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ST. LUCIE UNIT 1 CYCLE 6 SAFETY ANALYSIS REPORT

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ST. LUCIE UNIT 1 CYCLE 6

SAFETY ANALYSIS REPORT

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SAFETY ANALYSIS REPORT

1.0 INTRODUCTION

The results of the Safety Analysis for Cycle 6 of the St. Lucie Unit 1 nuclear plant are presented in this report. The topics addressed include operating history of the reference cycle, power distribution considerations, control rod reactivity requirements, temperature coefficient considerations, thermal-hydraulic design analysis, control rod ejection accident analysis, and setpoint analysis.

The Cycle 6 design includes 84 Exxon Nuclear Company (ENC) Reload Batch XN-1 assemblies enriched to 3.67 w/o U-235. The Cycle 6 design utilizes 656 B_4C-Al_2O_3 burnable absorber rods, each containing 23.8 mg B-10 per inch. The reload consists of 28 assemblies with no burnable absorber rods, 8 assemblies with 4 burnable absorber rods, 36 assemblies with 12 burnable absorber rods, and 12 assemblies with 16 burnable absorber rods.

2.0 SUMMARY

The St. Lucie Unit 1 nuclear plant is scheduled to operate in Cycle 6 beginning in April of 1983 with 84 fresh assemblies supplied by Exxon Nuclear Company (ENC) (Reload Batch XN-1 or H). The composition of the core during Cycle 6 will be 84 ENC assemblies in Batch H and a total of 133 Combustion Engineering (CE) assemblies; 64 in Batch G, 68 in Batch F and 1 in Batch E.

The characteristics of the fuel and the reloaded core are in conformance with existing Technical Specification limits or with proposed revised Technical Specification limits supported by ENC analyses regarding shutdown margin provisions and temperature coefficient limits. The ENC fuel design is presented in References 1 and 2. The Plant Transient Analysis is presented in Reference 3 and LOCA-ECCS Break Spectrum Analysis is provided in Reference 4. The results of the Control Rod Ejection Analysis are provided herein and are derived in part from the generic analysis described in Reference 5.

The neutronic characteristics of Cycle 6 are similar to those of Cycle 5. The range of kinetics coefficients reflected in the Plant Transient Analysis bounds those expected in Cycle 6. The excess shutdown margin at EOC HZP is calculated to be 1500 pcm. A postulated control rod ejection event is conservatively calculated to result in an energy deposition of less than 155 cal/gm.

The peaking factors F_r and F_{xy} at BOC HFP equilibrium xenon conditions are 1.51$\pm$.08 and 1.55$\pm$.08, respectively and occur in an ENC assembly. The peaking factors F_r and F_{xy} at BOC for Combustion Engineering assemblies are 1.39$\pm$.07 and 1.51$\pm$.08, respectively. The maximum values for the peaking factors F_r and F_{xy} occur at EOC in an ENC assembly and are 1.53$\pm$.08 and 1.56$\pm$.08, respectively. The St. Lucie Unit 1 Cycle 6 Technical Specification limit on F_r and F_{xy} is 1.70.

3.0 OPERATING HISTORY OF THE REFERENCE CYCLE

St. Lucie Unit 1 Cycle 5 has been chosen as the reference cycle with respect to Cycle 6 due to the close resemblance of the neutronic characteristics between these two cycles. The Cycle 5 operations began in December 1981 with shutdown scheduled for February 1983. The core has accrued about 10,000 MWD/MT as of the end of November 1982. The Cycle 5 core consisted of 217 Combustion Engineering assemblies.

The measured power peaking factors at hot-full-power equilibrium xenon conditions have remained below Technical Specification limits throughout Cycle 5. The linear heat rate including uncertainties of 7% for measurement, 3% for engineering, 1% for densification and 2% for power, has remained below 13.7 kw/ft for Cycle 5. The peaking factors F_r and F_{xy} have both remained below 1.60 for Cycle 5. Cycle 5 has operated essentially free of control rods and Cycle 6 is anticipated to operate in a similar manner.

A Cycle 5 assembly power distribution as measured by the Combustion Engineering code CECOR is shown in Figure 3.1 at 5,000 MWD/MTU. Calculations with the XTGPWR[9] code are also shown in Figure 3.1 for comparison with the measured data. The measured critical boron concentration versus exposure for Cycle 5 is shown in Figure 3.2. The critical boron curve predicted by ENC is plotted in Figure 3.2 for comparison with the measured data.

K H

	L	J	G	F	E	D	C	B	A
11	1 E2 .870 .884 1.61	2 F2 1.035 1.036 .10	3 F1 1.272 1.220 -4.09	4 E1 1.015 1.021 .59	5 F1 1.212 1.178 -2.81	6 E1 .982 1.001 1.93	7 F1 1.219 1.165 -4.43	8 F2 .997 .978 -1.91	
10									9 G1 .857 .843 -1.63
9	10 F2 1.035 1.036 .10	11 G4 1.266 1.221 -3.55	12 F2 1.075 1.081 .56	13 G3 1.262 1.229 -2.61	14 E1 .984 1.014 3.05	15 F1 1.250 1.230 -1.60	16 E1 1.004 1.010 .60	17 G3 1.099 1.069 -2.73	
8									18 D2 .390 .392 .51
7	19 F1 1.272 1.220 -4.09	20 F2 1.077 1.082 .46	21 F1 1.301 1.261 -3.07	22 D1 .966 .994 2.90	23 F1 1.253 1.230 -1.84	24 F2 1.005 1.032 2.69	25 F2 .957 .981 2.51	26 G1 1.095 1.111 1.46	
6	27 E1 1.015 1.021 .59	28 G3 1.265 1.228 -2.92	29 D1 .965 .998 3.42	30 F1 1.185 1.174 -.93	31 E1 .956 1.002 4.81	32 G3 1.231 1.204 -2.19	33 F2 .950 .981 3.26	34 G1 .884 .913 3.28	
5	35 F1 1.212 1.178 -2.81	36 E1 .991 1.015 2.42	37 F1 1.257 1.235 -1.75	38 E1 .969 1.005 3.72	39 F1 1.261 1.227 -2.70	40 F2 .979 .999 2.04	41 G1 1.080 1.073 -.65	42 E2 .373 .396 6.17	
4	43 E1 .982 1.001 1.93	44 F1 1.265 1.236 -2.29	45 F2 1.013 1.044 3.06	46 G3 1.240 1.207 -2.66	47 F2 .985 .998 1.32	48 G2 1.068 1.046 -2.06	49 E2 .397 .441 11.08		
3	50 F1 1.219 1.165 -4.43	51 E1 1.012 1.012 -----	52 F2 .965 .985 2.07	53 F2 .966 .983 1.76	54 G1 1.087 1.074 -1.20	55 E2 .411 .441 7.30	CECOR XTG $(\frac{X}{C}-1)\times100$		
2	56 F2 .997 .978 -1.91	57 G3 1.108 1.070 -3.43	58 G1 1.096 1.114 1.64	59 G1 .884 .915 3.51	60 E2 .356 .397 11.52				
1	61 G1 .860 .843 -1.98	62 E2 .385 .397 3.12							

Figure 3.1 St. Lucie Unit 1, Cycle 5 Power Distribution Comparison, 5,000 MWD/MT (3,670 EFPH), HFP, ARO

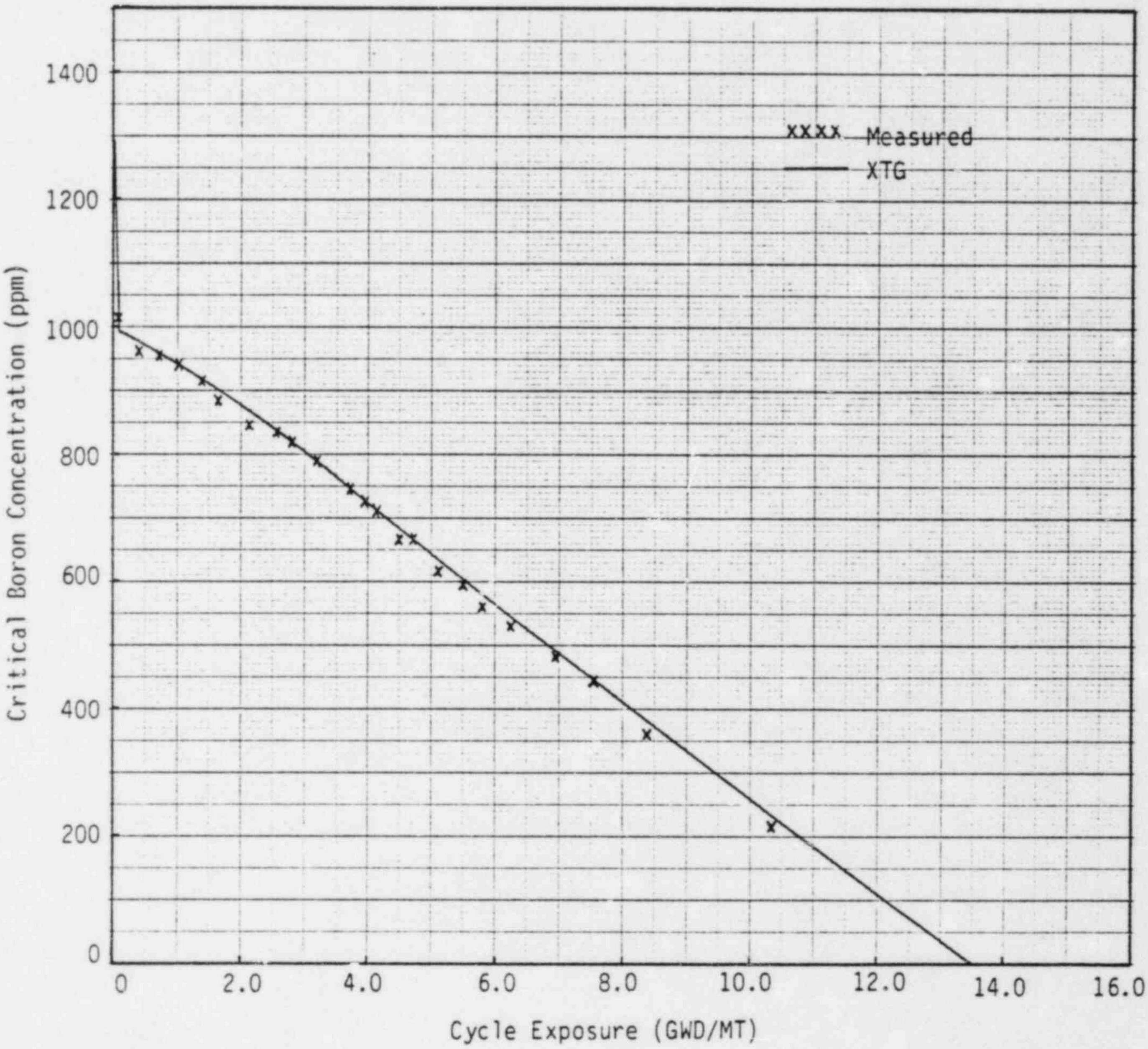

Figure 3.2 St. Lucie Unit 1, Cycle 5 Critical Boron Concentration vs. Exposure, ARO, HFP, 3-D XTG and Measured

4.0 GENERAL DESCRIPTION

The St. Lucie Unit 1 reactor consists of 217 assemblies, each having a 14x14 fuel rod array. The assemblies are composed of up to 176 fuel rods, a number of $B_4C-Al_2O_3$ burnable absorber rods in place of fuel rods (0 to 16 rods), and 5 control rod guide tubes or 4 control rod guide tubes and one instrument tube. The fuel rods consist of slightly enriched UO_2 pellets inserted into zircaloy tubes. The control rod guide tubes and instrument tubes are also made of zircaloy. Each ENC assembly contains nine zircaloy spacers with Inconel springs.

