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Nuclear

June 18, 2001

SVP-01-074

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Revision to Core Operating Limits Report for Quad Cities Units 1 and 2

In accordance with Technical Specifications section 5.6.5 "Core Operating Limits Report (COLR)" revisions to each units COLR are provided in the attachments.

On February 28, 2001 Siemens Power Corporation (SPC) notified Exelon of an error in the plant transient analysis codes affecting the operating limit Minimum Critical Power Ratio (MCPR) and vessel peak pressure event This error consisted of an inconsistency in the thermal conductivity constants used in the MICROBURN and COTRANSA2 fuel codes. The error resulted in an increase in delta CPR of up to 0.01 and an increase in the peak dome pressure of 1psi.

A historical review was conducted that confirmed that neither unit had been operated outside of their respective analysis. The current cycle transient analysis was also reviewed to confirm sufficient margin was present.

Framatome (formally SPC), has completed reanalysis of the affected documents and has provided the results of these analysis to Exelon. The required revisions to the COLR for Quad Cities Unit 1 Cycle 17 and Quad Cities Unit 2 Cycle 16 have been incorporated and verified.

Additionally, as a result of the GE9/GE10 LHGR improvement program, the LHGR limits for GE fuel will now be exposure based for Unit 2. This change was implemented on Unit One at the beginning of Cycle 17 in November of 2000.

A001

Should you have any questions concerning this letter, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

George P. Barnes for

Timothy J. Tulon
Site Vice President
Quad Cities Nuclear Power Station

Attachments:

Attachment A: Quad Cities Unit 1, Core Operating Limits Report Cycle 17
Attachment B: Quad Cities Unit 2, Core Operating Limits Report Cycle 16

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

Attachment A

Quad Cities Unit 1

Core operating limits report cycle17

Core Operating Limits Report

For

Quad Cities Unit 1 Cycle 17

May 2001

ISSUANCE OF CHANGES SUMMARY

Affected Section	Affected Pages	Summary of Changes	Date
All	All	Original Issue (Cycle 17)	10/00
1, 2, 3, 4 and 5	1-1, 2-1, 3-1, 4-1, 4-2, and 5-1	Updated for ITS	4/01
References, List of Tables, 4	iv, v, 4-1, and 4-2	Updated for revised MCPR limits as a corrective action to thermal conductivity issue (10CFR21 Notification Event #37874)	5/01
Table of Contents,3, 4	ii, 3-1, 4-1	Editorial changes and typographical error corrections	

TABLE OF CONTENTS

SPECIAL INSTRUCTIONS

1. This Core Operating Limits Report (COLR) contains the applicable reactor core limits and operational information mandated by Technical Specifications Section 5.6.5. When the COLR is referenced by applicable Technical Specifications or procedures for Technical Specification compliance, a controlled copy of this report shall be used as the official source of the applicable limit or requirement.

REFERENCES

1. Commonwealth Edison Company and MidAmerican Energy Company Docket No. 50-254, Quad Cities Station, Unit 1 Facility Operating License, License No. DPR-29.

2. Letter from D.M. Crutchfield to All Power Reactor Licenses and Applicants, Generic Letter 88-16; Removal of Cycle-Specific Parameter Limits from Technical Specifications.

3. "Quad Cities Unit 1 Cycle 17 Neutronics Licensing Report", Document ID # DG00-001158.

4. Quad Cities Nuclear Power Station, Units 1 and 2, SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-31345P, Revision 2, Class III, July 1989 (as amended).

5. EMF-96-037(P), Rev. 1, "Quad Cities Extended Operating Domain (EOD) and Equipment Out Of Service (EOOS) Safety Analysis for ATRIUM-9B Fuel", September 1996, NFS NDIT # 9600134 Seq 02.

6. EMF-2415, "Quad Cities Unit 1 Cycle 17 Plant Transient Analysis", Rev. 0, August 2000.

7. EMF-2416, "Quad Cities Unit 1 Cycle 17 Reload Analysis", Rev. 0, August 2000.

8. EMF-96-185(P), Revision 4, "Quad Cities LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-9B Fuel", August 1998, NDIT # NFM970015 Seq 3.

9. DEG:98:177, "Permission to Send the NRC Nonproprietary Transient Analysis and Reload Analysis Reports", D.E. Garber to R.J. Chin, June 1, 1998.

10. GE DRF C51-00217-01, “Instrument Setpoint Calculation Nuclear Instrumentation, Rod Block Monitor, Quad Cites 1 & 2”, December 14, 1999.

11. DEG:00:091, “Revised Measured Nodal Power Distribution Uncertainty for POWERPLEX Operation with Uncalibrated LPRMs”, David Garber to Dr. R. J. Chin, April 5, 2000.

12. J11-03692-LHGR, Revision 1, Class 3, February 2000, “ComEd GE9/GE10 LHGR Improvement Program”, Document ID# DG00-000467.

13. DEG:01:077, “Quad Cities Unit 1 Cycle 17 Evaluation of Fuel Thermal Conductivity (Non-Proprietary Version for Exelon),” David Garber to Dr. R. J. Chin, May 14, 2001.

LIST OF TABLES

1.0 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

1.1 TECHNICAL SPECIFICATION REFERENCE:

TS 3.3.2.1, Table 3.3.2.1-1 (COLR 1.2) and
TS 3.4.1 (COLR 1.3)

1.2 DESCRIPTION (TLO):

The Rod Withdrawal Block Monitor Upscale Instrumentation Allowable Value for two recirculation loop operation is determined from the following relationship:

$$\leq (0.65)\text{Wd} + 56.1\% \ **$$

1.3 DESCRIPTION (SLO):

The Rod Withdrawal Block Monitor Upscale Instrumentation Allowable Value for Single Loop Operation (SLO) is determined from the following relationship:

$$\leq (0.65)\text{Wd} + 51.4\% \ **$$

** Clamped with an allowable value not to exceed the allowable value for recirculation loop drive flow (Wd) of 100%

Wd is the percent of drive flow required to produce a rated core flow of 98 million lb/hr. Trip level setting is in percent of rated power (2511 MWth).

2.0 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

2.1 TECHNICAL SPECIFICATION REFERENCE:

TS 3.2.1 (COLR 2.2.b) and
TS 3.4.1 (COLR 2.3)

2.2 DESCRIPTION:

a. For operation with uncalibrated LPRMs from BOC to 500 MWd/MT a penalty of 15.52% must be applied to all MAPLHGR limits.

b. The base MAPLHGR limits are determined as follows:

The Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) vs. Average Planar Exposure for GE10-P8HXB311-8GZ-100M-145-CECO is determined from Table 2-1.

The Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) vs. Average Planar Exposure for GE10-P8HXB312-7GZ-100M-145-CECO is determined from Table 2-2.

The Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) vs. Average Planar Exposure for GE10-P8HXB332-8G5.0-100M-145-CECO is determined from Table 2-3.

The Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) vs. Average Planar Exposure for GE10-P8HXB333-4G5.0/6G4.0-100M-145-CECO is determined from Table 2-4.

The Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) vs. Average Planar Exposure for SPCA9-3.48B-11G6.5-ADV, SPCA9-3.60B-11G6.5-ADV, SPCA9-383B-11GZH-ADV, and SPCA9-382B-12GZL-ADV is determined from Table 2-5.

2.3 SINGLE LOOP OPERATION MULTIPLIER:

The tabulated values are multiplied by 0.85 for GE fuel and 0.90 for SPC fuel whenever Quad Cities enters Single Loop Operation.

TABLE 2-1

MAPLHGR vs. Average Planar Exposure for GE10-P8HXB311-8GZ-100M-145-CECO

LATTICE 1807: P8HXL071-8GE-100M-T
LATTICE 1806: P8HXL335-8G3.0-100M-T
LATTICE 1805: P8HXL353-2G4.0/6G3.0-100M-T
LATTICE 1804: P8HXL335-4G4.0/4G3.0-100M-T
LATTICE 1054: P8HXL071-NOG-100M-T

AVERAGE PLANAR EXPOSURE (GWD/ST)	**MAPLHGR LIMITS (KW/FT)**				
	1054	**1806**	**1805**	**1804**	**1807**
0.0	11.85	12.06	11.10	12.02	11.85
0.2	11.78	12.12	11.14	12.08	11.78
1.0	11.59	12.28	11.27	12.22	11.59
2.0	11.57	12.48	11.51	12.40	11.57
3.0	11.61	12.68	11.81	12.57	11.61
4.0	11.68	12.89	12.14	12.76	11.68
5.0	11.75	13.11	12.50	12.94	11.75
6.0	11.81	13.29	12.88	13.12	11.81
7.0	11.86	13.41	13.19	13.28	11.86
8.0	11.91	13.47	13.28	13.40	11.91
9.0	11.94	13.48	13.34	13.46	11.94
10.0	11.97	13.46	13.39	13.49	11.97
12.5	11.75	13.34	13.44	13.33	11.75
15.0	11.38	12.96	13.09	12.95	11.38
20.0	10.59	12.22	12.40	12.22	10.59
25.0	9.81	11.51	11.73	11.50	9.81
27.22	12.314	12.314	12.314	12.314	12.314
48.08	10.800	10.800	10.800	10.800	10.800
58.97	6.000	6.000	6.000	6.000	6.000

TABLE 2-2

MAPLHGR vs. Average Planar Exposure for GE10-P8HXB312-7GZ-100M-145-CECO

LATTICE 1811: P8HXL071-7GE-100M-T
LATTICE 1810: P8HXL336-7G3.0-100M-T
LATTICE 1809: P8HXL354-1G4.0/6G3.0-100M-T
LATTICE 1808: P8HXL336-3G4.0/4G3.0-100M-T
LATTICE 1054: P8HXL071-NOG-100M-T

AVERAGE PLANAR EXPOSURE (GWD/ST)	**MAPLHGR LIMITS (KW/FT)**				
	1054	**1810**	**1809**	**1808**	**1811**
0.0	11.85	12.04	11.27	12.01	11.85
0.2	11.78	12.11	11.31	12.08	11.78
1.0	11.59	12.27	11.42	12.23	11.59
2.0	11.57	12.49	11.65	12.43	11.57
3.0	11.61	12.72	11.93	12.65	11.61
4.0	11.68	12.96	12.24	12.88	11.68
5.0	11.75	13.15	12.58	13.09	11.75
6.0	11.81	13.30	12.94	13.22	11.81
7.0	11.86	13.41	13.15	13.32	11.86
8.0	11.91	13.46	13.32	13.40	11.91
9.0	11.94	13.47	13.43	13.45	11.94
10.0	11.97	13.45	13.50	13.47	11.97
12.5	11.75	13.35	13.45	13.35	11.75
15.0	11.38	12.97	13.10	12.97	11.38
20.0	10.59	12.24	12.41	12.23	10.59
25.0	9.81	11.52	11.74	11.51	9.81
27.22	12.314	12.314	12.314	12.314	12.314
48.08	10.800	10.800	10.800	10.800	10.800
58.97	6.000	6.000	6.000	6.000	6.000

TABLE 2-3

MAPLHGR vs. Average Planar Exposure for GE10-P8HXB332-8G5.0-100M-145-CECO

LATTICE 1054: P8HXL071-NOG-100T-T
LATTICE 2080: P8HXL358-8G5.0-100T-T
LATTICE 2081: P8HXL377-8G5.0-100T-T
LATTICE 2082: P8HXL071-8GE-100T-T

AVERAGE PLANAR EXPOSURE (GWD/ST)	**MAPLHGR LIMITS (KW/FT)**			
	1054	**2080**	**2081**	**2082**
0.0	11.85	11.98	11.55	11.85
0.2	11.78	12.05	11.58	11.78
1.0	11.59	12.18	11.65	11.59
2.0	11.57	12.33	11.80	11.57
3.0	11.61	12.48	11.97	11.61
4.0	11.68	12.57	12.11	11.68
5.0	11.75	12.67	12.25	11.75
6.0	11.81	12.77	12.38	11.81
7.0	11.86	12.88	12.47	11.86
8.0	11.91	12.85	12.57	11.91
9.0	11.94	12.83	12.67	11.94
10.0	11.97	12.84	12.77	11.97
12.5	11.75	13.05	12.92	11.75
15.0	11.38	12.89	12.77	11.38
20.0	10.59	12.17	12.24	10.59
25.0	9.81	11.46	11.50	9.81
27.22	12.314	12.314	12.314	12.314
48.08	10.800	10.800	10.800	10.800
58.97	6.0000	6.000	6.000	6.000

TABLE 2-4

MAPLHGR vs. Average Planar Exposure for GE10-P8HXB333-4G5.0/6G4.0-100M-145-CECO

LATTICE 1054: P8HXL071-NOG-100T-T
LATTICE 2077: P8HXL358-4G5.0/6G4.0-100T-T
LATTICE 2078: P8HXL377-4G5.0/6G4.0-100T-T
LATTICE 2079: P8HXL071-10GE-100T-T

