below the safety

limit thus providing design margin which can be utilized to raise the Tech Spec limit. To accomplish this usually requires new analytical methods which reduce the magnitude of the uncertainties, either through more sophisticated calculational methods or by factoring in new data that became available since the previous safety analysis was performed.

The expected peaking factors (nuclear peaking plus calculational and measurement uncertainties) for the low fluence core configurations will increased and therefore the Tech Spec limits need to be raised.

Table 2, Projected $F_{\Delta H}$ Margin at Turkey Point

This table compares the expected enthalpy rise peaking factor, $F_{\Delta H}$ for the various low fluence core designs with the corresponding Tech Spec limit and suggests ways in which the $F_{\Delta H}$ Tech Spec limit can be increased to accommodate the increased nuclear $F_{\Delta H}$. The values shown in this and the following table are projections only, based on previous generic sensitivity studies, and must be confirmed by plant specific calculations after the design has been finalized.

The table shows the $F_{\Delta H}$ values for the present low leakage core design typified by Turkey Point 4, Cycle 9 and three stages of contemplated fluence reduction designs: near term flux reduction measures, such as those contemplated for Turkey Point 3, Cycle 9; long term flux reduction schemes, such as placing natural or depleted uranium fuel on the flats; and replacing outer assemblies with dummy stainless steel assemblies. The $F_{\Delta H}$ for the present low leakage design is quite close to the current Tech Spec limit of 1.55, which is also the generic limit for all current Westinghouse fuel. The nuclear $F_{\Delta H}$ is expected to increase by 4, 7 or 10%, respectively for the designs with lower fluence. The table indicates that for Turkey Point 3, Cycle 9 the available DNB margin identified in the Westinghouse Rod Bow Topical Report (WCAP-8691), already approved by the NRC, can be utilized. For further flux reduction the Westinghouse Improved Thermal Design Procedure (ITDP), which is based on a new DNB correlation (WRB-1) and on statistical combination of uncertainties must be implemented. This methodology has been generically approved for Westinghouse fuel, but the uncertainties and sensitivities must be qualified on a plant specific basis.

From this table it can be concluded that with the implementation of the Improved Thermal Design Procedure there will be sufficient $F_{\Delta H}$ margin to accommodate any of the contemplated low fluence core designs.

Table 3; Projected Fq Margin at Turkey Point

This table compares the expected total peaking factor, Fq, for the low fluence core designs with corresponding Tech Spec limits and proposes ways to minimize or accommodate the increase in Fq. The increase in hot channel peaking inherent in the flux reduction designs has a dual effect on Fq margin. It raises the hot spot nuclear peaking Fq and simultaneously lower the allowable Fq as calculated by the LOCA analysis. To counteract these effects new methodologies must be applied. One is BART (Best estimate Analysis Reflood Transient), submitted by Westinghouse to the NRC in 1980 (WCAP-9561) and expected to be approved by the NRC in 1983. BART utilizes more favorable heat transfer coefficients and axial profiles during the reflood phase of a LOCA calculation. Another new methodology is BASH (Best estimate Analysis System Hydraulics) representing a still more advanced reflood model. BASH is to be submitted to the NRC in 1983 but NRC review will probably not be completed till 1985-86. Each of these new LOCA models is expected to increase the allowable Fq by about 0.1. To obtain additional margin the nuclear(expected) Fq can be reduced with axially zoned burnable poison rods with the active portion of the rods near the mid plane.

The conclusion from this table is that with NRC approval and implementation of the BART methodology and axially zoned burnable poison the low fluence core designs under consideration will have the required Fq margin. To implement dummy stainless steel assemblies would require approval and implementation of the BASH methodology.

Conclusion; Assessment of Safety Margins

1. It can be concluded that sufficient design margin exists at Turkey Point to implement low fluence core loadings at the current power level of 2200 MWth without exceeding safety limits, provided NRC approval of the BART LOCA methodology (already reviewed by Sandia for the NRC) is received in time for Turkey Point 3, Cycle 9 startup in December 1983.

2. To implement long term flux improvements would require approval of the Improved Thermal Design Procedure (already generically approved). To implement a core with dummy assemblies would require additional NRC approval of the BASH LOCA methodology, which can not be expected before 1985-86.

3. Relief from the rules or criteria of regulations, such as those of Appendix K of 10CFR50 is not needed.

ATTACHMENT D

TRANSIENT ANALYSES

PLANT SPECIFIC ANALYSIS — TURKEY POINT PLANT
SCOPE AND SCHEDULE

FPL is currently considering a plant specific analysis for the Turkey Point Plant. The intent of such an analysis would be to identify the dominant sequences of events which could lead to pressurized thermal shock of the reactor vessel. The results of this analysis would be used in the evaluation of modifications to plant systems, equipment and/or procedures. In addition, the analysis would support the continued operation of the Turkey Point nuclear units past the date when they exceed the RT_{NDT} screening criteria.