The projected Cycle 6 loading pattern is shown in Figure 4.1 with assemblies identified by their assembly number and Cycle 5 core location. The initial enrichment, number of $B_4C-Al_2O_3$ burnable absorber pins along with other descriptive parameters are shown in Table 4.1 for the fuel assemblies in Cycle 6. The calculated BOC 6 exposures, based on an EOC 5 exposure of 13,215 MWD/MTU, are shown in a quarter core representation in Figure 4.2 along with the fuel type identification. The core consists of 84 fresh ENC assemblies at an average enrichment of 3.67 w/o U-235 and 133 exposed Combustion Engineering assemblies. A low radial leakage fuel management plan has been developed and results in scatter-loading of the fresh fuel throughout the core with the fresh assemblies loaded in the core interior containing $B_4C-Al_2O_3$ burnable absorber rods. The

exposed fuel is also scatter-loaded in the center in a manner to control the power peaking. The $B_4C-Al_2O_3$ burnable absorber rods contain 23.8 mg B-10 per inch. A total of 656 burnable absorber rods are utilized; 8 assemblies with 4 burnable absorber rods, 36 assemblies with 12 burnable absorber rods and 12 assemblies with 16 burnable absorber rods.

Table 4.1 St. Lucie Unit 1 Principal Characteristics for Nuclear Analysis of Cycle 6 Fuel

	E2	F1	F2	G1	G2	G3	G4	H1	H2	H3	H4
Enrichment (w/o)	2.75	3.65	3.03	3.65	3.65	3.20	3.03	3.67	3.67	3.67	3.67
No. B_4C Pins	0	0	12	0	4	8	8	0	4	12	16
Nominal Density	94.75%	94.75%	94.75%	94.75%	94.75%	94.75%	94.75%	94%	94%	94%	94%
Pellet OD (in.)	.377	.377	.377	.377	.377	.377	.377	.370	.370	.370	.370
Clad OD (in.)	.440	.440	.440	.440	.440	.440	.440	.440	.440	.440	.440
Diametral Gap (in.)	.0075	.0075	.0075	.0075	.0075	.0075	.0075	.008	.008	.008	.008
Clad Thickness (in.)	.028	.028	.028	.028	.028	.028	.028	.031	.031	.031	.031
Rod Pitch, (in.)	.580	.580	.580	.580	.580	.580	.580	.580	.580	.580	.580
Spacer Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Bi-Metal.	Bi-Metal.	Bi-Metal.	Bi-Metal.
Fuel Supplier	CE	CE	CE	CE	CE	CE	CE	ENC	ENC	ENC	ENC
Fuel Stack Height Nominal (in.)	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7
No. of Assemblies	1	40	28	32	4	24	4	28	8	36	12
Regionwise Loading (MTU)	.389	15.56	10.13	12.44	1.52	8.92	1.49	10.67	2.98	12.78	4.16
Exposure (MWD/MT)											
BOC6	25992	26526	28321	12799	13820	15897	16752	0	0	0	0
EOC6	37539	41834	35561	29888	32206	31449	31445	14379	15796	20033	19651
Incremental Exposure	11547	15308	7240	17089	18386	15552	14693	14379	15796	20033	19651

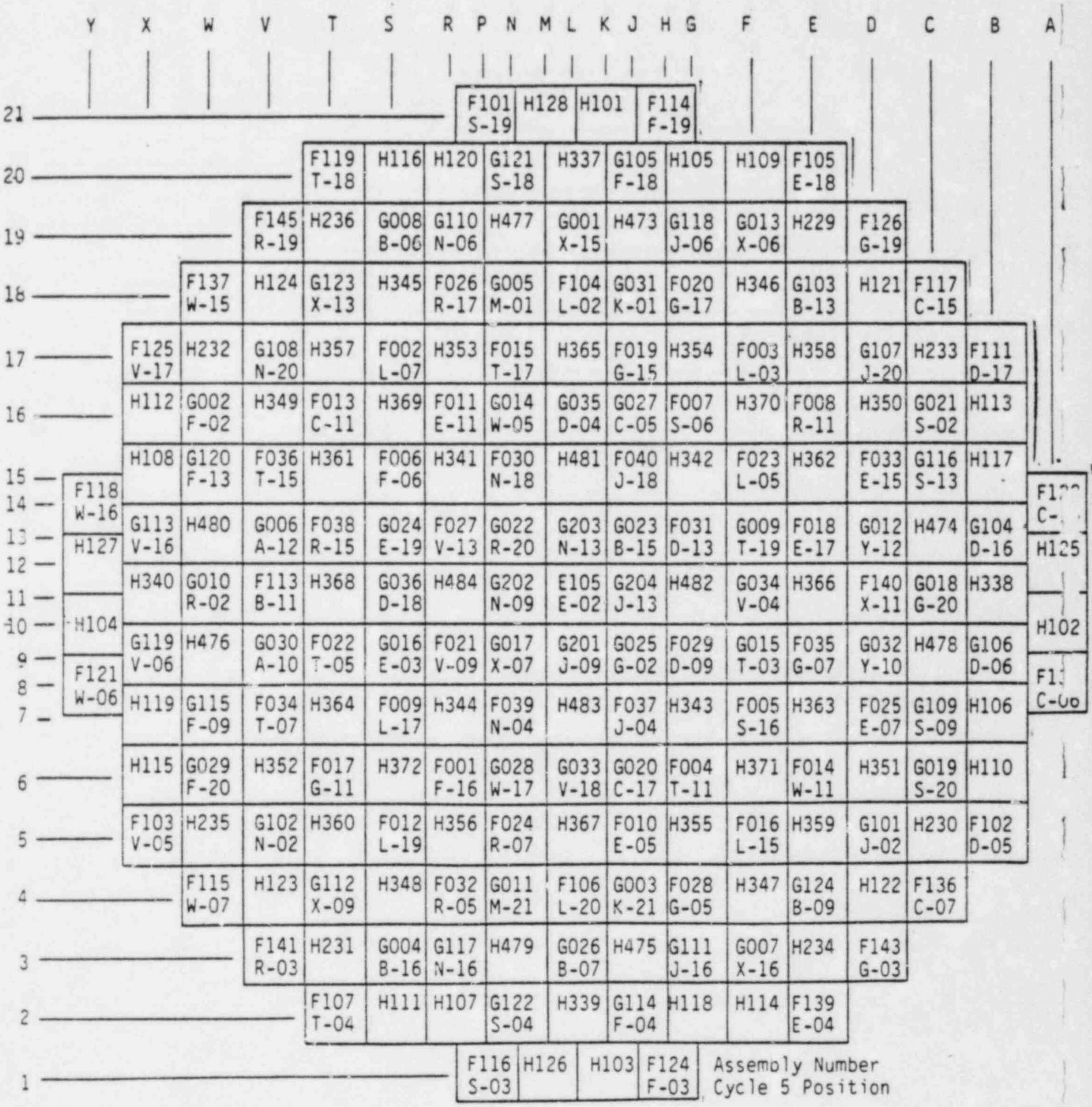

Figure 4.1 St. Lucie Unit 1, Cycle 6
Core Loading Pattern

	L	K	J	H	G	F	E	D	C	B	A
11	26.0		16.8		0.0	13.8	0.0	25.8	14.3	0.0	
10											0.0
9	16.8		14.3		25.2	13.8	25.9	11.4	0.0	16.3	
8											26.7
7	0.0		25.3		0.0	27.9	0.0	25.7	16.8	0.0	
6	13.8		13.8		27.9	0.0	27.6	0.0	11.7	0.0	
5	0.0		25.3		0.0	28.6	0.0	14.5	0.0	29.8	
4	25.8		11.4		25.8	0.0	14.5	0.0	29.7		
3	14.3		0.0		16.8	11.7	0.0	29.8			
2	0.0		16.4		0.0	0.0	29.8				
1		0.0		26.7							

BOC6 Assembly Exposure
GWD/MTU

ENC Fuel Types	Enrichment	Number of B_4C Rods
H1	3.67 w/o	0
H2	3.67 w/o	4
H3	3.67 w/o	12
H4	3.67 w/o	16

Figure 4.2 St. Lucie Unit 1 Cycle 6,
Loading Pattern with 656 B_4C Rods

5.0 FUEL SYSTEM DESIGN

A description of the Exxon Nuclear supplied fuel design and design methods is contained in References 1 and 2. This fuel has been specifically designed to be compatible with the resident fuel supplied by Combustion Engineering.

6.0 NUCLEAR CORE DESIGN

The neutronic characteristics of the projected Cycle 6 core are quite similar to those of the Cycle 5 core (see Section 6.1).

The nuclear design bases for the Cycle 6 core are as follows:

1. The design shall permit operation within the Technical Specifications for St. Lucie Unit 1 nuclear plant.
2. The length of Cycle 6 shall be determined on the basis of an assumed Cycle 5 energy of 13,215 MWD/MT exposure.
3. The Cycle 6 loading pattern shall be designed to achieve power distributions and control rod reactivity worths according to the following constraints:
   a) The peak LHR shall not exceed 15 kw/ft and the peaking factors F_r and F_{xy} shall not exceed 1.70 in any single fuel rod through the cycle under nominal full power operating conditions.
   b) The scram worth of all rods minus the most reactive rod shall exceed BOC and EOC shutdown requirements.

The neutronic design methods utilized to ensure the above requirements are consistent with those described in References 6, 7 and 8.

The Cycle 6 loading contains 656 fresh Al_2O_3-B_4C burnable absorber rods distributed among 56 of the 84 fresh ENC supplied assemblies. The reload consists of assemblies at 3.67 w/o U-235, 28 assemblies with no

burnable absorber rods, 8 assemblies with 4 burnable absorber rods, 36 assemblies with 12 burnable absorber rods and 12 assemblies with 16 burnable absorber rods. The Al_2O_3-B_4C burnable absorber rods each contain 23.8 mg/in of B-10. The core loading pattern has been designed to achieve a desirable power distribution while maximizing the benefit of assemblies with burnable absorbers to reduce the beginning of cycle (BOC) boron concentration.

6.1 PHYSICS CHARACTERISTICS

The neutronics characteristics of the Cycle 6 core are compared with those of Cycle 5 and are presented in Table 6.1. The data presented in the table indicates the neutronic similarity between Cycles 5 and 6. The reactivity coefficients of the Cycle 6 core are bounded by the coefficients used in the safety analysis. The safety analysis for Cycle 6 is applicable for Cycle 5 burnups of +500 MWD/MT and -500 MWD/MT about the nominal burnup of 13,215 MWD/MT.

The boron letdown curve for Cycle 6 is shown in Figure 6.1. The BOC6 HZP xenon free critical boron concentration is calculated to be 1,432 ppm. At 100 MWD/MT, equilibrium xenon, HFP, the critical boron concentration is 1,013 ppm. The Cycle 6 length is projected to be 15,492 MWD/MT ±300 MWD/MT at a core power of 2,700 MWt with 23 ppm soluble boron remaining.

6.1.1 Power Distribution Considerations

Representative calculated power maps for Cycle 6 are shown in Figures 6.2 and 6.3 for BOC (100 MWD/MTU, equilibrium xenon), and EOC conditions, respectively. The power distributions were obtained from a three-dimensional XTGPWR(9) model with moderator density and Doppler feedback effects incorporated.

As shown in Figure 6.2 for the design Cycle 6 loading pattern, the calculated BOC hot full power (HFP) equilibrium nuclear peaking factors F_r and F_{xy} are 1.51 and 1.55, respectively. The peak linear heat rate (LHR) at BOC HFP is calculated to be 11.1 kw/ft. At EOC conditions the corresponding values for F_r, F_{xy} and LHR are 1.53, 1.56 and 10.5 kw/ft, respectively. The Technical Specification limit on F_r and F_{xy} is 1.70.

The BOC HFP equilibrium xenon LHR including uncertainties of 7%(19) for measurement, 3% for engineering, 1% for densification and 2% for thermal power is 12.6 kw/ft and is comparable to the measured value for Cycle 5 BOC HFP equilibrium xenon of 13.7 kw/ft. At EOC HFP equilibrium xenon conditions the maximum LHR is calculated to be 11.9 kw/ft including uncertainties. The Technical Specification limit on LHR is 15 kw/ft.

6.1.2 Control Rod Reactivity Requirements

Shutdown margin evaluations for Cycle 5 and 6 are compared in Table 6.2. The Cycle 5 calculations are taken from the Cycle 5 Safety Analysis Report by Combustion Engineering and the Cycle

6 calculations were performed by ENC. The comparisons assume a Technical Specification for the required shutdown margin of 5000 pcm. The Cycle 5 excess shutdown margin is 200 pcm while for Cycle 6 it is 100 pcm. The ENC plant transient analysis[3] supports a technical specification change in the required shutdown margin from 5000 pcm to 3600 pcm. The excess shutdown margin at EOC6 HZP with a required shutdown margin of 3600 pcm is 1500 pcm.