AVERAGE PLANAR EXPOSURE (GWD/ST)	**MAPLHGR LIMITS (KW/FT)**			
	1054	**2077**	**2078**	**2079**
0.0	11.85	11.81	11.22	11.85
0.2	11.78	11.86	11.26	11.78
1.0	11.59	11.95	11.36	11.59
2.0	11.57	12.11	11.52	11.57
3.0	11.61	12.25	11.69	11.61
4.0	11.68	12.40	11.88	11.68
5.0	11.75	12.56	12.08	11.75
6.0	11.81	12.72	12.29	11.81
7.0	11.86	12.85	12.46	11.86
8.0	11.91	12.89	12.61	11.91
9.0	11.94	12.94	12.76	11.94
10.0	11.97	13.00	12.90	11.97
12.5	11.75	13.14	13.02	11.75
15.0	11.38	12.90	12.79	11.38
20.0	10.59	12.17	12.24	10.59
25.0	9.81	11.46	11.50	9.81
27.22	12.314	12.314	12.314	12.314
48.08	10.800	10.800	10.800	10.800
58.97	6.0000	6.000	6.000	6.000

TABLE 2-5

MAPLHGR vs. Average Planar Exposure for SPCA9-3.48B-11G6.5-ADV SPCA9-3.60B-11G6.5-ADV SPCA9-383B-11GZH-ADV and SPCA9-382B-12GZL-ADV

AVERAGE PLANAR EXPOSURE (GWD/MTU)	ATRIUM-9B MAPLHGR (KW/FT)
0.0	13.5
20.0	13.5
60.0	8.7
61.1	8.6

3.0 LINEAR HEAT GENERATION RATE (LHGR)

3.1 TECHNICAL SPECIFICATION REFERENCE:

TS 3.2.3 and
TS 3.2.4

3.2 DESCRIPTION

A. For operation with uncalibrated LPRMs from BOC to 500 MWd/MT a penalty of 15.52% must be applied to all LHGR limits.

B. The LHGR limit for the GE fuel types in the Q1C17 core are as follows:

GE10-P8HXB311-8GZ-100M-145-CECO

NODAL EXPOSURE (GWD/MTU)	LHGR (KW/ft)
0.0	14.40
12.87	14.40
27.16	12.31
48.91	10.80
60.61	6.00

GE10-P8HXB312-7GZ-100M-145-CECO

NODAL EXPOSURE (GWD/MTU)	LHGR (KW/ft)
0.0	14.40
13.00	14.40
27.27	12.31
49.01	10.80
60.70	6.00

GE10-P8HXB332-8G5.0-100M-145-CECO

NODAL EXPOSURE (GWD/MTU)	LHGR (KW/ft)
0.0	14.40
12.75	14.40
27.25	12.31
48.97	10.8
60.62	6.00

GE10-P8HXB333-4G5.0/6G4.0-100M-145-CECO

NODAL EXPOSURE (GWD/MTU)	LHGR (KW/ft)
0.0	14.40
12.69	14.40
27.11	12.31
48.87	10.80
60.54	6.00

C. The LHGR limits are provided in Table 3-1 for all of the SPC fuel types (ATRIUM-9B Offset) in the Q1C17 core.

The Protection Against Power Transient LHGR Limits for ATRIUM-9B Offset fuel are provided in Table 3-2.

TABLE 3-1

LHGR vs AVERAGE PLANAR EXPOSURE for ATRIUM-9B Steady State

AVERAGE PLANAR EXPOSURE (GWD/MTU)	ATRIUM-9B LHGR (KW/FT)
0.0	14.4
15.0	14.4
61.1	8.32

TABLE 3-2

LHGR vs AVERAGE PLANAR EXPOSURE for ATRIUM-9B Transient

AVERAGE PLANAR EXPOSURE (GWD/MTU)	ATRIUM-9B LHGR (KW/FT)
0.0	19.4
15.0	19.4
61.1	11.2

4.0 MINIMUM CRITICAL POWER RATIO (MCPR)

4.1 TECHNICAL SPECIFICATION REFERENCE:

TS 2.1.1.2,
TS 3.2.2 and
TS 3.4.1

4.2 DESCRIPTION

The MCPR Operating Limits are based on the dual loop MCPR Safety Limit of 1.11. For Single Loop Operation the MCPR Safety Limit is 1.12 which increases the MCPR Operating Limit by 0.01. The MCPR Safety Limit is based on the following equipment conditions:

50% of the LPRMs out of service
40% of the TIPs out of service
2500 EFPH LPRM calibration interval
Operation with uncalibrated LPRMs at startup
Single Loop Operation
No reused channels

NOTE: For operation with uncalibrated LPRMs at BOC, analysis results support these limits for cycle exposures up to 500.0 MWd/MTU and therefore, the Q1C17 MCPR Operating Limits are bounding.

The MCPR Operating Limits are based on a 15 psi reduction in steam dome pressure and Technical Specification SCRAM speeds.

The Operating Limit MCPR shall be determined as follows:

1. During steady-state operation at rated core flow, the Operating Limit MCPR shall be greater than or equal to the limits provided in Table 4-1 for the appropriate operating conditions.

2. During off-rated flow conditions in Manual Flow Control Mode, the Operating Limit MCPR for each fuel type at a specific core flow condition shall be determined from the greater of the following:

   a. Table 4-2 using the appropriate flow rate, or

   b. Table 4-1 using the appropriate operating condition.

Percent Rated Core Flow based on 98 MLB/hr with 110% Maximum Flow in Manual Flow Control. (Technical Requirements Manual 2.1.a.1 and Bases of TS 3.2.2)

3. During off-rated flow conditions in Automatic Flow Control Mode, the Operating Limit MCPR for each fuel type at a specific core flow condition shall be determined from Table 4-3, Table 4-4, or Table 4-5 using the appropriate operating conditions. *Percent Rated Core Flow based on 98 MLB/hr with 108% Maximum Flow in Automatic Flow Control Operation* (Technical Requirements Manual 2.1.a.1 and Bases of TS 3.2.2).

4. During PLU Out of Service Conditions a 0.980 MFLCPR Administrative Limit shall be used.

TABLE 4-1
Q1C17 Operating Limit MCPRs based on 1.11 SLMCPR

	GE10 OLMCPR	**ATRIUM-9B OLMCPR**
Normal Operation (Supports ICF and RVOOS)	1.51	1.47
EOD/EOOS Operation (FFTR, FHOOS, Coastdown, or any combination thereof)	1.55	1.51
1 Bypass Valve OOS* (Supports ICF and RVOOS)	1.51	1.47
All Bypass Valves OOS* (Supports ICF and RVOOS)	1.56	1.52

* Operation with bypass valves out-of-service (one or all) is **not** supported during coastdown.

TABLE 4-2
Q1C17 Operating Limit MCPRs for Manual Flow Control

Total Core Flow (% of Rated)	GE10 OLMCPR	ATRIUM-9B Offset OLMCPR
110	1.11	1.11
30	2.00	2.05
0	2.56	2.59

TABLE 4-3
Q1C17 Operating Limit MCPRs for Automatic Flow Control
(Normal Operation or 1 Bypass Valve OOS)

Total Core Flow (% of Rated)	GE10 OLMCPR	ATRIUM-9B Offset OLMCPR
108	1.51	1.47
30	2.83	2.82
0	3.73	3.68

TABLE 4-4
Q1C17 Operating Limit MCPRs for Automatic Flow Control EOD/EOOS

Total Core Flow (% of Rated)	GE10 OLMCPR	ATRIUM-9B Offset OLMCPR
108	1.55	1.51
30	2.90	2.90
0	3.82	3.79

TABLE 4-5
Q1C17 Operating Limit MCPRs for Automatic Flow Control All Bypass Valves OOS

Total Core Flow (% of Rated)	GE10 OLMCPR	ATRIUM-9B Offset OLMCPR
108	1.56	1.52
30	2.92	2.92
0	3.85	3.81

5.0 ANALYTICAL METHODS

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A-14, "General Electric Standard Application for Reactor Fuel," June 2000.
2. Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," Revision 0, November 1990.
3. Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," Revision 0, April 1991.
4. Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods – Neutronic Licensing Analyses," Revision 0, April 1991.
5. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19 (P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.
6. Exxon Nuclear Methodology for Boiling Water Reactors" Application of the ENC Methodology to BWR Reloads, XN-NF-80-19 (P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.
7. Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-90-19 (P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.
8. Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis, XN-NF-80-19 (P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.
9. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67 (P)(A), Revision 1, Exxon Nuclear Company, September 1986.
10. Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06 (P)(A), Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.
11. Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014 (P)(A), Revision 1, and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.
12. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98 (P)(A), Revision 1, and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.
13. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71 (P)(A), Revision 2 Supplements 1, 2 and 3, Exxon Nuclear Company, March 1986.
14. ANFB Critical Power Correlation, ANF-1125 (P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.
15. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524 (P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.
16. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913 (P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.
17. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048 (P)(A), Advanced Nuclear Fuels Corporation, January 1993.
18. Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.
19. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125 (P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.
20. ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125 (P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

Quad Cities Unit 1 Cycle 17

COLR Attachments

Attachments to COLR

1. Neutronics Licensing Report
2. Evaluation of Fuel Thermal Conductivity Letter
3. Reload Analysis Report
4. Plant Transient Analysis Report

Attachment 1

Quad Cities Unit 1 Cycle 17

Neutronics Licensing Report

DocID#: DG00-001158

NUCLEAR FUEL MANAGEMENT
TRANSMITTAL OF DESIGN INFORMATION

☒ SAFETY RELATED
☐ NON-SAFETY RELATED
☐ REGULATORY RELATED

Originating Organization
☒ Nuclear Fuel Management
☐ Other (specify) ________

NFM ID# NFM0000100
Sequence 0
Page 1 of 18

Station: Quad Cities Unit: 1 Cycle: 17 Generic: _____

To: Nancy J. Buck-Beaumont

Subject: **Quad Cities Unit 1 Cycle 17 Neutronics Licensing Report (NLR)**

George Touvannas		7/20/00
Preparer	Preparer's Signature	Date
Peter A. Weggeman		7.28.00
Reviewer	Reviewer's Signature	Date
Adelmo S. Pallotta		8/7/00
BND Group Lead	Approver's Signature	Date

Status of Information:
☒ Verified
☐ Unverified
☐ Engineering Judgement

Action Tracking # for Method and Schedule of Verification for Unverified DESIGN INFORMATION : ________________

Description of Information:
Seq. 0: Results of the Q1C17 Neutronics Licensing calculations performed by NFM.

Purpose of Information: Provide station and BSS group with the results of the Q1C17 Neutronics Licensing calculations

Source of Information: NFM calculation note BNDQ:00-043.

Supplemental Distribution:
Alex L. Misak (QC)
Adelmo S. Pallotta
Downers Grove Central File
David F. Schumacher (QC)
Quad Cities Central File

Licensing Basis

This document, in conjunction with References 1, 2, and 3, provides the licensing basis for Quad Cities Unit 1 Reload 16, Cycle 17. The calculations that support this report are given in References 4 through 12.

Table of Contents

author: GT 7/20/00

reviewer PAW 7-28-00

I. Nuclear Design

I.1 New Reload Fuel Assembly Nuclear Design

I.1.1 Assembly Average Enrichment

Assembly Name	Enrichment (w/o U-235)
SPCA9-383B-11GZH-ADV	3.83
SPCA9-382B-12GZL-ADV	3.82

I.1.2 Axial Enrichment and Burnable Poison Distribution

Assembly Name	Figure
SPCA9-383B-11GZH-ADV	1
SPCA9-382B-12GZL-ADV	1

I.1.3 Radial Enrichment and Burnable Poison Distribution

Lattice Name	Assembly Found In	Figure
SPCA9-4.15L-11G6.0	SPCA9-383B-11GZH-ADV	2
SPCA9-4.15L-11G8.0	SPCA9-383B-11GZH-ADV	3
SPCA9-4.32L-10G8.0	SPCA9-383B-11GZH-ADV	4
SPCA9-4.14L-10G6.0	SPCA9-382B-12GZL-ADV	5
SPCA9-4.14L-11G7.0	SPCA9-382B-12GZL-ADV	6
SPCA9-4.30L-11G7.0	SPCA9-382B-12GZL-ADV	7
SPCA9-4.30L-12G7.0	SPCA9-382B-12GZL-ADV	8

author: GT 7/20/03

reviewer PAW
7·28·03

I.2 Core Nuclear Design

I.2.1 Core Configuration and Licensing Exposure Limits

Assembly Name	Cycle Loaded	Number in Core
GE10-P8HXB311-8GZ-100M-145-CECO	14	40
GE10-P8HXB312-7GZ-100M-145-CECO	14	16
GE10-P8HXB332-8G5.0-100M-145-CECO	15	144
GE10-P8HXB333-4G5.0/6G4.0-100M-145-CECO	15	88
SPCA9-3.48B-11G6.5-ADV	16	152
SPCA9-3.60B-11G6.5-ADV	16	48
SPCA9-383B-11GZH-ADV	17	92
SPCA9-382B-12GZL-ADV	17	144

Basis	EOC 16 Core Average Exposure	EOC16 Cycle Incremental Exposure	BOC17 Core Average Exposure
Nominal EOC 16 (MWD/MTU)	29,220.0	12,834.0	17,112.1
Short EOC 16 (MWD/MTU)	28,685.9	12,300.0	16,673.2

Cycle 17 neutronics analyses are valid for EOC 16 exposures greater than 12,300.0 MWD/MT. The exposure window that validates the pressurization transients can be found in the Q1C17 reload analysis document.
The Cycle 17 incremental energy to LFPC is 1717.3 GWD (13,935 MWD/MT) based on a nominal EOC 16.