The current analysis schedule conservatively assumes that Turkey Point units will exceed the screening criteria in late 1989. Based on FPL's ongoing flux reduction program, the required submittal date is not expected until the mid-1990's. As stated earlier in this report, the vessel flux evaluation to be completed by the summer of 1983 will better define the analysis schedule.

ANALYSIS DEVELOPMENT PLAN

FPL has considered a number of different approaches to the Turkey Point plant specific analysis. The most promising general approach identified to date is similar to that taken by Westinghouse in their thermal shock probabilistic risk assessment (PRA)*. Cooldown sequences are identified by constructing event trees for the major transient and LOCA classes. The event trees are further resolved and quantified by developing fault trees for the systems and THERP diagrams for operator actions. The cooldown sequences are then passed through a thermal analysis screening. Using conservative criteria, the sequences are

identified as potential crack initiators or non-initiators. The high frequency potential initiators are then subject to a detailed fracture mechanics analysis to more clearly define the thermal shock scenarios.

At present, there are no established acceptance criteria for this type of analysis. FPL recognizes this is an ongoing NRC effort and is willing to assist the staff in developing such criteria.

*Summary of Evaluations Related to Reactor Vessel Integrity, Westinghouse Electric Corporation, May 1982

DEPARTMENTAL RESPONSIBILITIES

The analysis described in the previous section requires coordinating the efforts of a number of disciplines. Responsibility for the overall effort lies with FPL's Nuclear Energy Department. The tasks of the analysis have been assigned as follows: 1) Fuel Resources Department - thermal/hydraulic analyses and fluence calculations; 2) Nuclear Energy Department - vessel material properties and; 3) Power Plant Engineering Department - probabilistic risk assessment and fracture mechanics.

ACTION TO DATE

In planning the plant specific analysis, FPL engineers have reviewed much of the available literature on the thermal shock subject. In particular, a detailed comparison of the generic plant described in the Westinghouse thermal shock PRA to the Turkey Point plant was made. A number of significant differences were identified such as RWST temperature and High Pressure Safety Injection System performance characteristics. Based on this comparison, FPL concludes that the Turkey Point units would respond more favorably to the cooldown sequences identified than the generic Westinghouse plant.

TRANSIENT ANALYSES

FEBRUARY 1983

Introduction

Florida Power and Light has been actively pursuing the resolution of pressurized thermal shock concern both on a generic and on a plant specific basis. In mid 1981 when Rancho Seco overcooling transient highlighted this concern, the issue was given top priority by the analysis subcommittees of the Westinghouse and Combustion Engineering Owners' Groups. The Westinghouse Owners' Group (WOG) evaluated bounding over-cooling transients for all of their plants and concluded that in the near term all plants would operate safely. The analyses were documented in a report WCAP-10019 and were submitted to the Nuclear Regulatory Commission in December 1981. A plant specific evaluation of Turkey Point Units 3 and 4 submitted to the NRC in January 1982 concluded safe plant operation for the end of design plant life for bounding overcooling transients. Through dialogues with the NRC staff it was recognized that the overcooling transients resulting from multiple component failures need to be evaluated to completely address the pressurized thermal shock concern. A generic study, prepared through the Westinghouse Owners' Group and submitted to the NRC in May 1982, concluded that high probability overcooling transients resulting from multiple component failures would not cause flaw initiation in any Westinghouse plant over the next three year period. In mid 1982, with the formation of an FPL Task Committee for the resolution of PTS issue, an in-house investigation of small breaks was initiated to explore the benefits of plant modifications and operating procedure changes. In the longer term, dominant overcooling transients identified by Turkey Point probabilistic risk assessment will require further evaluation.

January 1982 Submittal

The plant specific submittal included calculations for the bounding overcooling transients initiated by large and small breaks in the primary and secondary systems. Plant specific thermal/hydraulic analyses were used as input for large break fracture mechanics calculations while the generic small break thermal/-hydraulic analyses for three loop Westinghouse plants provided input for small break calculations. Stress analysis and fracture mechanics evaluations were performed based on an end of life weld fluence of 6.3 X 10^{19} nvt which corresponds to an end of life RT_{NDT} of 407^{o}F. Operator action was assumed only for the large steam line break for isolating the

supply of auxiliary feedwater to the faulted steam generator within ten minutes. In case of a small primary break, a two inch break in the hot leg resulted in loop stagnation and therefore, no credit was taken for the mixing of safety injection with the primary fluid. Based on warm prestressing it was concluded that all cracks would arrest within three quarters of the vessel wall.