The control rod groups and insertion limits for Cycle 6 will remain unchanged from Cycle 5. With these limits the nominal worth of the control bank, Bank 7 inserted to the insertion limits at HFP is 120 pcm at BOC and 190 pcm at EOC.

6.1.3 Moderator Temperature Coefficient Considerations

The current Technical Specifications require that the moderator temperature coefficient be less than +5 pcm/°F at or below 70% of rated thermal power, less than 2 pcm/°F above 70% power and greater than -22 pcm/°F at 100% of rated thermal power. The ENC plant transient analysis accounts for revised Technical Specifications on the moderator temperature coefficients of less than or equal to +7 pcm/°F at HZP, less than or equal to +2 pcm/°F at HFP and greater than or equal to -28 pcm/°F at HFP. The BOC HZP, ARO moderator temperature

coefficient is calculated to be 4.8 pcm/^{o}F and will meet the revised Technical Specification limit at HZP conditions. The moderator temperature coefficient at 100% rated power, BOC, equilibrium xenon conditions is calculated to be -1.3 pcm/^{o}F. The moderator temperature coefficient at EOC, HFP, equilibrium xenon conditions is calculated to be -21 pcm/^{o}F.

6.2 ANALYTICAL METHODOLOGY

The methods used in the Cycle 6 core analysis are described in References 6, 7 and 8. In summary, the reference neutronic design analysis of the reload core was performed using the XTGPWR[9] reactor simulator code. The input isotopics data were based on quarter core depletion calculations performed for Cycle 6 using the XTGPWR code. The fuel shuffling between cycles was accounted for in the calculations.

Calculated values of LHR, F_r, and F_{xy} were determined with the XTGPWR reactor model. The calculational thermal-hydraulic feedback and axial exposure distribution effects on power shapes, rod worths, and cycle lifetime are explicitly included in the analysis.

Table 6.1 St. Lucie Unit 1 Neutronics Characteristics of Cycle 5 Compared with Cycle 6 Data

	Cycle 5 BOC	Cycle 5 EOC	Cycle 6 BOC	Cycle 6 EOC
Critical Boron (ppm)				
HFP, ARO, Equilibrium xenon	980(a)(d)	20(c)	1013(d)	23
HZP, ARO, No xenon	1396(a)		1432	--
Moderator Temperature Coefficient pcm/^{o}F				
HFP	-.4(a)	--	-1.3(d)	-20.7
HZP	5.8(a)	--	4.8	-14.4
Doppler Coefficient (pcm/^{o}F)	-1.3(b)	-1.5(b)	-1.2	-1.4
Boron Worth (pcm/ppm)				
HFP	--	--	-8.5(d)	-10.2
HZP	-9.9(a)	--	-8.8	-10.7
LHR (including uncertainties)	13.7(a)	--	12.6	11.9
Delayed Neutron Fraction	.0061(b)	.0052(b)	.0058	.0050
Control Rod Worth, ARI-1, HZP (pcm)	--	7300(b)	6490	7120
Excess Shutdown Margin, HZP (pcm)	--	200(b)(e)	--	100(e)

(a) Measured data
(b) Cycle 5 Safety Analysis Report
(c) ENC calculated
(d) 100 MWD/MTU
(e) A required shutdown margin of 5000 pcm is utilized for the comparison.

Table 6.2 Comparison of Shutdown Margin
Cycles 5 and 6

	EOC5* HZP	EOC6 HZP
Control Rod Worth (pcm)		
ARI	10300	9070
N-1	7300	7120
Power Dependent Insertion Limit (PDIL)	1500	1450
[(N-1) - PDIL] * .9	5200	5100
Shutdown Margin (pcm)		
Required Shutdown Margin	5000**	5000**
Excess Shutdown Margin	200	100

* From the Cycle 5 Safety Analysis Report.

** A revision to the Technical Specifications for St. Lucie Unit 1 Cycle 6 will be made to alter the required shutdown margin to 3600 pcm.

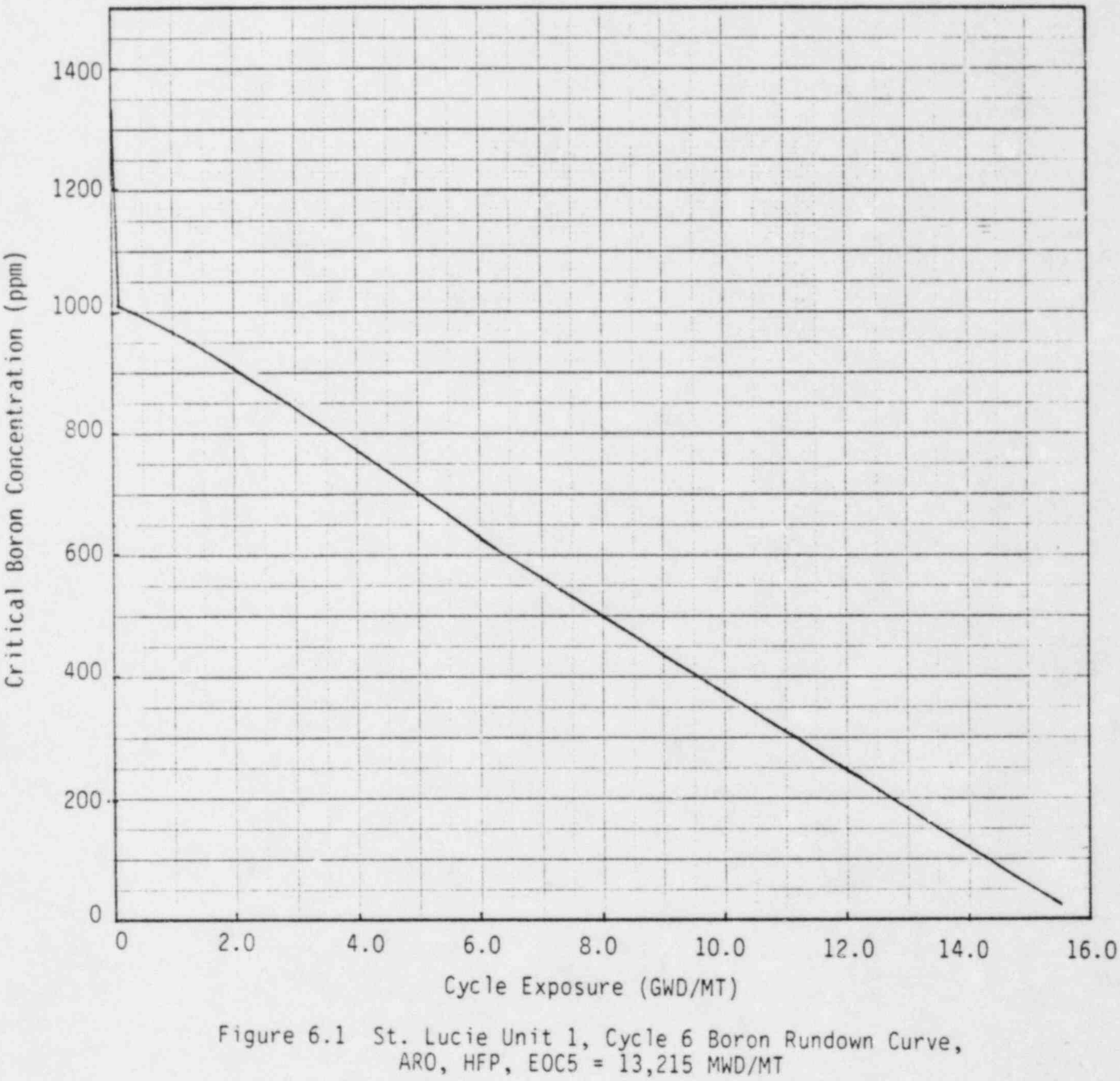

Figure 6.1 St. Lucie Unit 1, Cycle 6 Boron Rundown Curve, ARO, HFP, EOC5 = 13,215 MWD/MT

	L	K	J	H	G	F	E	D	C	B	A
11	.73		.91		1.12	1.24	1.26	.93	1.15	1.08	
10											.80
9	.91		1.07		1.02	1.21	1.06	1.22	1.12	.89	
8											.34
7	1.12		1.02		1.20	1.00	1.22	1.04	1.05	1.16	
6	1.24		1.21		1.00	1.17	1.00	1.24	1.19	.97	
5	1.26		1.07		1.21	.98	1.20	1.04	1.06	.35	
4	.93		1.23		1.04	1.23	1.04	1.10	.38		
3	1.15		1.13		1.05	1.18	1.06	.38			
2	1.08		.89		1.16	.97	.35				
1		.80		.34							

Peak Assembly = 1.26 (11E)

Peak F_R = 1.51 (11E)

Peak F_{xy} = 1.55 (11E)

Peak LHR = 11.1 kw/ft (11E)

Figure 6.2 St. Lucie Unit 1 Cycle 6, Relative Power Distribution, 100 MWD/MTU, 1,013 ppm, 2700 MWt, ARO

	L	K	J	H	G	F	E	D	C	B	A
11	.81		.96		1.23	1.16	1.25	.93	1.16	1.19	
10											.83
9	.96		1.06		1.02	1.12	1.03	1.15	1.24	.95	
8											.40
7	1.23		1.02		1.26	1.03	1.27	1.03	1.01	1.01	
6	1.16		1.12		1.03	1.29	1.06	1.27	1.06	.84	
5	1.25		1.03		1.27	1.05	1.28	1.04	.99	.37	
4	.93		1.15		1.03	1.26	1.04	1.02	.42		
3	1.16		1.24		1.01	1.06	.99	.42			
2	1.19		.95		1.01	.84	.37				
1		.83		.40							

Peak Assembly = 1.29 (6F)

Peak F_R = 1.53 (6F)

Peak F_{xy} = 1.55 (6F)

Peak LHR = 10.5 kw/ft (6F)

Figure 6.3 St. Lucie Unit 1 Cycle 6, Relative Power Distribution, 15,492 MWD/MTU, 23 ppm, 2700 MWt, ARO

7.0 THERMAL-HYDRAULIC DESIGN ANALYSIS

This section identifies thermal-hydraulic design criteria for ENC reload fuel at St. Lucie Unit 1. The hydraulic compatibility of ENC fuel with existing CE fuel at St. Lucie Unit 1 is quantified in terms of hydraulic loss coefficients. The relative DNB performance of the two fuel types is compared. Expected peak fuel rod power levels relative to the power levels required for centerline melt are also examined.

7.1 DESIGN BASES AND CRITERIA

The primary thermal-hydraulic design basis for Exxon Nuclear Company reload fuel is that fuel rod integrity should be maintained during normal operation and anticipated operational occurrences. Specific criteria are:

(1) Avoidance of boiling transition for the limiting fuel rod in the core with at least a 95% probability at a 95% confidence level.

(2) Fuel centerline temperatures should be below the melting point of the fuel pellets.

Observance of these criteria during anticipated operational transients is considered conservative relative to the requirement that anticipated operational transients not produce fuel rod failures or loss of functional capability.

The margin to boiling transition for ENC and CE fuel is assessed with ENC's XNB critical heat flux correlation[10].

7.2 SUMMARY OF THERMAL-HYDRAULIC DESIGN ANALYSIS RESULTS

The overall hydraulic loss coefficient of ENC reload fuel is found to be less than 4.0% greater than the loss coefficient of existing CE fuel. Thus, insertion of ENC fuel into the St. Lucie Unit 1 reactor will not significantly impact primary coolant flow.

Evaluation of thermal margin for both fuel types indicates MDNBRs are about 6.0% less for ENC fuel during Cycle 6* than for CE fuel evaluated for Cycle 5** operation under conditions representative of a severe operational transient event. The bulk of this MDNBR reduction for ENC fuel is associated with its higher spacer loss coefficient compared to CE fuel. An evaluation of future cycles in which a larger fraction of the core will be ENC fuel indicates slight increases in assembly flow for both ENC and CE fuel, and thereby increases limiting assembly MDNBR for both fuel types. Thus MDNBR results for the present Cycle 6 analysis will be bounding of MDNBRs in future cycles.