I.2.2 Core Reactivity Characteristics

All values reported below are for zero xenon at 68°F moderator temperature. The MICROBURN-B cold BOC target K-effective is 1.0060. The shutdown margin calculations are based on the short cycle 16 exposure given in Section I.2.1.

BOC Cold K-Effective, All Rods Out	1.11113
BOC Cold K-Effective All Rods In	0.96389
BOC Cold K-Effective, Strongest Rod Out	0.99423
BOC Shutdown Margin, % ΔK	1.17
Minimum Shutdown Margin, % ΔK	1.17
Cycle Exposure of Minimum Shutdown Margin, MWD/MT	0
Reactivity Defect (R-value) Total, % ΔK (including 0.04 adder for inverted tubes in Quad Cities 1)	0.04
Standby Liquid Control System Shutdown Margin, Cold Condition, 600 ppm, % ΔK	4.11

author: GT 7/20/00

reviewer PAW 7.27.00

II. Control Rod Withdrawal Error

Analysis was performed at 100% power, 100% flow, unblocked conditions only.

The value reported below bounds all fuel types found in the core.

Distance Withdrawn (ft)	ΔCPR
12	0.34

The design complies with the SPC 1% plastic strain and centerline melt criteria via conformance to the PAPT (Protection Against Power Transient) LHGR limits. The design also complies with the GE 1% plastic strain criteria via conformance to the GE mechanical overpower protection (MOP) criteria during a control rod withdrawal error event. The design complies with the GE centerline melt criteria via conformance to the GE thermal overpower protection (TOP) criteria during a control rod withdrawal error event.

III. Fuel Loading Error

The fuel loading error, including fuel mislocation and misorientation, is classified as an accident. By demonstrating that the fuel loading error meets the more stringent Anticipated Operational Occurrence (AOO) requirements, the offsite dose requirement is assured to be met. Because the fuel loading error results in a ΔCPR value that is less than that of the limiting transient, the AOO requirements and hence the off-site dose requirements are met for the fuel loading error.

The values reported below bound all bundle types found in the core.

Event	ΔCPR
Mislocated Bundle	0.20

The ΔCPR for the misoriented bundle error is bounded by the results for the mislocated bundle. The misoriented bundle ΔCPR is primarily effected by the pin enrichments and the gadolinia bearing rods near the wide-wide gap in the normal and misoriented configurations (i.e. local peaking), and by the changes in the bundle reactivity caused by changes in the water gaps associated with the misoriented configuration. For the specific reload bundle designs in QC1 C17, the misorientation does not result in significantly higher local peaking factors or higher bundle reactivities. The calculations performed for the misoriented bundle analysis have shown ΔCPR's significantly less than that from the mislocated bundle analysis.

For the fuel loading error, the design complies with the SPC 1% plastic strain and centerline melt criteria via conformance to the PAPT (Protection Against Power Transient) LHGR limits.

author: ET 7/31/00

reviewer PAW 7-31-00

IV. Control Rod Drop Accident

Quad Cities is a banked position withdrawal sequence plant. In order to allow the site the option of shutting down the reactor by inserting control rods using the simplified control rod sequences shown in Table 1, a control rod drop accident analysis was performed for the simplified sequences. The results from these simplified sequence analyses bound those where BPWS guidelines are followed. The results demonstrate that the simplified shutdown sequences meet the Technical Specification limit of 280 cal/gm for a control rod drop accident. Therefore, the simplified sequences are valid for control rod insertion for shutdown.

An adder of 0.32 %Δk is incorporated in this analysis (for other than 00 to 48 control rod drops) to account for possible rod mispositioning errors as well as clumping effects.

Maximum Dropped Control Rod Worth, %Δk (value corresponds to a 00 to 12 control rod drop)	1.53
Doppler Coefficient Used, (Δk/k)/°F	-9.67E-06
Effective Delayed Neutron Fraction Used	0.00523
Four-Bundle Local Peaking Factor Used	1.234
Maximum Deposited Fuel Rod Enthalpy, (cal/gm)	246
Number of Rods Greater than 170 cal/gm	772

Note that the limit on maximum deposited fuel rod enthalpy is 280 cal/gm and the limit on the number of rods greater that 170 cal/gm (failed rods) is 850 (Reference 13).

V. Loss of Feedwater Heating

The loss of feedwater heating event is analyzed at 100% of rated power for 87%, 100%, and 108% of rated flow with an assumed inlet temperature decrease of 145 °F. The event was analyzed from BOC to EOC. The ΔCPR value reported below is bounding for both the SPC and the co-resident GE fuel types and all the analyzed flows.

Event	ΔCPR
Loss of Feedwater Heating	0.13

The design complies with the SPC 1% plastic strain and centerline melt criteria via conformance to the PAPT (Protection Against Power Transient) LHGR limits. The design also complies with the GE 1% plastic strain criteria via conformance to the GE mechanical overpower protection (MOP) criteria during a loss of feedwater heating event. The design complies with the GE centerline melt criteria via conformance to the GE thermal overpower protection (TOP) criteria during a loss of feedwater heating event.

Criteria	Maximum Value (Calculated)	GE Limit
Mechanical Overpower (MOP), %	27	45
Thermal Overpower (TOP), %	22	25

author: GT 9/5/2000

reviewer PAW 9.6.00

VI. Exposure Limit Compliance

The exposure limits will be met for all the GE10 and SPC ATRIUM fuel in C17, based on the C16 nominal core average cycle exposure (29,220.0 MWD/MT) and the long C17 cycle exposure (14,435 MWD/MT). This corresponds to an EOC17 core average exposure of 31,546 MWD/MT. The table below shows the projected peak exposures and associated exposure limits.

Exposure Criteria	GE10 Projected Exposure (GWD/MT)	GE10 Exposure Limit (GWD/MT)	ATRIUM-9B Projected Exposure (GWD/MT)	ATRIUM-9B Exposure Limit (GWD/MT)
Peak Fuel Assembly	44.0	50.0**	38.6	48.0*
Peak Fuel Batch	39.1	42.0	N/A	N/A
Peak Fuel Rod	47.4	N/A	42.8	55.0*
Peak Fuel Pellet	59.2	65.0	53.8	66.0*

* The ATRIUM-9B exposure limits identified in the above table are not applicable until SPC document EMF-85-74 is added to the Technical Specifications (Tech Specs). Until that is done, the ATRIUM-9B exposure limits are 48 GWD/MT for Peak Fuel Assembly (no change), 50 GWD/MT for Peak Fuel Rod, and 60 GWD/MT for Peak Fuel Pellet.

**There is actually no peak fuel bundle exposure limit for the GE10 fuel. The limit reported above is based on the maximum channel exposure assumption used in developing the safety limit MCPR for Quad Cities 1 Cycle 17.

VII. New Fuel Vault and Spent Fuel Pool Criticality Compliance

For the Q1C17 reload, there are two new SPC ATRIUM-9B assembly types consisting of seven unique enriched lattices, as identified in Section I.1. As described in the Reference 15 transmittal, the two fresh assembly types comply with the spent fuel pool and new fuel vault criticality limits.

VII.1 New Fuel Vault Criticality Compliance

The fuel storage vault criticality analysis that is described in Reference 16 remains valid. All the new (ATRIUM-9B) assemblies comply with the fresh fuel vault criticality limits, i.e., all lattices have an enrichment of less than 5.00 wt % U-235 and a gadolinia content that is greater than 6 gadolinia rods at 2.0 wt% Gd_2O_3. Hence, the Q1C17 fresh fuel can be safely stored in the Quad Cities new fuel vault.

Note that the new fuel storage vault is a moderation controlled area which implies that hydrogenous materials will be limited within the new fuel storage array. Administrative controls as generally defined in GE SIL No. 152 (dated March 31, 1976) must continue to be incorporated for the area.

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VII.2 Spent Fuel Pool Criticality Compliance

The Quad Cities spent fuel pool criticality analyses that are described in References 17-18 remain valid. As shown below (Reference 15), all the new (ATRIUM-9B) assemblies comply with the spent fuel pool criticality limits of k-eff < 0.95.

Lattice Type	Maximum In-Core k-inf[1]	Maximum In-Rack k-eff with No Boraflex Degradation[2]	Maximum In-Rack k-eff with Boraflex Degradation[3]	Spent Fuel Pool k-eff Limit
SPCA9-4.14L-10G6.0	1.20474	< 0.85	< 0.884	0.95
SPCA9-4.14L-11G7.0	1.18481	< 0.85	< 0.884	0.95
SPCA9-4.30L-11G7.0	1.19644	< 0.85	< 0.884	0.95
SPCA9-4.30L-12G7.0	1.18998	< 0.85	< 0.884	0.95
SPCA9-4.15L-11G6.0	1.19830	< 0.85	< 0.884	0.95
SPCA9-4.15L-11G8.0	1.16647	< 0.85	< 0.884	0.95
SPCA9-4.32L-10G8.0	1.17204	< 0.85	< 0.884	0.95

[1] From 68 °F, uncontrolled CASMO-3G results.

[2] From Figure 6.1 of Reference 17.

[3] Based on Reference 18, assuming the following level of degradation of Boraflex racks:
(1) 7 inch gap in every panel
(2) 20% loss in thickness in every panel
(3) 10% loss in thickness in every panel,
a penalty of 0.034 Δk is added to the maximum in-rack k-eff values. This conservatively bounds the estimated degree of Boraflex degradation.

The table above indicates that, even with a very conservative estimate, there is a minimum of 0.06 Δk (or 6% Δk) margin to the spent fuel pool k-eff limit of 0.95.

VIII. References

1. Commonwealth Edison, Nuclear Fuel Services, NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods", as supplemented and approved.

2. Siemens Power Corporation document EMF-2416, Revision 0, "Quad Cities Unit 1 Cycle 17 Reload Analysis," August 2000.

3. Siemens Power Corporation document EMF-2415, Revision 0, "Quad Cities Unit 1 Cycle 17 Plant Transient Analysis," August 2000.

4. NFM Calculation Note BNDQ:00-023, Rev. 0, "QC1 C17 Final Licensing Loading Pattern (FLLP)," Rev. 0, March 15, 2000 and NDIT NFM0000038, Seq. 2, "Quad Cities 1 Cycle 17 Core Design – Transmittal of Letter NFM:BND:00-029, Quad Cities 1 Cycle 17 Final Licensing Loading Plan and Licensing Energies," March 15, 2000.

5. NFM Calculation Note BNDQ:00-036, Rev. 0, "Q1C17 Standby Liquid Control (SBLC) System Worth Licensing Analysis," August 28, 2000.

author: GT 9/5/2000 reviewer PAW 9.6.00

6. NFM Calculation Note BNDQ:00-042, Rev. 0, "QC1C17 Misorientation Analyses," August 28, 2000.
7. NFM Calculation Note BNDQ:00-041, Rev. 0, "Q1 C17 Fuel Assembly Mislocation Calculations," August 23, 2000.
8. NFM Calculation Note BNDQ:00-031, Rev. 0, "QC1 C17 Rod Withdrawal Error (RWE) ΔCPR Analysis," July 27, 2000.
9. NFM Calculation Note BNDQ:00-032, Rev. 0, "QC1 C17 RWE MOP/TOP Analysis," August 2, 2000.
10. NFM Calculation Note BNDQ:00-034, Rev. 0, "QC1 C17 Loss of Feedwater Heating (LFWH) Analysis," August 2, 2000.
11. NFM Calculation Note BNDQ:00-038, Rev. 0, "Q1 C17 Control Rod Drop Accident Licensing Analysis," September 5, 2000.
12. NFM Calculation Note BNDQ:00-040, Rev. 0, "Q1C17 Shutdown Margin for Licensing," August 29, 2000.
13. Quad Cities Updated Final Safety Analysis Report, Revision 4, April 1997, Section 15.4.10.
14. Siemens Power Corporation document EMF-2412(P), Revision 1, "Fuel Design Report for Quad Cities Unit 1 Cycle 17 ATRIUM-9B Fuel Assemblies," August 2000.
15. TODI NFM0000095, Seq. 0, "Quad Cities 1 Cycle 17 New Fuel Storage," June 22, 2000.
16. "Criticality Safety Analysis for ATRIUM-9B Fuel, Dresden and Quad Cities New Fuel Storage Vaults," Siemens Power Corporation, EMF-96-148(P), Revision 1, September 1996, transmitted to Quad Cities station via NFS NDIT 960127, Rev. 0, September 9, 1996.
17. "Criticality Safety Analysis for ATRIUM-9B Fuel, Quad Cities Units 1 and 2 Spent Fuel Storage Pool (Boraflex Racks)," Siemens Power Corporation, EMF-96-013(P), June 1996, transmitted to Quad Cities station via NFS NDIT 960105, Rev. 0, July 19, 1996.
18. "Results of BADGER Testing Performed on November 13-22, 1996 in Quad Cities Spent Fuel Pools," Letter from D. F. Schumacher (Quad Cities) to E. Kraft (Quad Cities), January 22, 1997.

author: GT 9/5/2000

reviewer PAW 9.6-00

Table 1

QC1 C17 Simplified Shutdown Sequence Prior to LFPC (13935 MWD/MT)

BPWS Rod Group	Insertion (Bank)	Comments
9 or 10	48 - 00	Either Group 9 or 10 may be inserted first.
8	48 - 00	Groups 9 and 10 must be fully inserted before inserting any group 8 rod.
7	48 - 12	Group 8 must be fully inserted before inserting any group 7 rod.
7	12 - 00	Group 7 must be at 12 before inserting any group 7 rod to 00.
5 or 6	48 - 00	Groups 5 and 6 may be inserted without banking anytime after Groups 9 and 10 have been inserted and before Group 4 is inserted.
4	48 - 00	Groups 5 through 10 must be fully inserted before inserting any group 4 rod.
3	48 - 00	Group 4 must be fully inserted before inserting any group 3 rod.
2	48 - 00	Group 3 must be fully inserted before inserting any group 2 rod.
1	48 - 00	Group 2 must be fully inserted before inserting any group 1 rod.