Analyses in Progress

In mid 1982 when the FPL/PTS task force was formed, it was decided to investigate higher probability small breaks further to generate plant specific thermal/-hydraulic transients and to assess the effects of plant modifications and operating procedure changes.

An analysis of a two inch small break loss-of-coolant in the hot leg concurrent with loss of offsite power which trips reactor coolant pumps is in progress. Minimum decay heat, maximum safety injection flow, maximum auxiliary feed water flow, minimum safety injection temperature and minimum auxiliary feed water temperature are assumed. The break size considered produces primary loop stagnation, thus minimizing the safety injection mixing and maximizing the reactor vessel cooldown.

Another analysis currently in progress is the small steam line break from zero reactor power initiated by a stuck open steam safety valve concurrent with loss of offsite power. Initial conditions and sequence of events are chosen such as to maximize cooldown.

Sensitivity studies which would provide an assessment of ways possible for minimizing the cooldown will be performed to evaluate the effects of safety injection temperature, auxiliary feed water flow rate, steam relief valve isolation and operator action. It is desirable that for high probability overcooling transients, the downcomer fluid temperature be maintained above the end of life RT_{NDT}. With the implementation of reduced flux core designs, the end of life RT_{NDT} is estimated to lie between 300^{o}F and 330^{o}F.

The system transient analysis is performed with the RETRAN computer code developed by the Electric Power Research Institute. FPL has contracted with Energy Incorporated to conduct an independent check of the Turkey Point model. A topical report on the RETRAN code has been submitted to the NRC for review by the utility RETRAN Users' Group.

Analyses Being Considered - Near Term

FPL is considering carrying the transient analyses for small breaks further to evaluate mixing of safety injection, thermal and pressure stresses in the reactor vessel and crack growth. Since the end of life RT_{NDT} is expected to lie between 300 and 330°F, it is desirable to demonstrate that the flaws would not initiate for high probability small breaks and for others, the cracks would arrest in less than three quarters of vessel thickness without having to depend on warm prestressing.

A dialogue has been established with EPRI to acquire their computer codes for performing mixing, stress and fracture analyses. EPRI is at present performing pressurized thermal shock analyses for Robinson-2, Calvert Cliffs and TMI-1 using the COMMIX code for mixing, the ABAQUS code for stress analysis and the PTS-1 code for fracture mechanics analysis.

Long Term - PTS Analyses

Long term PTS analyses would address dominant events identified by Turkey Point probabilistic risk assessment. The overcooling events which have cooldown rates higher than 100°F/hr and which result in downcomer water temperature below the end of life RT_{NDT} would be considered potential flaw initiaters. These transients would be further investigated for crack initiation and arrest using fracture mechanics codes. Analysis results from probability events would then be evaluated to assess plant modifications and operating procedure changes to prevent crack initiation. Low probability events would be investigated for crack arrest. The long term effort would aim to demonstrate that the plants could operate safely at the end of life with an $RT_{NDT} >$ 300°F.

Conclusion

The analyses submitted to the NRC thus far have demonstrated that probable overcooling transients would not initiate flaw propagation for the next few years. The analyses have further demonstrated that flaws would be arrested for the end of plant design life. The near term and the long term analyses would provide an evaluation of beneficial plant modifications and operating procedure changes in case the end of life RT_{NDT} exceeds the screening limit of 300°F.

PRESSURIZED THERMAL SHOCK
TURKEY POINT UNITS 3 & 4
PLANT SPECIFIC ANALYSES

o DECEMBER 1981	GENERIC EVALUATION WCAP 10019
o JANUARY 1982	PLANT SPECIFIC
o MAY 1982	GENERIC PRA
o NEAR TERM	EVALUATION OF SMALL BREAKS o PLANT MODIFICATIONS o OPERATING PROCEDURES
o LONG TERM	EVALUATE DOMINANT TRANSIENTS o PRA

SUMMARY OF PLANT SPECIFIC ANALYSES SUBMITTED IN JANUARY 1982

EVENTS ANALYZED

- o LARGE LOCA
- o SMALL LOCA (GENERIC TRANSIENT)
- o LARGE SLB
- o SMALL SLB (GENERIC TRANSIENT)

ASSUMPTIONS

- o EOL $RT_{NDT} = 407^{\circ}F$
- o 3/4 T CRACK ARREST
- o WARM PRESTRESSING
- o NO MIXING, SMALL LOCA
- o 10 MINUTE OPERATOR ACTION, LARGE SLB