7.3 HYDRAULIC CHARACTERIZATION

Table 7.1 shows component loss coefficients for both fuel types. These loss coefficients are based on pressure drop tests performed in ENC's Portable Loop Hydraulic Test Facility. The single phase loss coefficients in Table 7.1 are referenced to the respective bare rod regions in each of the two fuel types. The upper and lower tie plate loss coefficients include reversible losses due to area change and losses due to simulated upper and lower core support plates.

Figure 7.1 provides a diagrammatic comparison of ENC and CE fuel pressure loss coefficients for normal full-power operation. The overall assembly loss coefficient of ENC fuel differs by less than 4.0% from that of CE fuel at St. Lucie Unit 1. Since the core loss is about one third of the

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* Cycle 6 represents the first ENC reload core consisting of 84 ENC Assemblies and 133 CE assemblies.

** Cycle 5 is a reference core consisting solely of CE fuel

total loop loss, the impact of an all-ENC core on primary loop flow is less than 0.5%.

7.4 CORE FLOW DISTRIBUTION ANALYSIS

The core flow distribution analysis is performed to assess crossflow between assemblies in the core for use in subsequent MDNBR subchannel analyses. The core flow distribution analysis is particularly important for mixed fuel loadings where hydraulically different fuel types are coresident in the core.

In the analysis each fuel assembly in an octant of the St. Lucie Unit 1 core is modeled as a hydraulic channel. The calculations are performed with the XCOBRA-IIIC computer code[16]. Crossflow between adjacent assemblies in the open lattice core is directly modeled. The single phase loss coefficients given in Table 7.1 are used to hydraulically characterize the ENC and CE fuel types. To establish limiting assembly mass, energy and momentum crossflows for subsequent MDNBR subchannel analyses, two separate core flow distribution calculations are made. In the first, an ENC fuel assembly is assumed to be limiting during Cycle 6 with a significantly higher power level relative to the remainder of the core. In the second calculation, a CE fuel assembly is assumed to be limiting during Cycle 5 operation with a high assembly power level. For the present analysis of relative DNB performance, a maximum F_r limit of 1.70 is assumed and imposed on the limiting assembly power for both the ENC and CE limiting cases. The two coreflow calculations thus establish two limiting assembly axially varying crossflow boundary conditions to be used in respective MDNBR evaluations for ENC and CE fuel types.

Table 7.2 summarizes thermal hydraulic parameters used in the core flow distribution calculations and subsequent MDNBR analyses. The calculations are performed at a 130% overpower condition (1.30 x 2700 MWt) in order to maximize potential differences in limiting assembly MDNBR between ENC and CE fuel, and to provide MDNBRs which are representative of those that may be expected during a severe operational transient event (i.e. MDNBRs near the XNB 95:95 correlation limit). Figures 7.2 and 7.3 show the core loading for the ENC fuel limiting and CE fuel limiting cases, respectively. In each case, 5% inlet flow maldistribution is assumed for the limiting and surrounding assemblies.

7.5 MDNBR SUBCHANNEL ANALYSIS

The MDNBR subchannel analysis uses the XCOBRA-IIIC computer code to evaluate thermal-hydraulic conditions in each of the subchannels of the limiting assembly. Crossflow between rod subchannels is determined in the calculations. Figures 7.4 and 7.5 show the 1/8 assembly subchannel model used for the ENC and CE MDNBR analyses, respectively. The local fuel rod power distributions shown in Figures 7.4 and 7.5 are conservatively flat and are representative of the power distributions for each fuel type at their respective average discharge exposures.

The MDNBR subchannel analysis includes factors to account for manufacturing uncertainties and [illegible]ication effects. Specifically a 3% engineering factor is applied to the limiting rod power. This factor is to account for fabrication tolerances on pellet diameter, density and enrichment. Also included is the tolerance for cladding diameter. These manufacturing uncertainties potentially impact heat flux at the limiting DNB location in the assembly. The 3% engineering factor is applied to both ENC and CE fuel designs.

Fuel densification can result in a shortening of the pellet column length and thus potentially causes an increase in the average linear heat generation rate. In-reactor shortening of the active fuel column length is conservatively evaluated from the following relationship for anisotropic densification:

$$\frac{\Delta L}{L} = \frac{\Delta \rho}{2}$$

where ΔL = decrease in fuel column length

L = fuel column length

$\Delta \rho$ = fractional density change

With an upper bound on density change of 3.5%, the increase in heat flux would be 1.75%. The fuel column length decrease due to densification is compensated by an increase in column length due to thermal expansion. This length increase is approximately 1% of the active column length. Thus, the net effect due to axial densification and thermal expansion results in a .75% increase in heat flux. In the present analysis, a conservative 1.0% heat flux penalty has been applied to allow for densification.

Interassembly crossflow between the limiting and adjacent fuel assemblies was accounted for in thermal margin calculations by imposing an axially varying crossflow boundary condition on the subchannel model used to determine MDNBR. Mass, energy and momentum crossflows were established from core flow distribution analyses for both ENC and CE limiting cases. Reference 17 describes in detail the application of ENC's thermal margin methodology for pressurized water reactors.

The XNB correlation with correction factors for non-uniform heat flux profile is used to determine the margin to boiling transition for both

ENC and CE fuel assemblies.

Conditions for the present thermal margin analysis are summarized in Table 7.2. The important factor is that the calculations have been performed for 130% overpower.

The relative impact of ENC fuel assemblies on existing core thermal margins is about an 6.0% decrease in MDNBR relative to results for a limiting CE assembly in Cycle 5. In the present analysis, the limiting ENC assembly is assumed to consist of 164 active fuel rods, while the CE limiting assembly is composed of 168 active rods. Due to a slight difference in the number of active fuel rods in Cycle 6 and Cycle 5, ENC rods have about a 1.0% higher heat flux which yields a proportionate decrease in MDNBR. The remaining 6% difference in MDNBR between ENC and CE fuel is due to hydraulic dissimilarities between the two fuel designs.

7.6 ROD BOW

In accordance with ENC rod bow methodology[11,18], the magnitude of rod bow for St. Lucie Unit 1 fuel was estimated. The calculations indicate that 50% closure of rod-to-rod gap occurs at an assembly exposure of about 85,000 MWD/MTM for ENC's 14x14 design. Significant impact to MDNBR due to rod bow does not occur until gap closures exceed 50%. Since the maximum design exposure for ENC fuel in the St. Lucie Unit 1 core is significantly less, rod bow does not limit MDNBR for ENC fuel. A further consequence of the small amount of rod bow for ENC fuel is that total power peaking is not significantly impacted.

7.7 FUEL CENTERLINE TEMPERATURE

The power level required to produce centerline melt in zircaloy clad urania fuel rods is about 21 kW/ft. Loss-of-coolant accident considerations for St. Lucie Unit 1 limit the steady state peak LHR to 15 kW/ft. The 40% margin between 21 kW/ft and 15 kW/ft is large enough that fuel centerline melt is not a limiting factor for anticipated operational transients.

Table 7.1 Assembly Component Loss Coefficients

	ENC	CE
Lower Tie Plate*	5.197 $Re^{-0.0356}$	3.55
Spacer	1.752 $Re^{-0.067}$	1.019 $Re^{-0.04}$
Upper Tie Plate*	7.42	7.45
Bare Rod Friction	0.1987 $Re^{-0.20}$	0.1987 $Re^{-0.27}$

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* Includes reversible losses due to change in area and losses due to core support hardware.

Table 7.2 Thermal-Hydraulic Design Data

Operating Conditions	
Rated Power (Core	2700 MWt
Fraction of Heat Generated in Fuel	.975
Pressurizer Pressure	2250 psia
Core Inlet Temperature	549.0^{o}F
Total Reactor Coolant Flow	139.4 x 10^6 lb/hr
Active Coolant Flow	134.8 x 10^6 lb/hr
Limiting Assembly Peaking Factors	
Axial	1.60
Engineering	1.03
F_r	1.70
Total Nuclear	2.80

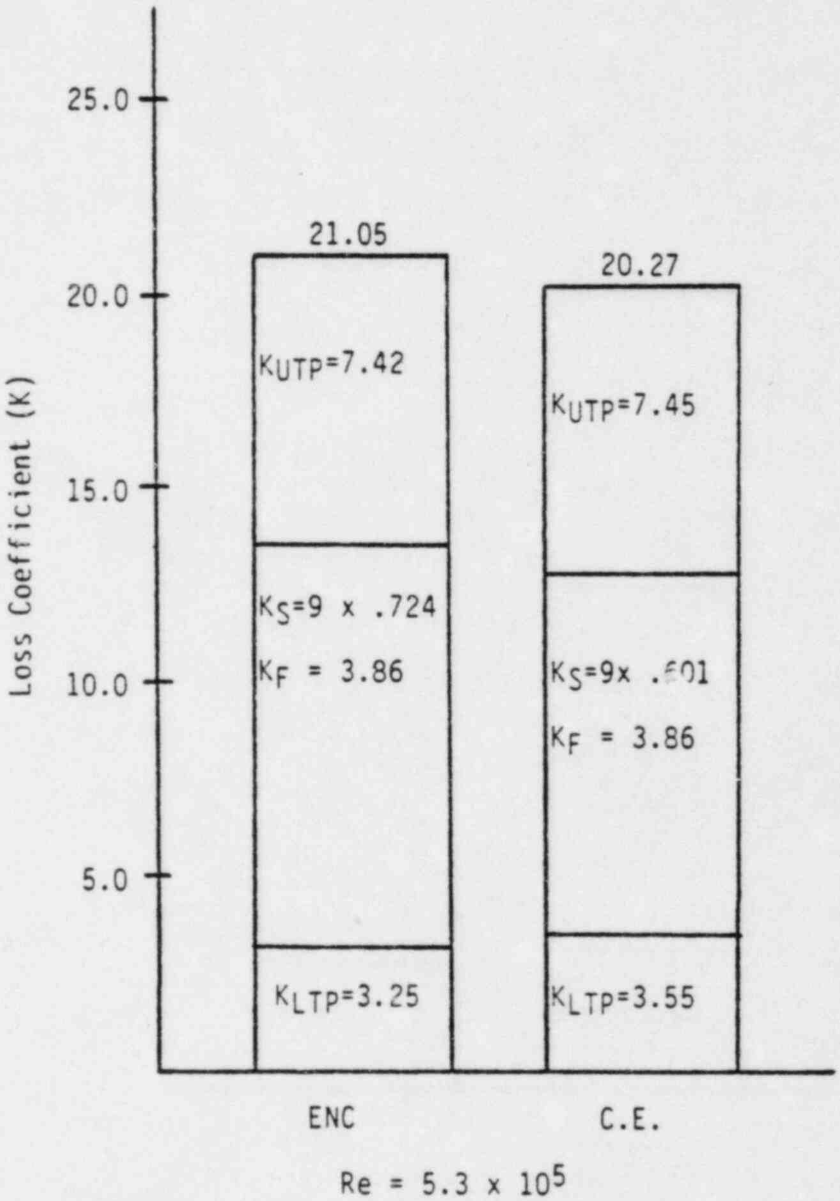
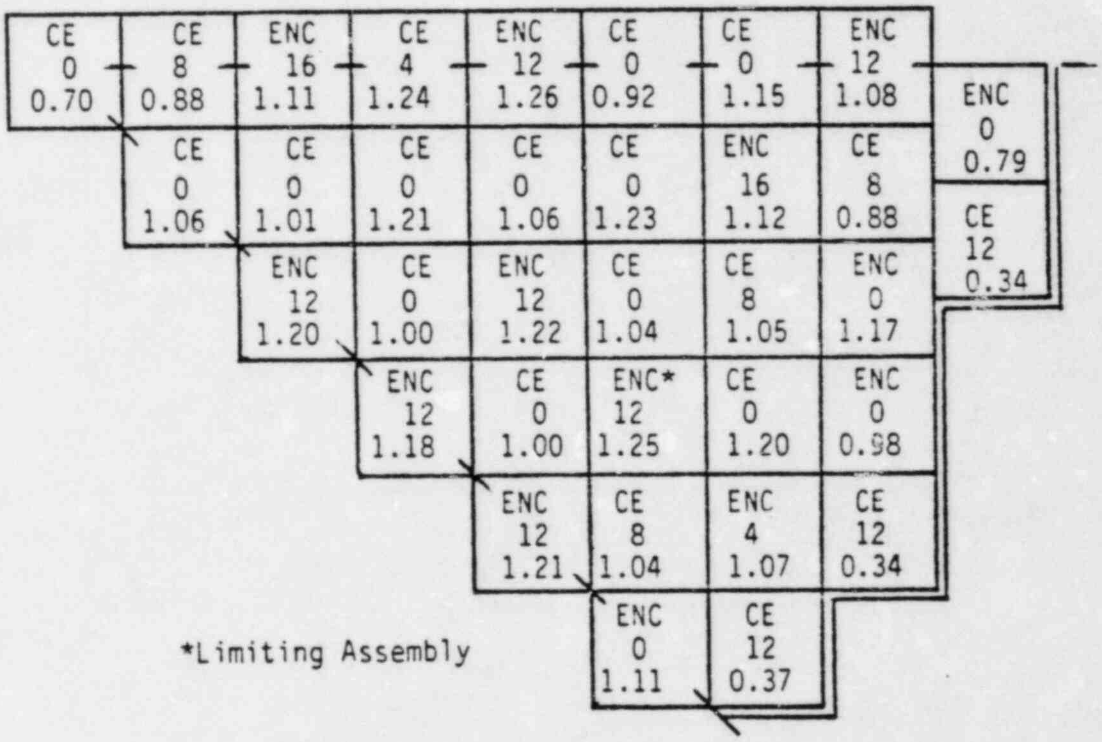