QC1 C17 Simplified Shutdown Sequence After LFPC (13935 MWD/MT)

BPWS Rod Group	Insertion (Bank)	Comments
9 or 10	48 - 00	Either Group 9 or 10 may be inserted first.
8	48 - 00	Groups 9 and 10 must be fully inserted before inserting any group 8 rod.
7	48 - 12	Group 8 must be fully inserted before inserting any group 7 rod.
7	12 - 08	Group 7 must be at 12 before inserting any group 7 rod to 08.
7	08 - 00	Group 7 must be at 08 before inserting any group 7 rod to 00.
5 or 6	48 - 00	Groups 5 and 6 may be inserted without banking anytime after Groups 9 and 10 have been inserted and before Group 4 is inserted.
4	48 - 00	Groups 5 through 10 must be fully inserted before inserting any group 4 rod.
3	48 - 00	Group 4 must be fully inserted before inserting any group 3 rod.
2	48 - 00	Group 3 must be fully inserted before inserting any group 2 rod.
1	48 - 00	Group 2 must be fully inserted before inserting any group 1 rod.

author: ET 7/31/00

reviewer PAW
7-31-00

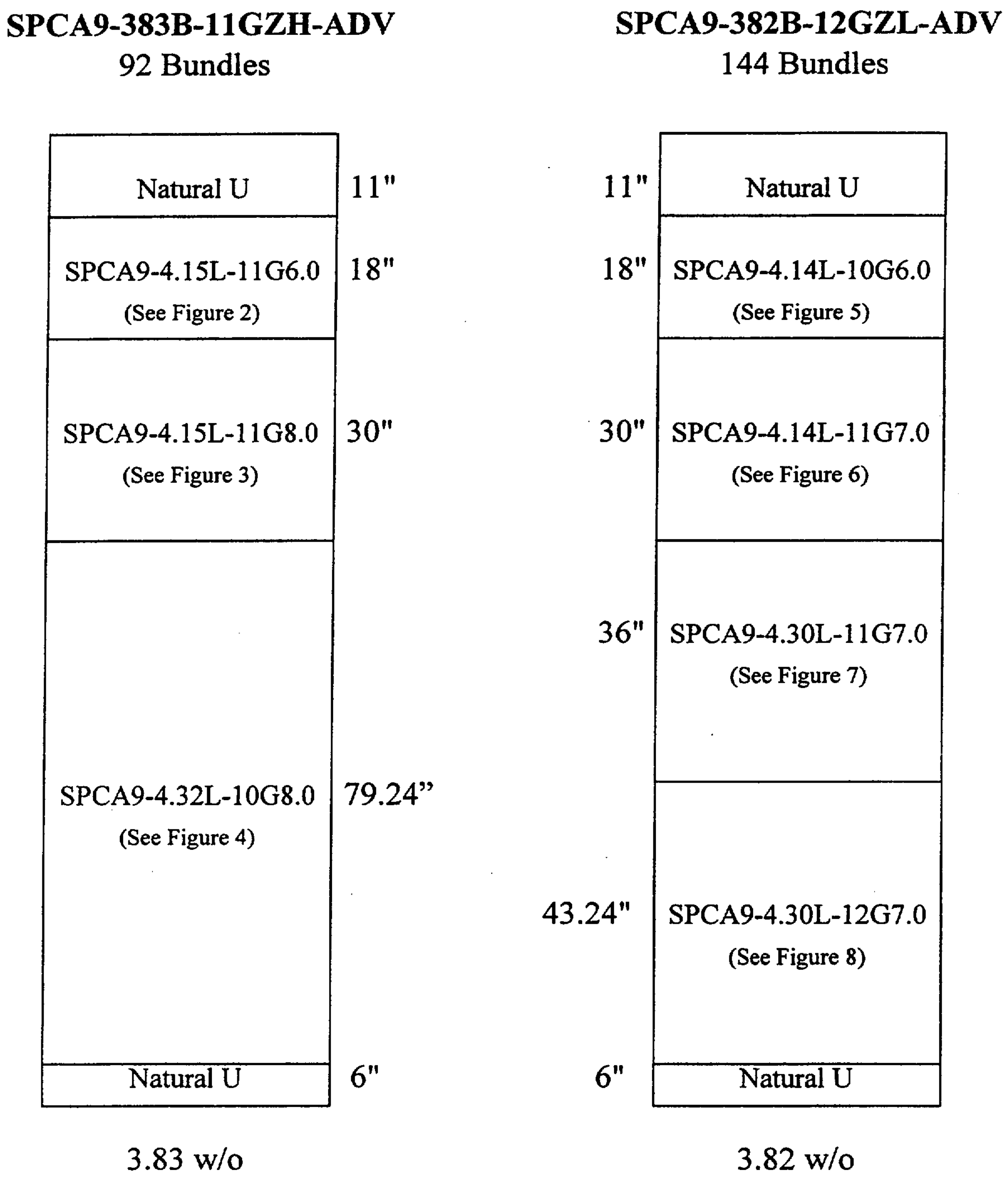

Figure 1. Q1C17 Bundle Designs

author: CT 7/20/00

reviewer DAU 7-28-00

1 2.10	2 2.80	3 3.20	4 3.45	5 4.10	5 4.10	5 4.10	3 3.20	2 2.80
2 2.80	3 3.20	5 4.10	6 3.85 6.00	8 4.70	8 4.70	6 3.85 6.00	5 4.10	4 3.45
3 3.20	5 4.10	6 3.85 6.00	8 4.70	8 4.70	8 4.70	8 4.70	6 3.85 6.00	8 4.70
4 3.45	6 3.85 6.00	8 4.70	Internal Water Channel			8 4.70	8 4.70	8 4.70
5 4.10	8 4.70	8 4.70				7 4.95	6 3.85 6.00	8 4.70
5 4.10	8 4.70	8 4.70				7 4.95	8 4.70	8 4.70
5 4.10	6 3.85 6.00	8 4.70	8 4.70	7 4.95	7 4.95	6 3.85 6.00	8 4.70	8 4.70
3 3.20	5 4.10	6 3.85 6.00	8 4.70	6 3.85 6.00	8 4.70	8 4.70	6 3.85 6.00	5 4.10
2 2.80	4 3.45	8 4.70	8 4.70	8 4.70	8 4.70	8 4.70	5 4.10	2 2.80

1	**Rods (1)**	2.10 w/o U-235
2	**Rods (5)**	2.80 w/o U-235
3	**Rods (5)**	3.20 w/o U-235
4	**Rods (4)**	3.45 w/o U-235
5	**Rods (12)**	4.10 w/o U-235
6	**Rods (11)**	3.85 w/o U-235 + 6.0 w/o Gd203
7	**Rods (4)**	4.95 w/o U-235
8	**Rods (30)**	4.70 w/o U-235

Figure 2. SPCA9-4.15L-11G6.0 **Enrichment Distribution**

author: GT 7/20/00

reviewer PAW
7-28-00

NUCLEAR FUEL MANAGEMENT
TRANSMITTAL OF DESIGN INFORMATION

NDIT No. NFM0000100
Seq. No. 0
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1 2.10	2 2.80	3 3.20	4 3.45	5 4.10	5 4.10	5 4.10	3 3.20	2 2.80
2 2.80	3 3.20	5 4.10	6 3.85 8.00	8 4.70	8 4.70	6 3.85 8.00	5 4.10	4 3.45
3 3.20	5 4.10	6 3.85 8.00	8 4.70	8 4.70	8 4.70	8 4.70	6 3.85 8.00	8 4.70
4 3.45	6 3.85 8.00	8 4.70	Internal Water Channel			8 4.70	8 4.70	8 4.70
5 4.10	8 4.70	8 4.70				7 4.95	6 3.85 8.00	8 4.70
5 4.10	8 4.70	8 4.70				7 4.95	8 4.70	8 4.70
5 4.10	6 3.85 8.00	8 4.70	8 4.70	7 4.95	7 4.95	6 3.85 8.00	8 4.70	8 4.70
3 3.20	5 4.10	6 3.85 8.00	8 4.70	6 3.85 8.00	8 4.70	8 4.70	6 3.85 8.00	5 4.10
2 2.80	4 3.45	8 4.70	8 4.70	8 4.70	8 4.70	8 4.70	5 4.10	2 2.80

1	Rods (1)	2.10 w/o U-235
2	Rods (5)	2.80 w/o U-235
3	Rods (5)	3.20 w/o U-235
4	Rods (4)	3.45 w/o U-235
5	Rods (12)	4.10 w/o U-235
6	Rods (11)	3.85 w/o U-235 + 8.0 w/o Gd203
7	Rods (4)	4.95 w/o U-235
8	Rods (30)	4.70 w/o U-235

Figure 3. SPCA9-4.15L-11G8.0 Enrichment Distribution

author: GT 7/20/00

reviewer PAU
7.28.00

1 2.10	2 2.80	3 3.20	4 3.85	5 4.10	5 4.10	5 4.10	3 3.20	2 2.80
2 2.80	3 3.20	5 4.10	6 4.10 8.00	7 4.95	7 4.95	6 4.10 8.00	5 4.10	4 3.85
3 3.20	5 4.10	7 4.95	7 4.95	7 4.95	7 4.95	7 4.95	6 4.10 8.00	7 4.95
4 3.85	6 4.10 8.00	7 4.95	Internal Water Channel			7 4.95	7 4.95	7 4.95
5 4.10	7 4.95	7 4.95				7 4.95	6 4.10 8.00	7 4.95
5 4.10	7 4.95	7 4.95				7 4.95	7 4.95	7 4.95
5 4.10	6 4.10 8.00	7 4.95	7 4.95	7 4.95	7 4.95	6 4.10 8.00	7 4.95	7 4.95
3 3.20	5 4.10	6 4.10 8.00	7 4.95	6 4.10 8.00	7 4.95	7 4.95	6 4.10 8.00	5 4.10
2 2.80	4 3.85	7 4.95	7 4.95	7 4.95	7 4.95	7 4.95	5 4.10	2 2.80

1	Rods (1)	2.10 w/o U-235
2	Rods (5)	2.80 w/o U-235
3	Rods (5)	3.20 w/o U-235
4	Rods (4)	3.85 w/o U-235
5	Rods (12)	4.10 w/o U-235
6	Rods (10)	4.10 w/o U-235 + 8.0 w/o Gd203
7	Rods (35)	4.95 w/o U-235