CONCLUSION

- o CRACK ARREST FOR EOL

ANALYSES IN PROGRESS

EVENTS BEING ANALYZED

- o SMALL LOCA (STAGNANT LOOP)
- o SMALL SLB

OBJECTIVES

- o PLANT SPECIFIC TRANSIENTS
- o PLANT MODIFICATION EVALUATION
  - o RWST TEMPERATURE
  - o AUXILIARY FEEDWATER
  - o BLOCK VALVE ON ATMOSPHERIC DUMP
- o OPERATING PROCEDURES EVALUATION

METHODS

- o RETRAN MODEL FOR TURKEY POINT

ANALYSES BEING CONSIDERED

OBJECTIVE

- SHOW IMPROVEMENT OVER JANUARY 1982 SUBMITTAL
    - PREVENT CRACK INITIATION
    - CRACK ARREST 0 - 1/2 T
    - CRACK ARREST WITHOUT WPS

EVENTS

- SMALL LOCA (STAGNANT LOOP)
- SMALL SLB

ASSUMPTION

- EOL RT_{NDT} 300 - 360^{o}F

CALCULATIONS

- MIXING OF SI
- STRESS ANALYSIS
- FRACTURE MECHANICS

METHODS

- SIMPLE MIXING MODEL/COMMIX
- ABAQUS
- PTS-1

LONG TERM - PTS ANALYSES

OBJECTIVE

- o EVALUATE DOMINANT PTS TRANSIENTS IDENTIFIED BY TURKEY POINT PRA

DESIRED GOAL

- o DEMONSTRATE SAFE PLANT OPERATION AT EOL $RT_{NDT} > 300^{o}F$

ATTACHMENT E

SURVEILLANCE PROGRAM

VESSEL INSPECTION

The ultrasonic weld examinations performed on the Turkey Point Units 3 and 4 reactor pressure vessels utilized 0°, 45° and 60° angle beam techniques. All examinations were performed in accordance with the requirements of the ASME B&PV Code Section XI, Appendix I of the 1974 edition with addenda through the summer of 1975 plus, the requirements of the USNRC regulatory guide 1.150 were closely adhered to. Contact examination techniques were conducted on the vessel interior clad surfaces.

The 0 degree straight beam examination was relied upon to detect flaws oriented essentially parallel to the surface and to monitor sound transmission efficiency.

The 45 degree angle beam technique was modified to a full vee technique in order to monitor the area directly under the cladding. Sensitivity for this examination area utilized a two inch by .140 inch notch (2% code notch).

The 60 degree angle beam technique was relied upon to complement the 45 degree beam in the detection of flaws oriented essentially perpendicular to the surface of the vessel.

In addition, during the Unit 4 examination, a dual 70 degree refracted longitudinal team technique was employed to complement the 45 degree beam in the detection and/or evaluation of flaws located at the clad interface and the area beneath the clad for a distance of one inch.

During the examination of both units, the vessel girth welds joining the upper shell-to-intermediate shell and intermediate shell-to-lower shell courses were covered 100 percent. There are no existing axial welds in either vessel.

The Unit 3 examination exhibited no recordable indications.

The Unit 4 examination exhibited indications oriented at the vessel outside surface which were attributed to probable surface anomalies. Cladding indications were detected with the 45 degree beam, but not confirmed by the 70 degree technique and thus attributed to cladding irregularity. These indications are not indicative of flaws in the base material or in the clad-base material interface.

45° NEAR SURFACE EXAM. (clad area + 1 inch)
Reference Level = notch response (.140 x 2")
Recording Level = 50% of reference (notch)

70° NEAR SURFACE EXAM (Clad interface + 1 inch)

Reference Level = 1/16" dia. SD hole DAC curve
Recording Level - 50% of reference

BALANCE OF EXAMINATION VOLUME

Reference Level = .312" dia. SD hole DAC curve
Recording Level = 20% of reference

The Turkey Point Surveillance Program has six capsules remaining only two of which contain weld material. This leaves a relatively small sample of critical material to be managed over plant life. The Unit 3 weld material is representative of both critical welds in Units 3 and 4 in that it contains the same weld wire number and flux lot as both critical welds in Units 3 and 4. The flux lot number in Unit 4 capsule is different than those found in the critical welds.

It is FP&l's plan to remove a capsule at the completion of Cycle 10 which is sometime in 1986. At the present time we are considering integrating our surveillance program so the capsule removed may be either from Unit 3 or 4 but not both.

Some other options which are being considered are:

- Changing a lagging capsule to a leading position.
- Removing a capsule and inserting it into a test reactor to end of life fluence.
- Reconstituting charpy samples to either more fully develop Energy Temperature curves at existing radiation levels or create additional capsules.
- Modifying existing WOL samples to obtain better fracture toughness information.

FP&L is continuing a search for archival materials and archival materials information.