Figure 7.1 Comparison of ENC and C.E. Component Loss Coefficients



xx	Fuel Type
x	No. of B_4C Pins
xxx	Radial Assembly Power Factor

Figure 7.2 Cycle 6 1/8 Core Model

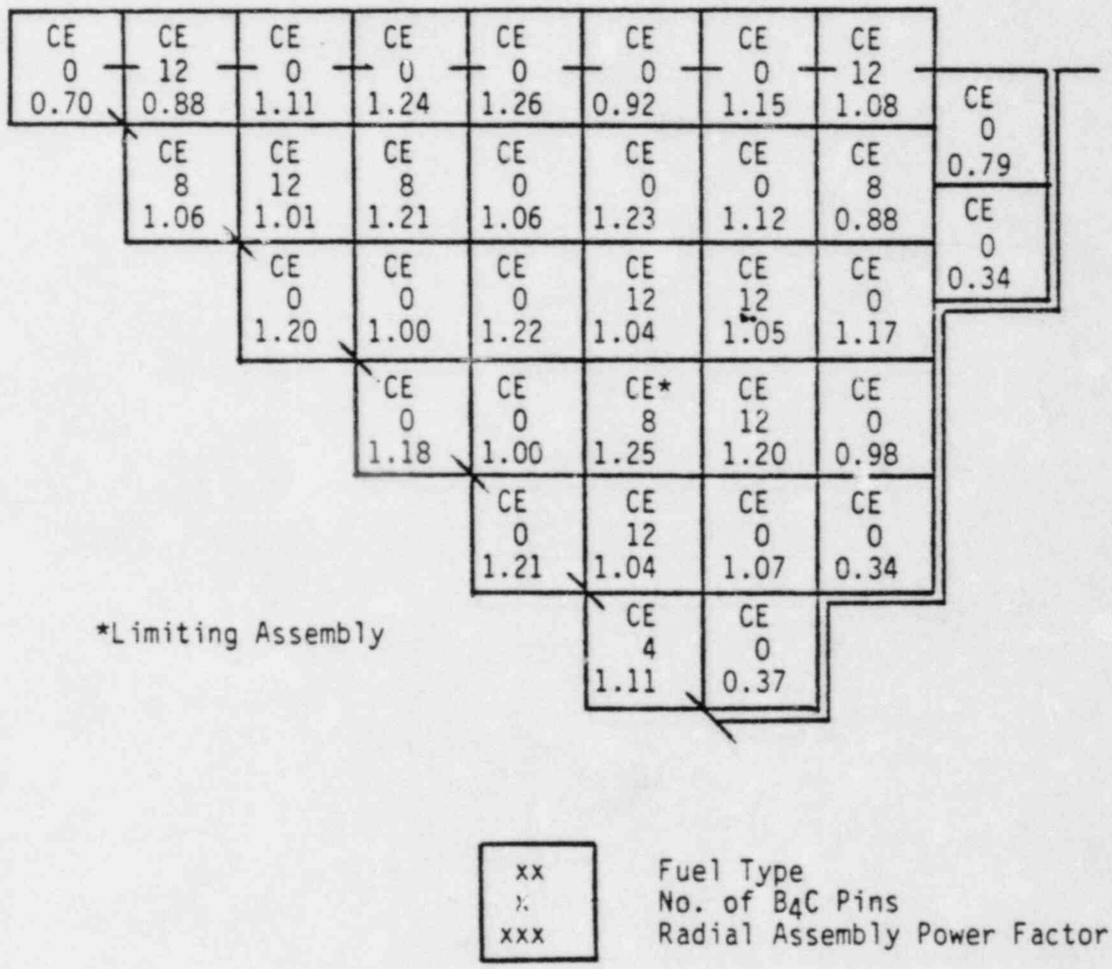


Figure 7.3 Cycle 5 1/8 Core Model

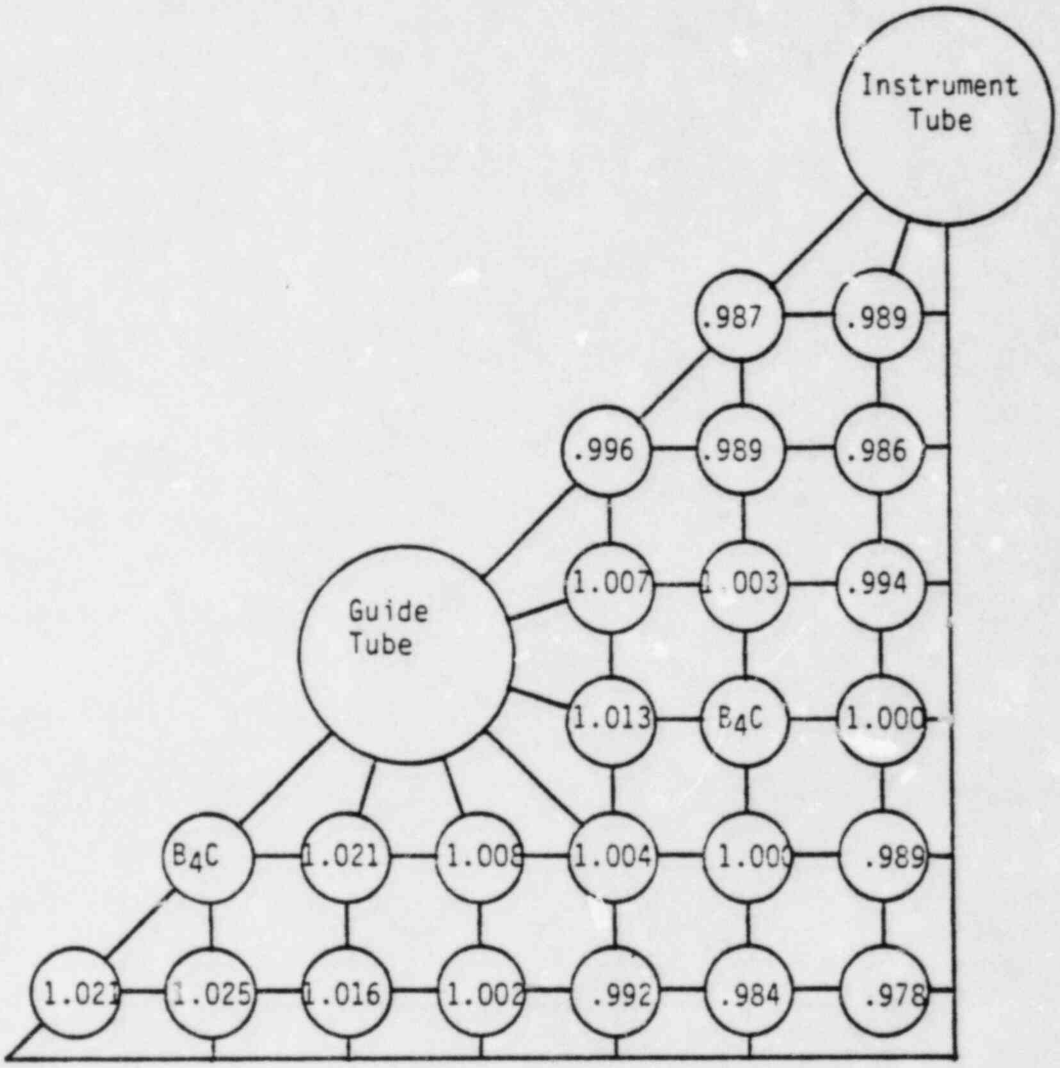


Figure 7.4 ENC 1/8 Assembly Subchannel Model

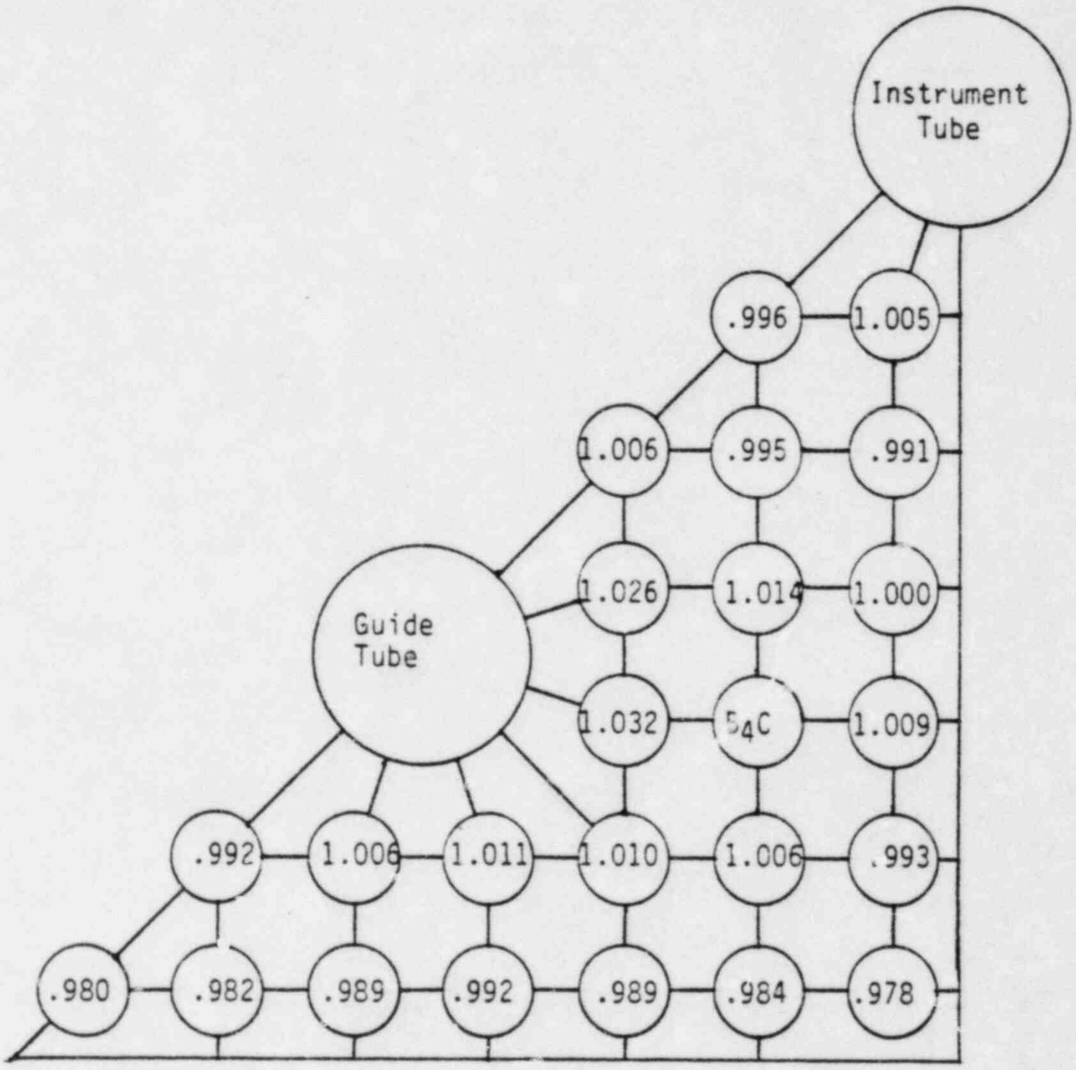


Figure 7.5 CE 1/8 Assembly Subchannel Model

8.0 ACCIDENT AND TRANSIENT ANALYSES

8.1 PLANT TRANSIENT ANALYSIS

Plant transient analyses for the ENC fuel that is being placed in St. Lucie Unit 1 this cycle are reported in Reference 3. Thermal margins from this analysis are applicable to mixed core loadings of ENC and the resident Combustion Engineering fuel.