Figure 4. SPCA9-4.32L-10G8.0 Enrichment Distribution

author 7/20/00

reviewer PAW
7.28.00

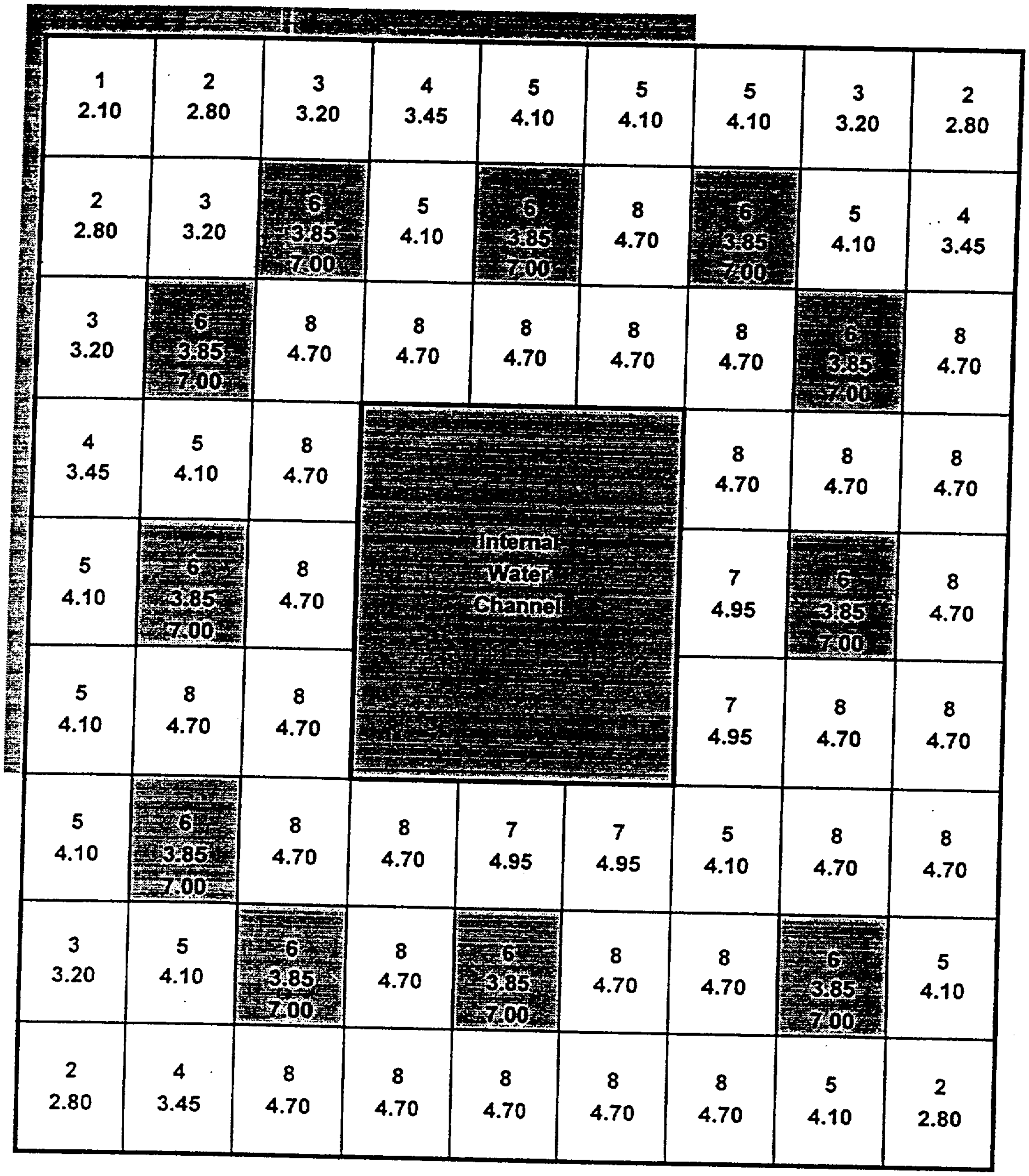
1 2.10	2 2.80	3 3.20	4 3.45	5 4.10	5 4.10	5 4.10	3 3.20	2 2.80
2 2.80	3 3.20	6 3.85 6.00	5 4.10	5 4.10	8 4.70	6 3.85 6.00	5 4.10	4 3.45
3 3.20	6 3.85 6.00	8 4.70	8 4.70	8 4.70	8 4.70	8 4.70	6 3.85 6.00	8 4.70
4 3.45	5 4.10	8 4.70	Internal Water Channel			8 4.70	8 4.70	8 4.70
5 4.10	5 4.10	8 4.70				7 4.95	6 3.85 6.00	8 4.70
5 4.10	8 4.70	8 4.70				7 4.95	8 4.70	8 4.70
5 4.10	6 3.85 6.00	8 4.70	8 4.70	7 4.95	7 4.95	6 3.85 6.00	8 4.70	8 4.70
3 3.20	5 4.10	6 3.85 6.00	8 4.70	6 3.85 6.00	8 4.70	8 4.70	6 3.85 6.00	5 4.10
2 2.80	4 3.45	8 4.70	8 4.70	8 4.70	8 4.70	8 4.70	5 4.10	2 2.80

1	Rods (1)	2.10 w/o U-235
2	Rods (5)	2.80 w/o U-235
3	Rods (5)	3.20 w/o U-235
4	Rods (4)	3.45 w/o U-235
5	Rods (14)	4.10 w/o U-235
6	Rods (10)	3.85 w/o U-235 + 6.0 w/o Gd203
7	Rods (4)	4.95 w/o U-235
8	Rods (29)	4.70 w/o U-235

Figure 5. SPCA9-4.14L-10G6.0 Enrichment Distribution

author: 6T 7/20/00

reviewer PAW
7.28.00



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2 2.80	3 3.20	6 3.85 7.00	5 4.10	6 3.85 7.00	8 4.70	6 3.85 7.00	5 4.10	4 3.45
3 3.20	6 3.85 7.00	8 4.70	8 4.70	8 4.70	8 4.70	8 4.70	6 3.85 7.00	8 4.70
4 3.45	5 4.10	8 4.70	Internal Water Channel			8 4.70	8 4.70	8 4.70
5 4.10	6 3.85 7.00	8 4.70				7 4.95	6 3.85 7.00	8 4.70
5 4.10	8 4.70	8 4.70				7 4.95	8 4.70	8 4.70
5 4.10	6 3.85 7.00	8 4.70	8 4.70	7 4.95	7 4.95	5 4.10	8 4.70	8 4.70
3 3.20	5 4.10	6 3.85 7.00	8 4.70	6 3.85 7.00	8 4.70	8 4.70	6 3.85 7.00	5 4.10
2 2.80	4 3.45	8 4.70	8 4.70	8 4.70	8 4.70	8 4.70	5 4.10	2 2.80

1	Rods (1)	2.10 w/o U-235
2	Rods (5)	2.80 w/o U-235
3	Rods (5)	3.20 w/o U-235
4	Rods (4)	3.45 w/o U-235
5	Rods (13)	4.10 w/o U-235
6	Rods (11)	3.85 w/o U-235 + 7.0 w/o Gd203
7	Rods (4)	4.95 w/o U-235
8	Rods (29)	4.70 w/o U-235

Figure 6. SPCA9-4.14L-11G7.0 **Enrichment Distribution**

author: GT 7/20/00

reviewer PAW
7-28-00

1 2.10	2 2.80	3 3.20	5 3.85	6 4.10	6 4.10	6 4.10	3 3.20	2 2.80
2 2.80	3 3.20	7 4.10 7.00	6 4.10	7 4.10 7.00	8 4.95	7 4.10 7.00	6 4.10	5 3.85
3 3.20	7 4.10 7.00	8 4.95	8 4.95	8 4.95	8 4.95	8 4.95	7 4.10 7.00	8 4.95
5 3.85	6 4.10	8 4.95				8 4.95	8 4.95	8 4.95
6 4.10	7 4.10 7.00	8 4.95		Internal Water Channel		8 4.95	7 4.10 7.00	8 4.95
6 4.10	8 4.95	8 4.95				8 4.95	8 4.95	8 4.95
6 4.10	7 4.10 7.00	8 4.95	8 4.95	8 4.95	8 4.95	6 4.10	8 4.95	8 4.95
3 3.20	6 4.10	7 4.10 7.00	8 4.95	7 4.10 7.00	8 4.95	8 4.95	7 4.10 7.00	6 4.10
2 2.80	5 3.85	8 4.95	8 4.95	8 4.95	8 4.95	8 4.95	6 4.10	2 2.80

1	Rods (1)	2.10 w/o U-235
2	Rods (5)	2.80 w/o U-235
3	Rods (5)	3.20 w/o U-235
5	Rods (4)	3.85 w/o U-235
6	Rods (13)	4.10 w/o U-235
7	Rods (11)	4.10 w/o U-235 + 7.0 w/o Gd203
8	Rods (33)	4.95 w/o U-235

Figure 7. SPCA9-4.30L-11G7.0 Enrichment Distribution

author: 7/20/00

reviewer PAW 7-28-00

NUCLEAR FUEL MANAGEMENT
TRANSMITTAL OF DESIGN INFORMATION

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Page 18 of 18

1 2.10	2 2.80	3 3.20	5 3.85	6 4.10	6 4.10	6 4.10	3 3.20	2 2.80
2 2.80	3 3.20	7 4.10 7.00	6 4.10	7 4.10 7.00	8 4.95	7 4.10 7.00	6 4.10	5 3.85
3 3.20	7 4.10 7.00	8 4.95	8 4.95	8 4.95	8 4.95	8 4.95	7 4.10 7.00	8 4.95
5 3.85	6 4.10	8 4.95	Internal Water Channel			8 4.95	8 4.95	8 4.95
6 4.10	7 4.10 7.00	8 4.95				8 4.95	7 4.10 7.00	8 4.95
6 4.10	8 4.95	8 4.95				8 4.95	8 4.95	8 4.95
6 4.10	7 4.10 7.00	8 4.95	8 4.95	8 4.95	8 4.95	7 4.10 7.00	8 4.95	8 4.95
3 3.20	6 4.10	7 4.10 7.00	8 4.95	7 4.10 7.00	8 4.95	8 4.95	7 4.10 7.00	6 4.10
2 2.80	5 3.85	8 4.95	8 4.95	8 4.95	8 4.95	8 4.95	6 4.10	2 2.80

1	Rods (1)	2.10 w/o U-235
2	Rods (5)	2.80 w/o U-235
3	Rods (5)	3.20 w/o U-235
5	Rods (4)	3.85 w/o U-235
6	Rods (12)	4.10 w/o U-235
7	Rods (12)	4.10 w/o U-235 + 7.0 w/o Gd203
8	Rods (33)	4.95 w/o U-235

Figure 8. SPCA9-4.30L-12G7.0 Enrichment Distribution

author: GT 7/20/00

reviewer PAW 7.28.00

Attachment 2

Quad Cities Unit 1 Cycle 17

Evaluation of Fuel Thermal Conductivity Letter

FRAMATOME ANP

May 14, 2001
DEG:01:077

Ori : Doc ID
Copy : RWST
JKW
A. Aisak (QC)
Copy Cover Ltr Only : ADG
C de la Hoz

Dr. R. J. Chin
Nuclear Fuel Services (Suite 400)
Exelon Corporation
1400 Opus Place
Downers Grove, IL 60515-5701

Dear Dr. Chin:

Quad Cities Unit 1 Cycle 17 Evaluation of Fuel Thermal Conductivity (Non-Proprietary Version for Exelon)

Ref.: 1. Letter, D. E. Garber to R. J. Chin (DEG:01:057) dated April 13, 2001. Subject: "Quad Cities Unit 1 Cycle 17 Evaluation of Fuel Thermal Conductivity."

The attached document is identical to that provided in Reference 1 except the proprietary statements have been removed.

Very truly yours,

DE Garber/ccm
David Garber
Project Manager

Attachment

Framatome ANP Richland, Inc.

2101 Horn Rapids Road
Richland, WA 99352
Tel: (509) 375-8100
Fax: (509) 375-8402

Quad Cities Unit 1 Cycle 17
Evaluation of Fuel Thermal Conductivity

As reported in Reference A.1, the Framatome ANP Richland, Inc. (FRA-ANP) computer code PRECOT2 produces an incorrect thermal conductivity for the fuel rod. The reference thermal conductivity is computed at 1000°F in PRECOT2 rather than the reference temperature of 1200°F used in MICROBURN-B and COTRANSA2. This inconsistency caused an 11% over-prediction of the thermal conductivity in COTRANSA2 calculations. The error has a nonconservative impact on Quad Cities Unit 1 Cycle 17 transient analysis results and thermal limits reported in References A.2 and A.3. The limiting events, as determined in References A.2 and A.3, were reanalyzed with the correction to the thermal conductivity and the revised thermal limits are provided in Table 2 of this letter report.

The limiting base case LRNB and FWCF events are at 100% power/108% flow at end of full power (EOFP). Table 1 presents base case transient results. Table 2 presents the MCPR operating limits (OLMCPRs) for base case operation based on these events. Table 3 presents key parameter results for these events.

The limiting events that determine the OLMCPR for EOD/EOOS operation are the LRNB 100% power/108% flow at coastdown and the FWCF 100% power/108% flow at combined coastdown with final feedwater temperature reduction (FFTR). Tables 5 and 6 present key parameters, ΔCPRs, and change from base case operation results for these limiting LRNB and FWCF EOD/EOOS transient events, respectively. Table 2 presents the OLMCPR for EOD/EOOS operation based on these events. The EOD/EOOS OLMCPR penalty was determined to be 0.04, which is unchanged from Reference A.2. This penalty is required to support operation with FFTR, FHOOS, Coastdown or any combination thereof. ~~when~~ Coastdown is defined as core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1 of Reference A.3. Other EOD/EOOS conditions listed in Table 2.4 of Reference A.2 require no OLMCPR penalty. per SPC 5/24/2001

The limiting EOFP FWCF event at 100% power/108% flow was analyzed with all bypass valves out-of-service (BPVOOS) and with cycle-specific parameters for 1 BPVOOS. Table 7 presents key parameters, ΔCPR, and change from base case operation results for BPVOOS operation. Table 2 presents the OLMCPRs for all BPVOOS and 1 BPVOOS operation based on these events.