8.2 ECCS ANALYSIS

The LOCA-ECCS analysis for ENC fuel at St. Lucie Unit 1 is reported in Reference 4. This analysis is applicable to ENC fuel for mixed core loadings of ENC and the resident Combustion Engineering fuel at St. Lucie Unit 1. The analysis remains valid so long as any NSSS modifications and changes in system operational parameters continue to be bounded by the analysis.

8.3 ROD EJECTION ANALYSIS

A Control Rod Ejection Accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a Control Element Assembly (CEA) and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The rod ejection accident has been evaluated with the procedures developed in the ENC Generic Rod Ejection Analysis[5]. The ejected rod worths and hot pellet peaking factors were calculated using the XTGPWR[9]

code. No credit was taken from the power flattening effects of Doppler or moderator feedback in the calculation of ejected rod worths or resultant peaking factors. The calculations made for Cycle 6 using a full core XTGPWR model were two-dimensional. The total peaking factor, F_Q^T, were determined as the product of radial peaking (as calculated using XTGPWR) and a conservative axial peaking factor. The pellet energy deposition resulting from an ejected rod was conservatively evaluated explicitly for BOC and EOC conditions. The HFP pellet energy deposited was calculated to be 151.3 cal/gm at BOC and 144.1 cal/gm at EOC. the HZP pellet energy deposition was calculated to be less than 25 cal/gm for both BOC and EOC conditions. The rod ejection accident was found to result in an energy deposition of less than the 280 cal/gm limit as stated in Regulatory Guide 1.77. The significant parameters for the analyses, along with the results, are summarized in Tables 8.1 and 8.2.

A conservative core pressure surge associated with the CEA ejection is calculated in the ENC Generic Rod Ejection Analysis[5] to be 150 psia. Since the nominal core pressure for St. Lucie Unit 1 is 2250 psia, this indicates a maximum pressure of 2400 psia. The primary coolant system pressure will not exceed the limit of 2750 psia.

Table 8.1 St. Lucie Unit 1, Cycle 6, Ejected Rod Analysis, HFP

		BOC		EOC	
		Value	Contribution[a] to Energy Deposition, (cal/gm)	Value	Contribution[a] to Energy Deposition, (cal/gm)
A.	Initial Fuel Enthalpy (cal/gm)	76.9		75.5	
B.	Generic Initial Fuel Enthalpy (cal/gm)	40.8		40.8	
C.	Delta Initial Fuel Enthalpy (cal/gm)	36.1	36.1	34.7	34.7
D.	Maximum Control Rod Worth (pcm)	100	120	116	121
E.	Doppler Coefficient (pcm/^{o}F)	-1.2	.95[b]	-1.4	.89[b]
F.	Delayed Neutron Fraction, β	.0058	1.02[b]	.0050	.04[b]
G.	Power Peaking Factor[c]	3.0	--	3.2	--
H.	Power Peaking Factor Used [d]	4.0	--	4.0	--
	Total Fuel Enthalpy cal/gm		151.3[c]		144.1[c]

(a) The contribution to the total pellet energy deposition is a function of initial fuel enthalpy, maximum control rod worth, Doppler coefficient, and delayed neutron fraction. The energy deposition contribution values and factors are derived from data calculated in the "Generic Analysis of the Control Rod Ejection Transient...." document.

(b) These values are multiplication factors applied to (C+D).

(c) A conservative axial peaking factor of 1.50 is used in conjunction with a two dimensional calculation of the radial peaking factor for the Ejected Rod. The calculations are performed with the feedbacks turned off.

(d) The energy deposition due to maximum control rod worth is a function of the power peaking factor.

(e) Total pellet energy deposition (cal/gm) calculated by the equation -
Total (cal/gm) = (C+D) (E) (F)

Table 8.2 St. Lucie Unit 1, Cycle 6, Ejected Rod Analysis, HZP

	BOC		EOC	
	Value	Contribution[a] to Energy Deposition, (cal/gm)	Value	Contribution[a] to Energy Deposition, (cal/gm)
A. Initial Fuel Enthalpy (cal/gm)	16.7		16.7	
B. Generic Initial Fuel Enthalpy (cal/gm)	16.7		16.7	
C. Delta Initial Fuel Enthalpy (cal/gm)	0	0	0	0
D. Maximum Control Rod Worth (pcm)	214	20	292	20
E. Doppler Coefficient (pcm/$^{\circ}$F)	-1.0[e]	1.03[b]	-1.0[e]	1.03[b]
F. Delayed Neutron Fraction, β	.0058	1.05[b]	.0050	1.20[b]
G. Power Peaking Factor[c]	7.1	--	8.8	--
H. Power Peaking Factor Used[d]	9.0	--	9.0	--
Total Fuel Enthalpy cal/gm		21.6[f]		24.2[f]

(a) The contribution to the total pellet energy deposition is a function of initial fuel enthalpy, maximum control rod worth, Doppler coefficient, and delayed neutron fraction. The energy deposition contribution values and factors are derived from data calculated in the "Generic Analysis of the Control Rod Ejection Transient...." document.

(b) These values are multiplication factors applied to (C+D).

(c) A conservative axial peaking factor of 2.0 is used in conjunction with a two dimensional calculation of the radial peaking factor for the Ejected Rod. The calculations are performed with the feedbacks turned off.

(d) The energy deposition due to maximum control rod worth is a function of the power peaking factor.

(e) For this Doppler coefficient a conservative value of -1.0 was assumed at BOC and EOC.

(f) Total pellet energy deposition (cal/gm) calculated by the equation -

$$\text{Total (cal/gm)} = (C+D)\,(E)\,(F)$$

9.0 SUMMARY OF OPERATING LIMITS

Operating limits for the St. Lucie Unit 1 nuclear plant are summarized below. Exxon Nuclear Company methods of analysis for determining or verifying the operating limits are detailed in Subsection 9.5 and References 12, 13, and 14. Results of the ENC statistical analyses indicate that operating limits previously established for St. Lucie Unit 1 are adequate for Cycle 6 operation with ENC reload fuel present in the core.

9.1 REACTOR PROTECTION SYSTEM

The reactor protection system (RPS) is designed to assure that the reactor is operated in a safe and conservative manner. The input parameters for the RPS are denoted as limiting safety system settings (LSSS). The values or functional representation of the LSSSs are calculated to ensure adherence to the specified acceptable fuel design limits (SAFDLs) during steady state and anticipated operational occurrences (AOOs). The safe operation of the reactor is also maintained by restricting reactor operation to be in conformance with the limiting conditions for operation (LCOs) which are administratively applied at the reactor plant. The LSSS and LCO parametric values are presented in the following sections.

9.2 SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS

The specified acceptable fuel design limits (SAFDLs) are experimentally or analytically based limits on the fuel and cladding which preclude fuel damage. These limits may be exceeded neither during

steady-state operation nor during AOOs. The SAFDLs are used to establish the reactor setpoints to ensure safe operation of the reactor. The specific SAFDLs used to establish the setpoints are:

- o The local power density (LPD) which coincides with fuel centerline melt.
- o The MDNBR corresponding to the accepted criterion which protects against the occurrence of DNB.

The LPD limit for St. Lucie Unit 1 has been 21 kw/ft in prior cycles and this limit is being retained for Cycle 6. It is noted that ENC reload fuel for Cycle 6 does not contain gadolinia-bearing fuel rods and, therefore, will not exhibit any significant change in either fuel operating temperature or fuel melt temperature relative to prior CE fuel.

The ENC critical heat flux correlation, XNB, was used in the ENC thermal margin analysis with statistical parameters corresponding to an upper 95/95 value of 1.22 which is conservative relative to the 95/95 limit for XNB. Observance of the limiting conditions for operation will protect against DNB with 95% probability at a 95% confidence level during an AOO.

9.3 LIMITING SAFETY SYSTEM SETTINGS

9.3.1 Local Power Distribution Control

The local power distribution (LPD) trip limit is the locus of the limiting values of core power level versus axial shape

index that will produce a reactor trip to prevent exceeding the 21 kw/ft LPD limit. The correlation between allowed core power level and peripheral axial shape index, ASI, was determined using ENC methods which take into account the total calculated nuclear peaking and the measurement and calculational uncertainties associated with power peaking. The LPD barn for operation at 2700 MW_{th} is shown in Figure 9.1 as a locus of power and ASI pairs which is conservatively bounded by the ENC calculated power and ASI pairs. In this figure ASI is defined as the difference between the bottom core power reading and the top core reading divided by the sum of the top and bottom readings.

9.3.2 Thermal Margin/Low Pressure

The thermal margin/low pressure (TM/LP) trip protects against the occurrence of DNB during steady state operations and for many, but not all, AOOs. This reactor trip system monitors primary system pressure, core inlet temperature, core power and ASI and a reactor trip occurs when primary system pressure falls below the computed limiting core pressure, P_{var}. As with the LPD trip ENC has used its statistical setpoint methodology to verify the adequacy of the existing TM/LP trip for Cycle 6. The ENC methodology for the TM/LP trip accounts for uncertainties in core operating conditions, XNB (the ENC critical heat flux correlation) uncertainties and uncertainties in power peaking. The TM/LP existing trip function for operation at 2700 MWt, which was verified by ENC, is given by:

$$P_{var} = 2061*A1(ASI)*QR1(Q)+15.85*T_{in}-8950,$$

where Q is the higher of the thermal power and the nuclear flux power, T_{in} is the inlet temperature in ^{O}F and A1 and QR1 are shown in Figures 9.2 and 9.3, respectively.

9.3.3 Additional Trip Functions

In addition to the LPD and TM/LP trip functions, other reactor system trips have been determined to provide adherence to reactor system design criteria. The setpoints for these trips, shown in Table 9.1, are unchanged from the Cycle 5 values.

9.4 LIMITING CONDITIONS FOR OPERATION

9.4.1 DNB Monitoring

The validity of the existing LCO for allowable core power as a function of ASI was verified to ensure adherence to thermal-hydraulic fuel design limits during a postulated CEA drop and loss-of-flow operational occurrences. The ENC statistical analysis accounted for the effects of uncertainties associated in core operating parameters, the XNB critical heat flux correlation, and power peaking. The allowed core power as a function of ASI for the existing LCO is shown to be conservatively bounded by the present analysis in Figure 9.4.

9.4.2 Linear Heat Rate Monitoring

In the event that the in-core detector system is not in operation for an extended period of time, the peak linear heat rate will be monitored through the use of a linear heat rate LCO. The

verification of this LCO was performed in a fashion similar to that used in verifying the LPD limiting safety system setting (Section 9.3.1) and the verification is shown in Figure 9.5. The Linear Heat Rate LCO limits core power so that an LPD of 15 kw/ft is not exceeded. This LPD value is based on loss-of-coolant (LOCA) considerations.

9.5 SETPOINT ANALYSIS

9.5.1 Limiting Safety System Setting: Local Power Distribution

The local power distribution (LPD) trip monitors core power and ASI in order to initiate a reactor scram which precludes exceeding fuel centerline melt conditions. In the analysis for this trip function 1374 axial power distribution cases were examined to establish bounding values of total power peaking, F_Q, versus ASI. These cases were generated in a manner consistent with that discussed in Reference 12. ENC statistical methods were then employed to account for the uncertainties in the parameters that are given in Table 9.2.

The peak linear heat rate in the core occurs at the position of the maximum total peaking factor, F_Q. The maximum total peaking factor, F_Q, is referred to as the "hot spot" in the core. It is the ratio of the maximum linear heat generation rate in the core to the average linear heat generation rate in the core. F_Q is, therefore, the product of F_{xy} (the ratio of the power of the peak fuel pin to the

average fuel pin in the plane) times the core average axial power peaking factor, F_z, at the same axial position. The total peaking factor with uncertainties is, therefore:

$$F_Q = F_{xy} \times F_z.$$

Bounding values are utilized for F_{xy}, i.e., 1.70 for unrodded planes and 1.87 for rodded planes.

The actual allowed power for each ASI was calculated statistically incorporating the uncertainties listed in Table 9.2 as described in References 13 and 14 to produce the series of points in Figure 9.1. The results in Figure 9.1 which bound the existing St. Lucie Unit 1 LPD trip and thus, verify the adequacy of the existing trip function.

Thermal Margin/Low Pressure LSSS

The TM/LP trip is designed to shut the reactor down should the reactor conditions (ASI, inlet temperature, core power and pressure) approach the point where DNB might occur during either normal operation or an AOO. The present analysis uses ENC's XNB critical heat flux correlation and ENC's statistical setpoint methodology and is consistent with the NRC's Standard Review Plan in requiring DNB be avoided with 95% probability at a 95% confidence level.

The uncertainties shown in Tables 9.2 and 9.3 were included in the verification of the TM/LP trip as described in References 13 and 14. The minimum excess margin of protection provided by the trip was 38 psi.

9.5.2 Limiting Conditions for Operation

DNB Monitoring

The TM/LP trip system does not monitor reactor coolant flow and does not consider changes in power peaking which do not significantly change ASI. Thus, the TM/LP trip generally does not provide DNB protection for the four pump coastdown and CEA drop AOOs. The ENC analysis of these transients is given in References 14 and 15. The LCO presented here administratively protects the DNB SAFDL for these transients.

The method used by ENC to establish the DNB LCO involved simulations of the CEA drop and the loss-of-flow transients using the core thermal hydraulic code, XCOBRA-IIIC[16], to determine the initial power, as a function of ASI, which provides protection from DNB with 95% probability. The uncertainties listed in Tables 9.4 and 9.5 were applied using the methodology described in Reference 13. The results of the statistical analyses are summarized by the points in Figure 9.4. The points bound and, thus, verify the adequacy of the existing DNB LCO for St. Lucie Unit 1 which is shown in the same figure by the straight line segments.

LPD Monitoring

The plant technical specifications allow plant operation for limited periods of time if the in-core detectors are out of service. In this situation, the LPD barn provides protection in

steady state operation against penetration of the 15 kw/ft LPD limit established by LOCA considerations. ENC statistical methodology for the LPD LCO is essentially the same as that for the LPD LSSS except:

(1) The peak LPD limit is 15 kw/ft, and

(2) The uncertainties listed in Table 9.4 were used, as opposed to the values in Table 9.2.

The allowed power versus ASI was statistically analyzed to account for the appropriate uncertainties. The points in Figure 9.5 represent the statistical calculation of the LHR curve as described in Reference 13. The existing LCO curve is shown by the straight line segments in Figure 9.5 and are conservative relative to the ENC calculated points.

Table 9.1 Additional LSSS Trip Functions

Parameter	Set Point	Uncertainty
Low steam generator pressure	600 psia	22 psi
Low steam generator water level	37%	10 in
Variable high power	9.61% of rated (<10% of rated)	2%
Low reactor coolant flow	95%	2%
High pressurizer pressure	2400 psia	22 psi
Asymmetric steam generator pressure	135 psia	22 psi
High containment pressure	3.3 psig	--

Table 9.2 Uncertainties Applied for LPD Trip Calculation

Source	Value*
Engineering tolerance	$\pm$ 0.03
Peaking uncertainty	$\pm$ 0.085
Trip processing & decalibration	- 0.01 - 0.05 of rated
ASI uncertainty	$\pm$ 0.06

* The distributions are treated as normal and the uncertainty range represents $\pm$ 2σ values.

Table 9.3 Uncertainties Applied to Only the DNBR Calculation

Source	Value*
Pressure Measurement	± 126.66 psi
Trip bias	-30 psi
Inlet coolant temperature	± 2°F
XNB correlation(10)	1.261 0.843
Flow measurement	± 0.036 of rated

* The distributions are treated as normal and the uncertainty range represents ± 2σ values.

Table 9.4 Uncertainties Applied for the LCO Based on LPD

Source	Value*
Engineering tolerance	$\pm$ .0.03
Peaking uncertainty	$\pm$ 0.085
Power measurement	$\pm$ 0.02 of rated
ASI uncertainty	$\pm$ 0.05

* The distributions are treated as normal and the uncertainty range represents $\pm$ 2σ.

Table 9.5 Uncertainties Applied to LCO Based on DNB

Source	Value*	
	CEA Drop	Loss of Flow
Pressure measurement	± 30 psi	± 22 psi
Inlet coolant temperature	± 28°F	± 2°F
XNB correlation(10)	1.261 0.841	1.261 0.841
Flow measurement	± 0.036 of rated	± 0.036 of rated
Scram delay	--	± 0.15 seconds
Trip setpoint	--	± 0.02 of rated flow
Scram speed	--	± 8.35 inches/second
Rod worth	--	± 0.5%
Flow coastdown	--	± 0.098† seconds

* The distributions are treated as normal and the uncertainty range represents ± 2σ.

† Time for flow to drop from 90% to 80% of rated flow.