The limiting ASME overpressurization event with main steam isolation valve (MSIV) closure shown in Reference A.2 was analyzed to determine the impact of the corrected thermal conductivity term. Table 2 presents the maximum pressurization summary for this event. Table 3 presents the key parameters for the ASME event.

The limiting load rejection no bypass – unpiped safety valve margin (LRNB-USM) state point was determined to be the 100% power/100% flow at coastdown with the safety/relief valve out-of-service (SRVOOS) conditions. Table 8 presents the results for this analysis.

Analysis was performed to determine the maximum fraction of the limiting critical power ratio (MFLCPR) multipliers that protect the safety limit MCPR (SLMCPR) when the power load unbalance (PLU) is out-of-service. The analysis was performed for 100% power/108% flow for both EOFP and coastdown conditions. Table 9 presents transient results and MFLCPR multipliers for PLUOOS.

Since the ATRIUM™-9B* OLMCPRs were revised, the automatic flow control (AFC) reduced flow MCPR ($MCPR_f$) limits must be analyzed. AFC $MCPR_f$ limits are needed for base case operation, EOD/EOOS conditions, and BPVOOS. Table 4 and Figures 1–3 present the reduced flow MCPR results for AFC.

FRA-ANP evaluated the impact of the fuel thermal conductivity on the Reference A.4 1% plastic strain results. A comparison between Reference A.4 and this revised analysis of maximum nodal power ratios for similar events shows a negligible increase. A large portion of this increase is still within the Reference A.4 bounding curve. The portions that are outside the bounding curve are not significantly greater than the Reference A.4 analysis. Therefore, it is concluded that the 1% plastic strain criteria for GE fuel is met.

References:

A.1 Letter, D. E. Garber (FRA-ANP) to R. J. Chin (Exelon), "Transmittal of Condition Report 9191," DEG:01:038, February 27, 2001.

A.2 EMF-2415 Revision 0, *Quad Cities Unit 1 Cycle 17 Plant Transient Analysis*, Siemens Power Corporation, August 2000.

A.3 EMF-2416 Revision 0, *Quad Cities Unit 1 Cycle 17 Reload Analysis*, Siemens Power Corporation, August 2000.

A.4 Letter, D. E. Garber (SPC) to R. J. Chin (Exelon), "Quad Cities Unit 1 Cycle 17 Transient Power History Data for Confirming Mechanical Limits for GE Fuel," DEG:00:132, June 5, 2000.

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* ATRIUM is a trademark of Framatome ANP.

Table 1 Quad Cities Unit 1 Cycle 17 Base Case ΔCPRs at Rated Power With TSSS Insertion Times

Transient	ΔCPR	
	GE10	ATRIUM-9B Offset
Load Rejection No Bypass		
100% power / 108% flow	0.39	0.36
Feedwater Flow Controller Failure		
100% power / 108% flow	0.40	0.35

Table 2 Quad Cities Unit 1 Cycle 17 MCPR Operating Limit and Maximum Pressurization Summary

MCPR Operating Limit*, †

Operating Conditions	GE10	ATRIUM-9B Offset
Base case	1.51	1.47
Base case with 1 BPVOOS	1.51	1.47
Base case with all BPVOOS	1.56	1.52
EOD/EOOS‡	1.55	1.51

Maximum Pressurization (psig)

Transient	Steam Dome	Lower Plenum	Steam Lines
MSIV closure without position scram (100% power / 100% flow, EOD/EOOS)	1336	1361	1335

---

* Based on a plant Technical Specification two-loop SLMCPR of 1.11 and analysis of the limiting system transient analyzed in this report. The actual cycle operating limit may be higher if analyses within Exelon's scope of responsibility result in a ΔCPR higher than those in Table 1. For single-loop operation, the Technical Specification SLO SLMCPR of 1.12 increases the OLMCPR by 0.01. Refer to Table 4 for reduced flow MCPR limits.

† The operating limit for GE10 is based on FWCF 100% power/108% flow. The operating limit for ATRIUM-9B is based on LRNB 100% power/108% flow. These results are shown in Table 1.

‡ The cycle-specific OLMCPR penalty of 0.04 is required to support operation with FFTR, FHOOS, coastdown or any combination thereof. ~~is only applied when~~ core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1 of Reference A.3. Other EOD/EOOS conditions listed in Table 2.4 of Reference A.2 require no OLMCPR penalty. Coastdown is defined as

per SPC 5/24/2001

Table 3 Quad Cities Unit 1 Cycle 17 Results of Plant Transient Analysis With TSSS Insertion Times

Event	Maximum Neutron Flux (% of rated)	Maximum Core Average Heat Flux (% of rated)	Maximum Vessel* /Dome Pressure (psig)
Load Rejection No Bypass			
100% power / 108% flow	636	130	1301 / 1267
Feedwater Flow Controller Failure			
100% power / 108% flow	619	133	1189 / 1154
MSIV Closure Without Position Scram (EOD/EOOS)			
100% power / 100% flow	329	130	1361 / 1336

---

* Lower plenum pressure.

Table 4 Flow-Dependent MCPR Results
ATRIUM-9B Offset Fuel

Total Core Flow (% of rated)	1.47 OLMCPR	1.51 OLMCPR	1.52 OLMCPR
108	1.470	1.510	1.520
100	1.563	1.605	1.616
90	1.695	1.741	1.753
80	1.843	1.893	1.906
70	1.999	2.054	2.068
60	2.164	2.225	2.240
50	2.339	2.404	2.421
40	2.533	2.603	2.621
30	2.819	2.898	2.918

Table 5 Quad Cities Unit 1 Cycle 17 Coastdown Operation MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power / Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel* / Dome Pressure (psig)	ΔCPR†	Change in ΔCPR From Limiting Rated Power Case†
LRNB	100 / 108	684	133	1310 / 1276	0.41 / 0.40	0.01 / 0.04

Table 6 Quad Cities Unit 1 Cycle 17 Combined FFTR/Coastdown MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power / Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel* / Dome Pressure (psig)	ΔCPR	Change in ΔCPR From Limiting Rated Power Case†
FWCF	100 / 108	578	139	1143 / 1109	0.44 / 0.40	0.04 / 0.04

---

* Lower plenum.

† Values for GE10/ATRIUM-9B offset fuel.

Table 7 Quad Cities Unit 1 Cycle 17 Bypass Valve(s) Out-of-Service MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power / Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel* / Dome Pressure (psig)	ΔCPR†	Change in ΔCPR From Limiting Rated Power Case†
FWCF						
1 BPVOOS	100 / 108	640	134	1202 / 1168	0.40 / 0.36	0.00 / 0.00
All BPVOOS	100 / 108	674	136	1308 / 1274	0.41 / 0.41	0.01 / 0.05

Table 8 Margin to Opening Unpiped Safety Valve Results

Transient	Exposure	Power / Flow	Maximum SRV Pressure (psia)	Margin (psi)
LRNB-USM SRVOOS	EOFP + 1500 MWd/MTU	100 / 100	1242.1	12.6

---

* Lower plenum.

† Values for GE10/ATRIUM-9B offset fuel.

Table 9 Quad Cities Unit 1 Cycle 17
Power Load Unbalance Out-of-Service Results

MCPR Results and Comparison to Corresponding Base Case LRNB ΔCPR Results

Transient	Power / Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel[*] / Dome Pressure (psig)	ΔCPR[†]	Δ(ΔCPR)[†, ‡]
PLUOOS EOFP	100 / 108	739	134	1304 / 1270	0.41 / 0.39	0.02 / 0.03
PLUOOS Coastdown	100 / 108	793	137	1314 / 1279	0.44 / 0.43	0.03 / 0.03

MFLCPR Multipliers

Transient	Power / Flow	OLMCPR[†]	MFLCPR Multiplier[†, §]
PLUOOS EOFP	100 / 108	1.51 / 1.47	0.986 / 0.980
PLUOOS Coastdown	100 / 108	1.55 / 1.51	0.981 / 0.980

---

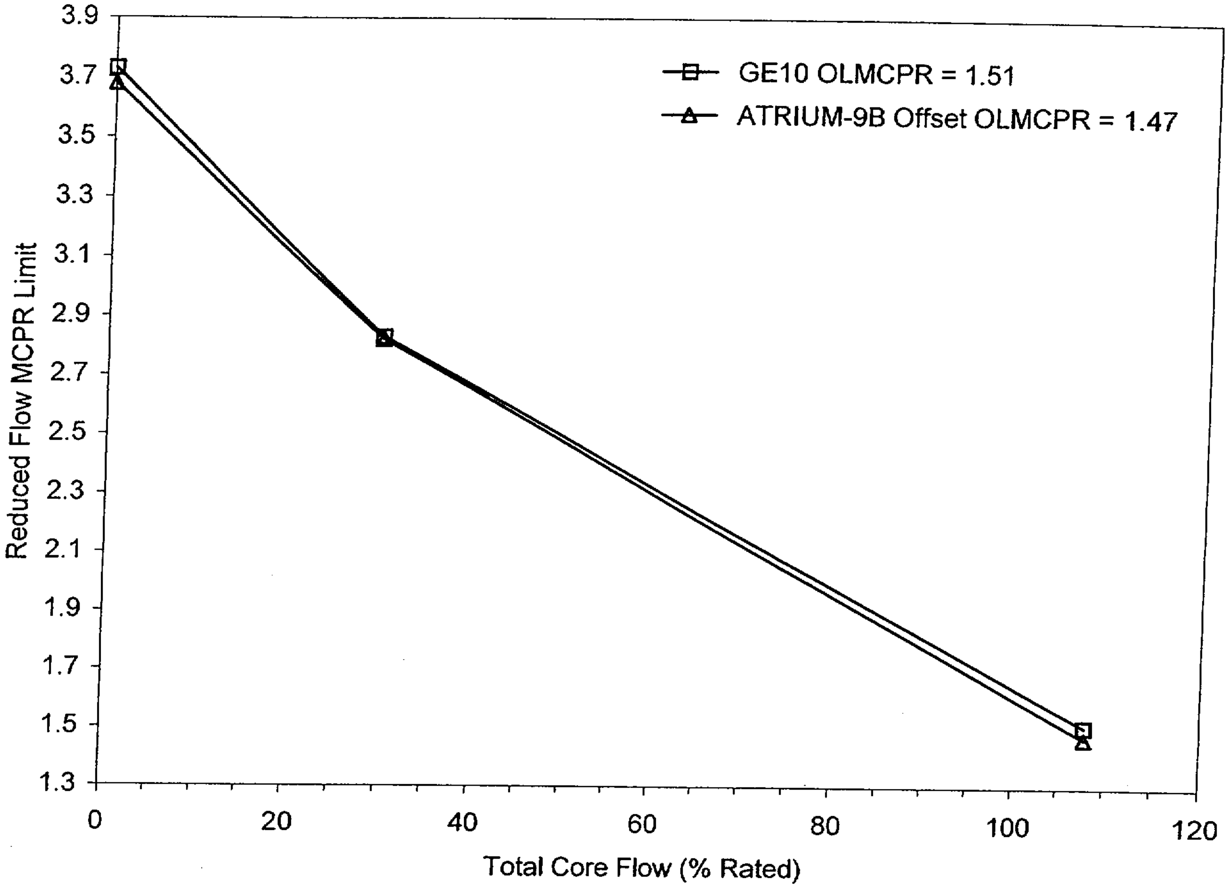
[*] Lower plenum.

[†] Values for GE10/ATRIUM-9B offset fuel.

[‡] Based on PLUOOS results and corresponding base case and EOD/EOOS LRNB results.