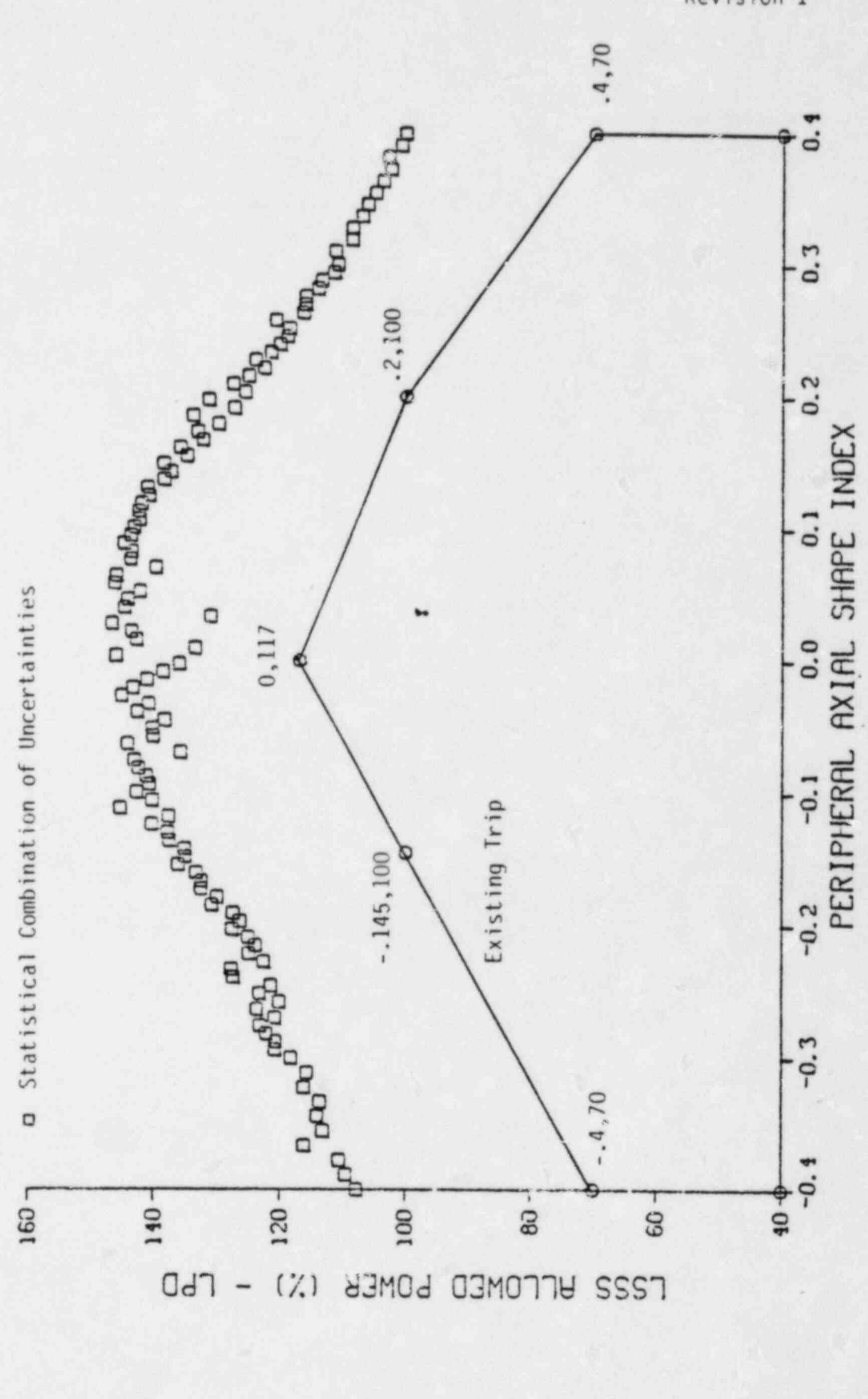


Figure 9.1 St. Lucie Unit 1 LPD Barn

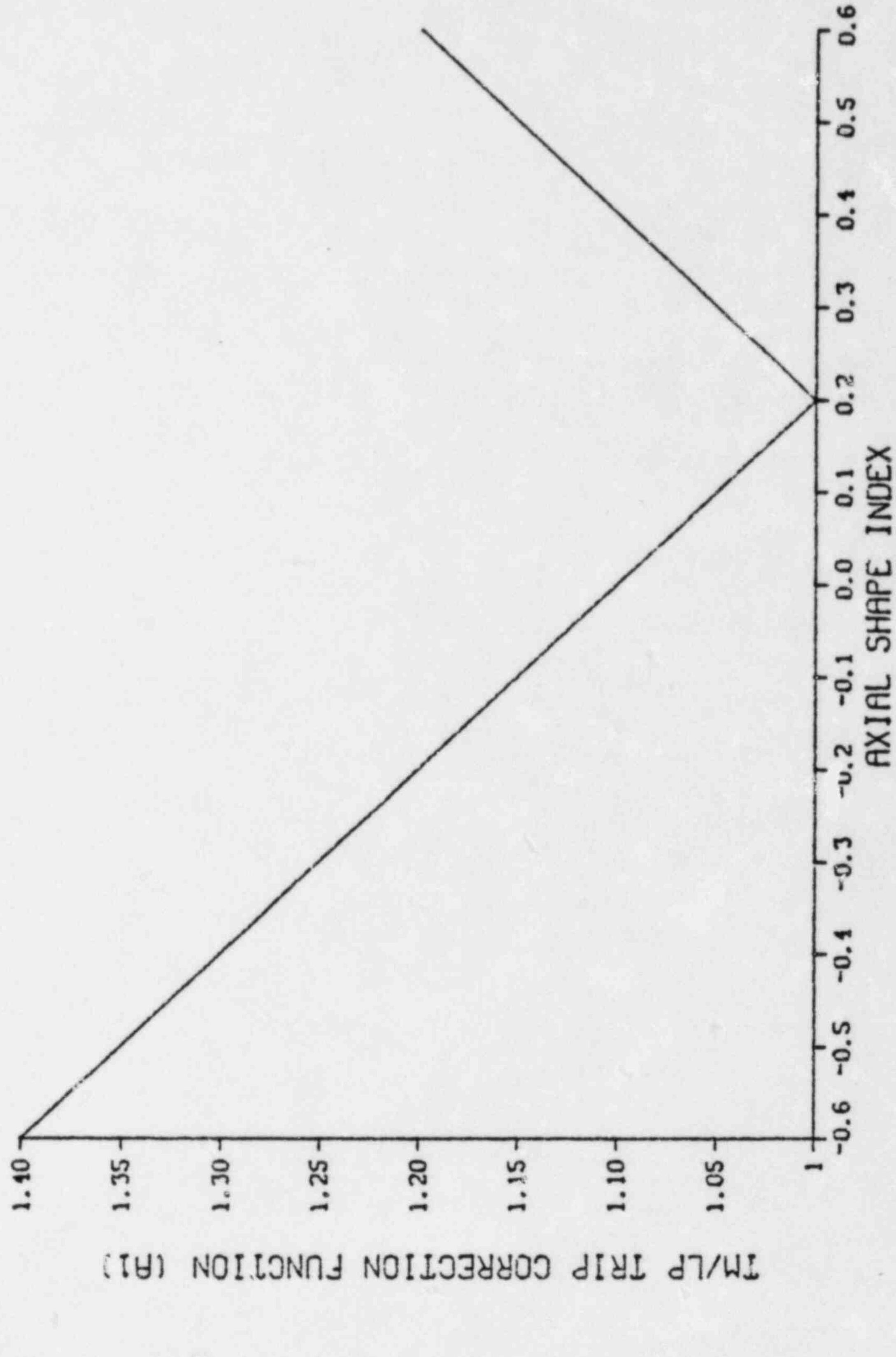


Figure 9.2 St. Lucie Unit 1 TM/LP Trip Function AI

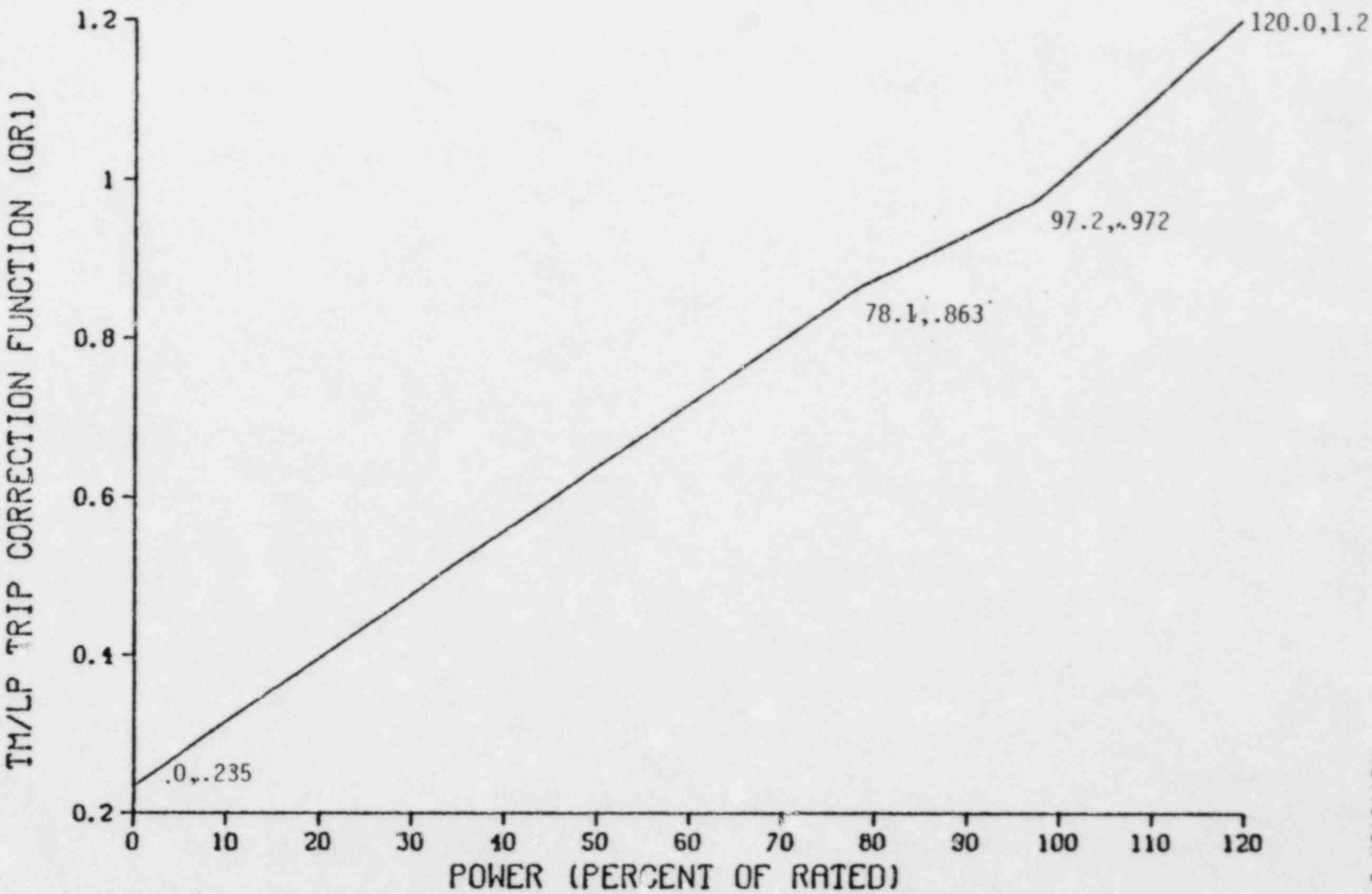


Figure 9.3 St. Lucie Unit 1 TM/LP Trip Function QR1

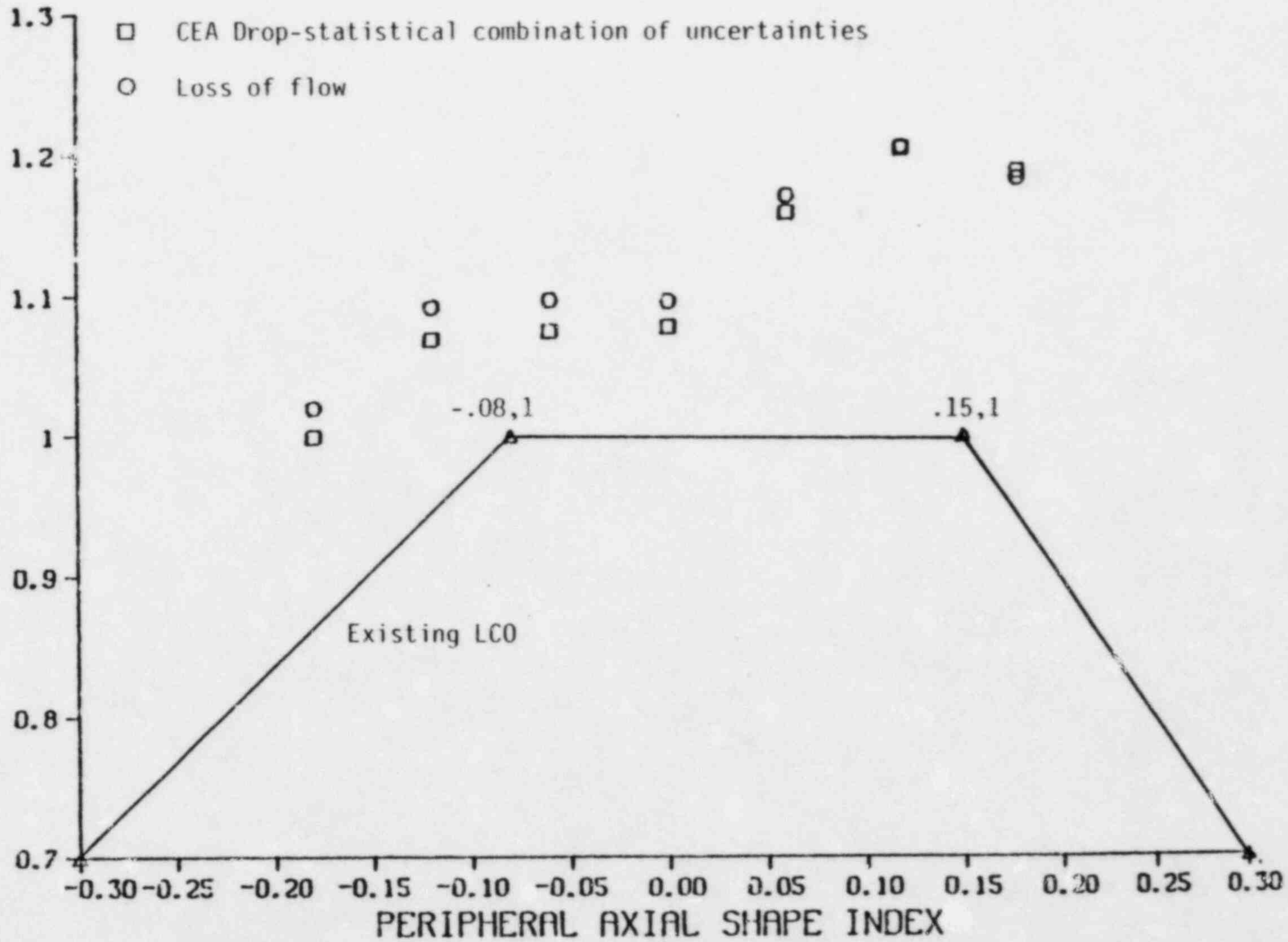


Figure 9.4 St. Lucie Unit 1 Limiting Condition for Operation DNB Barn

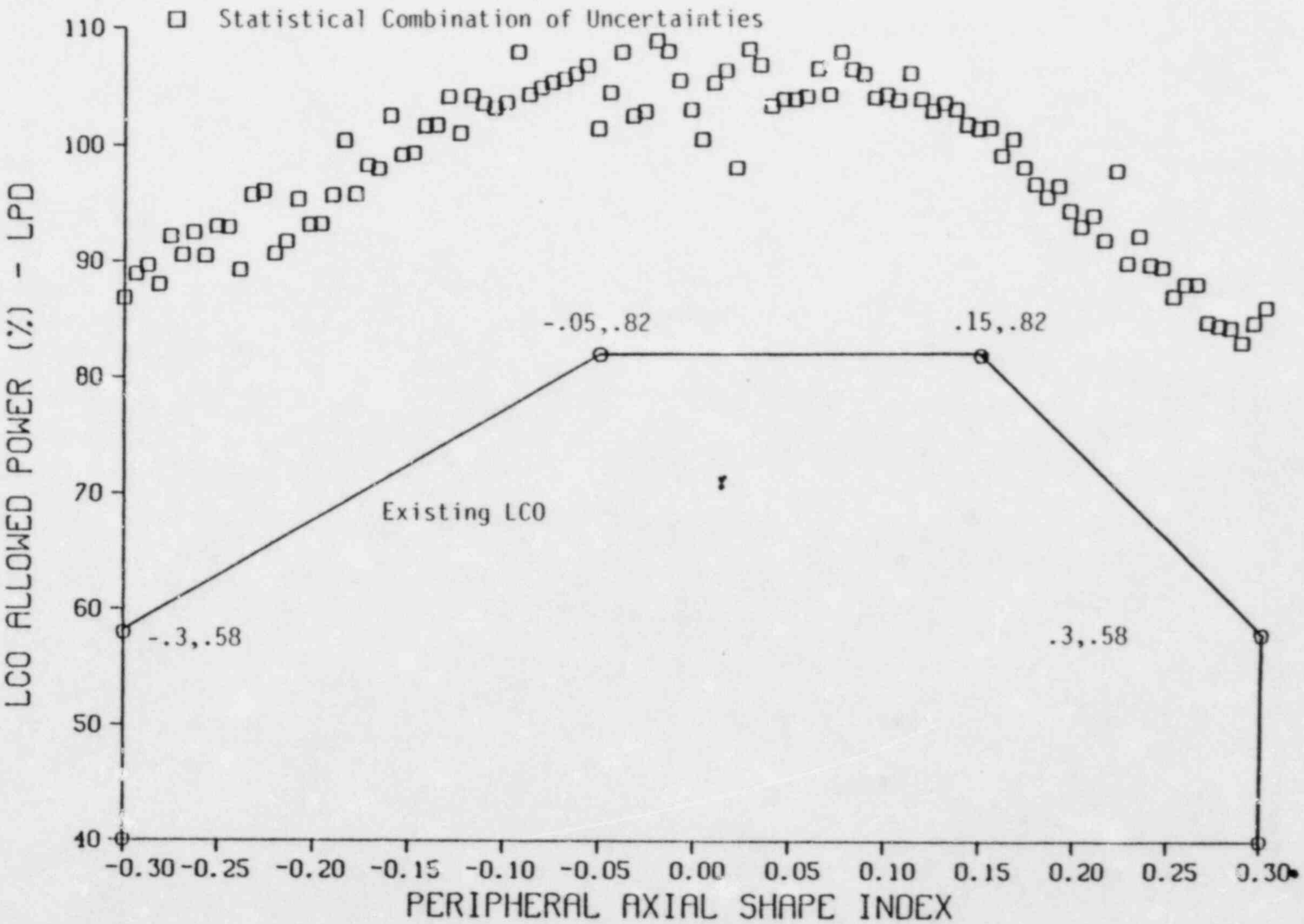


Figure 9.5 St. Lucie Unit 1 Limiting Condition for Operation LHR Barn

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