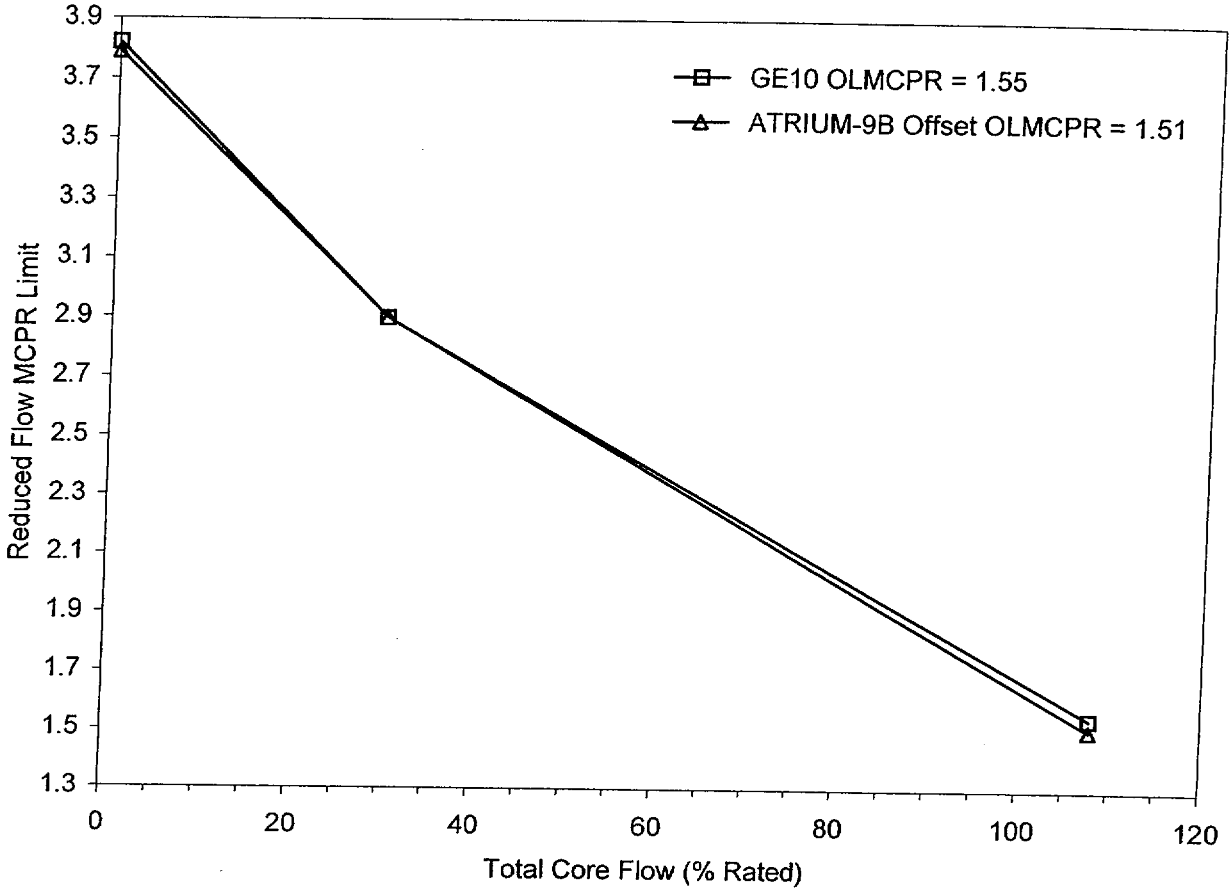
[§] The MFLCPR multipliers are calculated using the following equation (results were conservatively rounded down):

$$\text{MFLCPR Multiplier} = \frac{\text{OLMCPR}}{\text{OLMCPR} + \Delta(\Delta\text{CPR})}$$



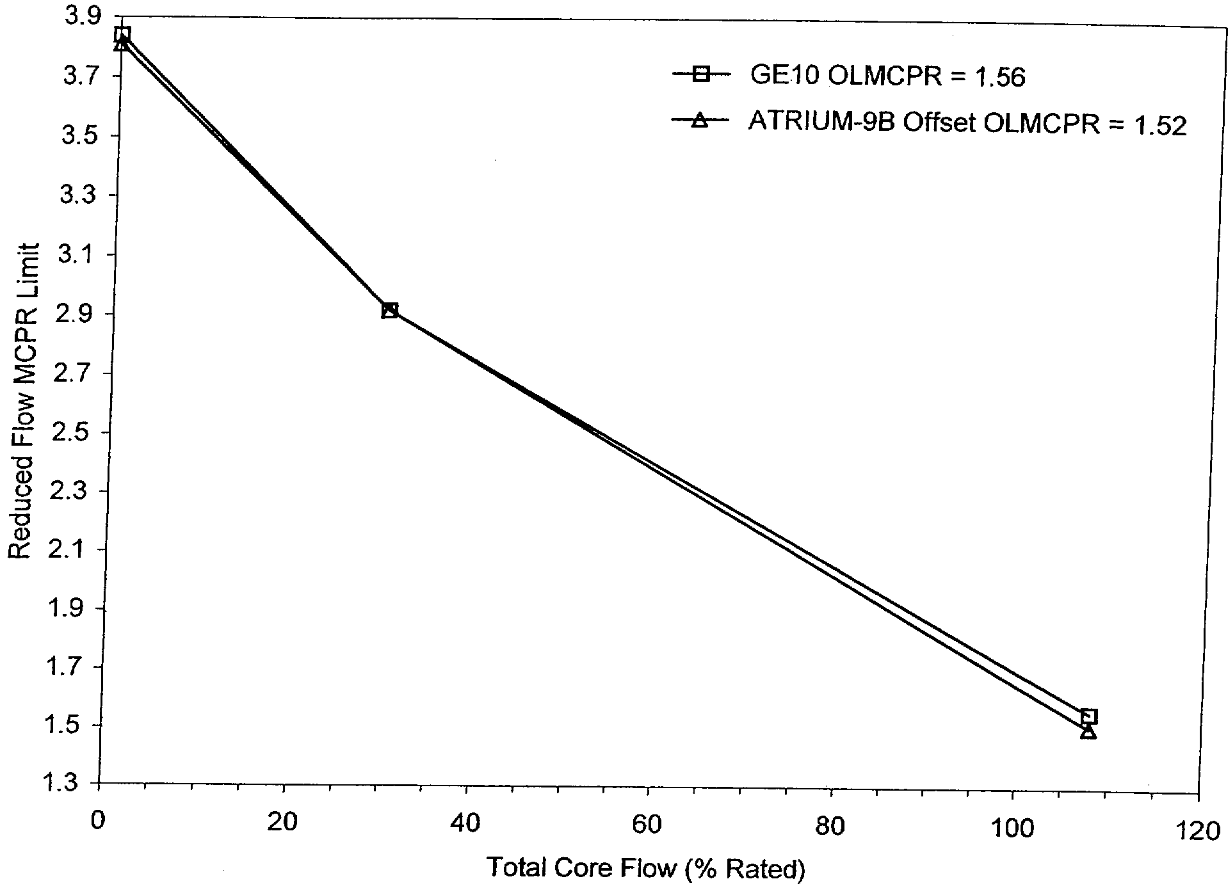
Total Core Flow (% of rated)	GE10 MCPR$_f$ Limit for OLMCPR = 1.51	ATRIUM-9B Offset MCPR$_f$ Limit for OLMCPR = 1.47
108	1.51	1.47
30	2.83	2.82
0	3.73	3.68

Figure 1 Reduced Flow MCPR Limit for Automatic Flow Control (Base Case OLMCPR)



Total Core Flow (% of Rated)	GE10 $MCPR_f$ Limit for OLMCPR = 1.55	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR = 1.51
108	1.55	1.51
30	2.90	2.90
0	3.82	3.79

Figure 2 Reduced Flow MCPR Limit for Automatic Flow Control (EOD/EOOS OLMCPR)



Total Core Flow (% of rated)	GE10 $MCPR_f$ Limit for OLMCPR = 1.56	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR = 1.52
108	1.56	1.52
30	2.92	2.92
0	3.85	3.81

Figure 3 Reduced Flow MCPR Limit for Automatic Flow Control (All BPVOOS OLMCPR)

Attachment 3

Quad Cities Unit 1 Cycle 17

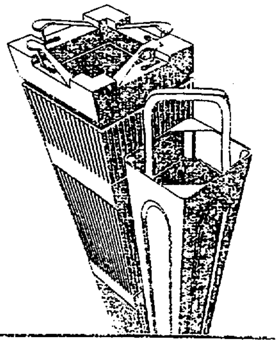
Reload Analysis Report

SIEMENS

EMF-2416
Revision 0

Quad Cities Unit 1 Cycle 17 Reload Analysis

August 2000



Siemens Power Corporation
Nuclear Division

Siemens Power Corporation

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DOCUMENT SYSTEM
DATE: 8/11/00

EMF-2416
Revision 0

Quad Cities Unit 1 Cycle 17
Reload Analysis

Prepared:
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T. M. Howe, Manager
Product Mechanical Engineering
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D. J. Denver, Manager
Commercial Operations
8-3-00
Date

sp

Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

Contents

Tables

Figures

Nomenclature

APRM	average power range monitor
BPVOOS	bypass valve(s) out of service
BOC	beginning of cycle
CPR	critical power ratio
CRWE	control rod withdrawal error
DR	decay ratio
ECCS	emergency core cooling system
EFPH	effective full power hour
EOC	end of cycle
EOD	extended operating domain
EOFP	end of full power
EOOS	equipment out of service
FFTR	final feedwater temperature reduction
FHOOS	feedwater heater out of service
FWCF	feedwater controller failure
ICF	increased core flow
LFWH	loss of feedwater heater
LHGR	linear heat generation rate
LOCA	loss of coolant accident
LPCI	low pressure core injection
LPRM	local power range monitor
LRNB	load rejection no bypass
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MSIV	main steam isolation valve
MWR	metal-water reaction
OLMCPR	operating limit minimum critical power ratio
RVOOS	relief valve out of service
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SPC	Siemens Power Corporation
SRVOOS	safety/relief valve out of service
TIP	traversing in-core probe
TIPOOS	traversing in-core probe out of service
TTNB	turbine trip no bypass
UFSAR	updated final safety analysis report
ΔCPR	change in critical power ratio

1.0 Introduction

This report provides the results of the analysis performed by Siemens Power Corporation (SPC) in support of the Cycle 17 reload for Quad Cities Unit 1. This report is intended to be used in conjunction with the SPC topical Report XN-NF-80-19(P)(A), Volume 4, Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, which describes the analyses performed in support of this reload, identifies the methodology used for those analyses, and provides a generic reference list. Section numbers in this report are the same as corresponding section numbers in XN-NF-80-19(P)(A), Volume 4, Revision 1. Methodology used in this report which supersedes XN-NF-80-19(P)(A), Volume 4, Revision 1 is referenced in Section 8.0.

For Quad Cities Unit 1 Cycle 17, Commonwealth Edison Company (ComEd) has responsibility for portions of the reload safety analysis. This document describes only the Cycle 17 analyses performed by SPC; ComEd analyses are described elsewhere. Hence, this document alone does not necessarily identify the limiting events or the appropriate operating limits for Cycle 17. The limiting events and operating limits must be determined in conjunction with results from ComEd analyses.

The Quad Cities Unit 1 Cycle 17 core consists of a total of 724 fuel assemblies, including 236 unirradiated QCA-2 ATRIUM™-9B* offset assemblies, 200 irradiated ATRIUM-9B offset assemblies and 288 irradiated GE10 assemblies. The reference core configuration is described in Section 4.2.1.

The design and safety analyses reported in this document were based on the design and operational assumptions in effect for Quad Cities Unit 1 during the previous operating cycle. The effects of channel bow are explicitly accounted for in the safety limit analysis. SPC has performed time step size sensitivity studies to assure that the numerical solution in the COTRANSA2 code converged.

Analyses and limits presented in this report support base case operation up to EOFP with bypass valve(s) out of service (BPVOOS) and operation with various extended operating domain (EOD) and equipment out-of-service (EOOS) conditions. The EOD/EOOS conditions addressed in this report are identified in Table 1.1.

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* ATRIUM is a trademark of Siemens.

Table 1.1 EOD and EOOS Operating Conditions*

Extended Operating Domain Conditions

- Increased Core Flow (ICF)
- Final Feedwater Temperature Reduction (FFTR)
- Coastdown
- Combined ICF/FFTR
- Combined ICF/Coastdown
- Combined FFTR/Coastdown
- Combined ICF/FFTR/Coastdown

Equipment Out-of-Service Conditions†

- Feedwater Heater(s) Out of Service (FHOOS)
- Single-loop Operation (SLO) - Recirculation Loop Out of Service
- Relief Valve Out of Service (RVOOS)
- Safety/Relief Valve Safety Function Out of Service (SRVOOS) for ASME Events
- Up to 40% TIP Strings Out of Service (TIPOOS)‡

---

* Base case operation up to EOFP with bypass valve(s) out of service is analyzed separately and is not considered for EOD/EOOS operation.

† EOOS conditions, with the exception of FHOOS, are supported for both EOD conditions and standard operating domain conditions.

‡ 40% TIPOOS with 100% TIP strings available at startup, 50% of the LPRMs out of service (LPRM substitution model on or off), and 2500 EFPH LPRM calibration interval.

2.0 Fuel Mechanical Design Analysis

Applicable SPC Fuel Design Reports References 9.7 and 9.8

To assure that the power history for the fuel to be irradiated during Cycle 17 of Quad Cities Unit 1 is bounded by the assumed power history in the fuel mechanical design analysis, LHGR operating limits have been specified. In addition, LHGR limits for Anticipated Operational Occurrences have been specified in the references. Steady-state LHGR limits are provided in Section 7.2.3. ATRIUM-9B steady-state and transient LHGR limits are presented in Figure 7.1.

From Reference 9.7, the maximum discharge exposures for ATRIUM-9B offset fuel are:

- 48 GWd/MTU assembly exposure
- 55 GWd/MTU rod exposure

The corresponding pellet exposure in the mechanical analysis is 66 GWd/MTU.

3.0 Thermal-Hydraulic Design Analysis

3.2 *Hydraulic Characterization*

3.2.1 Hydraulic Compatibility

Component hydraulic resistances for the constituent fuel types in the Quad Cities Unit 1 Cycle 17 core have been determined in single-phase flow tests of full-scale assemblies. The hydraulic demand curves for SPC ATRIUM-9B offset and GE10 fuel in the Quad Cities Unit 1 core are provided in Reference 9.7 (Figures 4.2 and 4.3 in the reference).

3.2.3 Fuel Centerline Temperature

ATRIUM-9B Offset	Reference 9.7, Figure 3.3

3.2.5 Bypass Flow

Calculated Bypass Flow Fraction at 100% power/100% flow at EOC*	12.7%	Reference 9.3

3.3 *MCPR Fuel Cladding Integrity Safety Limit (SLMCPR)*

Two-Loop Operation	1.11†	Reference 9.3
Single-Loop Operation	1.12†	

3.3.1 Coolant Thermodynamic Condition

Thermal Power (at SLMCPR)	3796 MWt
Feedwater Flow Rate (at SLMCPR)	14.9 Mlb/hr
Core Pressure	1030 psia
Feedwater Temperature	352.7°F‡

---

[*] Includes water rod/internal water channel flow.

[†] Includes the effects of channel bow, up to 40% of the TIP strings out of service (but 100% TIP strings available at startup), a 2500 EFPH calibration interval, and up to 50% of the LPRMs out of service (LPRM substitution model on or off). For operation with uncalibrated LPRMs at startup, these limits are supported for cycle exposures up to 500 MWd/MTU.

[‡] As determined by SPC heat balance calculations.

3.3.2 Design Basis Radial Power Distribution

Figure 3.1 shows the limiting radial power distribution used in the MCPR Fuel Cladding Integrity Safety Limit analysis.

3.3.3 Design Basis Local Power Distribution

Figures 3.2 and 3.3 show the conservative local power distributions used in the MCPR Fuel Cladding Integrity Safety Limit analysis.

SPCA9-383B-11GZH-ADV	Figure 3.2
SPCA9-382B-12GZL-ADV	Figure 3.3

3.4 *Licensing Power and Exposure Shape*

The licensing axial power profile used by SPC for the plant transient analyses bounds the projected end of full power (EOFP) axial power profile. The conservative licensing axial power profile as well as the corresponding axial exposure ratio are given below. Future projected Cycle 17 power profiles are considered to be in compliance when the EOFP normalized power generated in the bottom of the core is greater than the licensing axial power profile at the given state conditions.

State Conditions for Power Shape Evaluation

Power, MWt	2511.0
Core Pressure, psia	1030.0
Inlet Subcooling, Btu/lbm	23.05
Flow, Mlb/hr	98.0

Licensing Axial Power Profile

Node	Power
Top 24	0.247
23	0.499
22	1.156
21	1.410
20	1.550
19	1.605
18	1.619
17	1.604
16	1.568
15	1.514
14	1.454
13	1.372
12	1.294
11	1.187
10	1.071
9	0.950
8	0.830
7	0.710
6	0.600
5	0.509
4	0.442
3	0.392
2	0.319
Bottom 1	0.098

Licensing Axial Exposure Ratio (EOFP)
Average Bottom 8 ft/12 ft = 1.1124

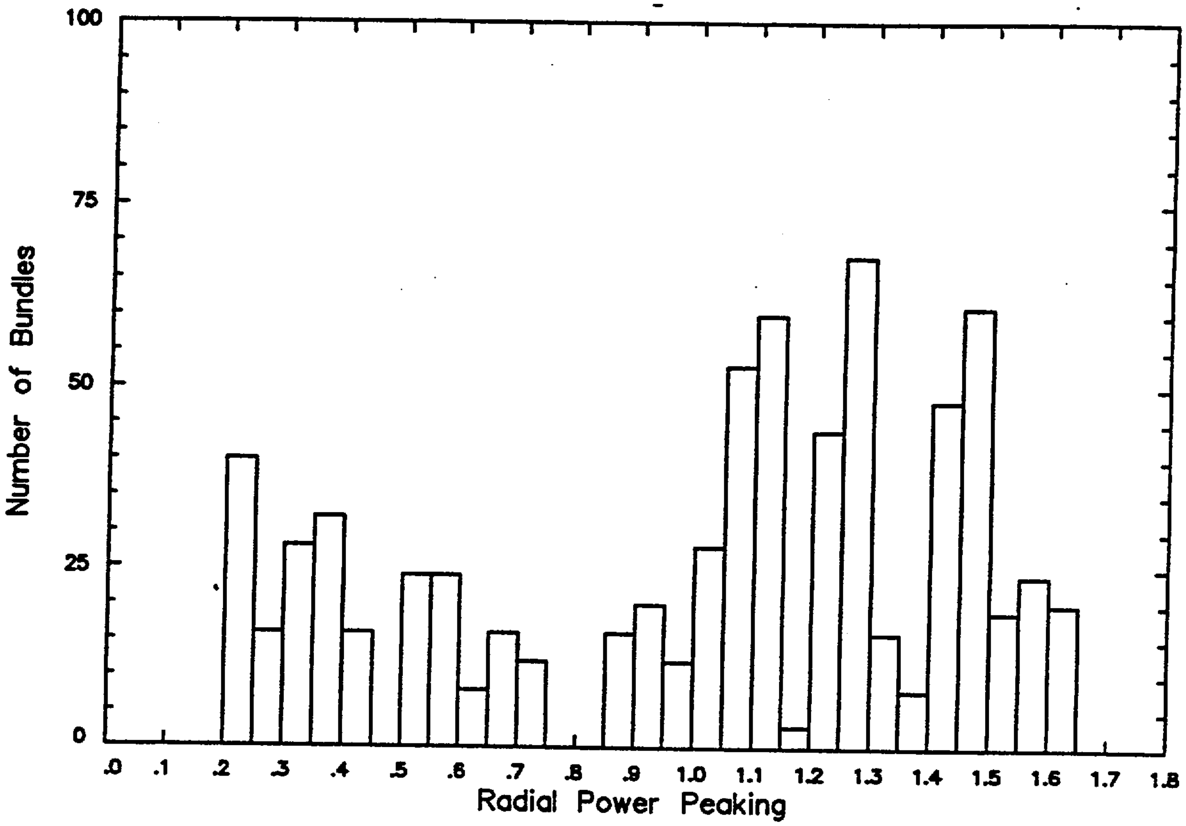

Figure 3.1 Design Basis Radial Power Distribution for SLMCPR Determination

Control Rod Corner

Control Rod Corner

1.009	1.026	1.044	1.051	1.108	1.096	1.095	0.992	0.977
1.026	0.987	1.060	0.978	1.075	1.056	0.930	0.994	0.970
1.044	1.060	0.978	1.126	1.117	1.076	1.020	0.876	1.049
1.051	0.978	1.126		Internal		1.034	0.962	1.005
1.108	1.075	1.117		Water		1.069	0.828	0.982
1.096	1.056	1.076		Channel		1.026	0.918	0.963
1.095	0.930	1.020	1.034	1.069	1.026	0.821	0.909	0.960
0.992	0.994	0.876	0.962	0.828	0.918	0.909	0.810	0.915
0.977	0.970	1.049	1.005	0.982	0.963	0.960	0.915	0.822

Figure 3.2 Quad Cities Unit 1 Cycle 17 Safety Limit Local Peaking Factors With Channel Bow at Assembly Exposure of 25,000 MWd/MTU (SPCA9-383B-11GZH-ADV)

Control Rod Corner

Control Rod Corner

1.011	1.029	1.047	1.047	1.105	1.095	1.096	0.993	0.979
1.029	0.994	1.007	1.026	1.003	1.055	0.931	0.996	0.972
1.047	1.007	1.100	1.110	1.113	1.075	1.021	0.877	1.051
1.047	1.026	1.110		Internal		1.035	0.964	1.006
1.105	1.003	1.113		Water		1.070	0.829	0.983
1.095	1.055	1.075		Channel		1.027	0.920	0.964
1.096	0.931	1.021	1.035	1.070	1.027	0.822	0.911	0.961
0.993	0.996	0.877	0.964	0.829	0.920	0.911	0.811	0.917
0.979	0.972	1.051	1.006	0.983	0.964	0.961	0.917	0.823

Figure 3.3 Quad Cities Unit 1 Cycle 17 Safety Limit Local Peaking Factors With Channel Bow at Assembly Exposure of 25,000 MWd/MTU (SPCA9-382B-12GZL-ADV)

4.0 Nuclear Design Analysis

4.1 *Fuel Bundle Nuclear Design Analysis*

Assembly Average Enrichment

ATRIUM-9B offset	(QCA-2 Type H)	3.83 wt%
	(QCA-2 Type L)	3.82 wt%

Radial Enrichment Distribution

SPCA9-4.15L-11G6.0	Reference 9.7
SPCA9-4.15L-11G8.0	Reference 9.7
SPCA9-4.32L-10G8.0	Reference 9.7
SPCA9-4.14L-10G6.0	Reference 9.7
SPCA9-4.14L-11G7.0	Reference 9.7
SPCA9-4.30L-11G7.0	Reference 9.7
SPCA9-4.30L-12G7.0	Reference 9.7

Axial Enrichment Distribution	Figure 4.1
Burnable Absorber Distribution	Figure 4.1
Non-Fueled Rods	Reference 9.7
Neutronics Design Parameters	Table 4.1

Fuel Storage*

Quad Cities New Fuel Storage Vault	Reference 9.10

The QCA-2 Reload Batch fuel designs meet the fuel design limitations defined in Table 2.1 of Reference 9.10 and therefore can be safely stored in the vault.

Quad Cities Spent Fuel Storage Vault	Reference 9.9

The QCA-2 Reload Batch fuel designs may be stored in the storage pool provided the array k-eff is ≤ 0.95 as determined by the procedure defined in Section 6.5 of Reference 9.9.

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* The ATRIUM-9B offset fuel is bounded by the referenced analysis.

4.2 *Core Nuclear Design Analysis*

4.2.1 Core Configuration

Figure 4.2

Core Exposure at EOC16, MWd/MTU (nominal value)	29,201
Core Exposure at BOC17, MWd/MTU (from nominal EOC16)	17,113
Core Exposure at EOC17, MWd/MTU (licensing basis)	31,548

NOTE: Analyses in this report are applicable to a core exposure of 31,548 MWd/MTU. EOD/EOOS cycle extension analyses (References 9.3 and 9.6) are applicable for Cycle 17 provided full power capability is lost prior to reaching a core exposure of 31,548 MWd/MTU.

< Cycle 17 short window exposure to be furnished by ComEd. >

4.2.2 Core Reactivity Characteristics

< This data is to be furnished by ComEd. >

4.2.4 Core Hydrodynamic Stability

Quad Cities Unit 1 utilizes the BWROG Interim Corrective Actions (ICAs) to address thermal hydraulic instability issues. This is in response to Generic Letter 94-02. When the long-term solution OPRM is fully implemented, the ICAs will remain as a backup to the OPRM system.

In order to support the ICAs and remain cognizant of the relative stability of one cycle compared with previous cycles, decay ratios are calculated at various points on the power to flow map and at various points in the cycle. This satisfies the following functions.

- Provides trending information to qualitatively compare the stability from cycle to cycle.
- Provides decay ratio sensitivities to rod line and flow changes near the ICA regions.
- ComEd reviews this information to determine if any administrative conservatisms are appropriate beyond the existing requirements.

The results of the evaluation of decay ratio for several points along the current exclusion region boundary of the power/flow operation map are shown below. This analysis was performed using the design basis step-through control rod pattern projection, hence, it explicitly models the effects of Cycle 17 exposure. The calculated decay ratios are provided to assist ComEd in performing the three functions described above.

	% Power/% Flow State Points	Decay Ratio (ΔDR)[*]			
		Global		Regional	
1.	64/38.8[†]	>1.00	---	0.90	(.15)
2.	68.5/45[‡]	0.78	(.07)	0.73	(.17)
3.	58.5/45[§]	0.57	(.02)	0.48	(.13)
4.	23/19.4[**]	0.46	(.16)	0.37	(.17)
5.	37/38.8[††]	0.29	(.01)	0.23	(.06)

For reactor operation under conditions of coastdown, feedwater heaters out of service, and single-loop, it is possible that higher decay ratios could be achieved than are shown for normal operation. Operation under these conditions will be acceptable in Cycle 17 as long as operating procedures and precautions defined in the ICAs are followed.

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[*] DR_{CY17} - DR_{CY16} values are in parenthesis.

[†] APRM rod block line - two-pump minimum flow.

[‡] APRM rod block line - 45% flow.

[§] 100% rod line - 45% flow.

[**] 70% rod line - natural circulation flow.

[††] 70% rod line - two-pump minimum flow.

Table 4.1 Neutronic Design Values

Number of Fuel Assemblies	724
Rated Thermal Power, MWt	2511
Rated Core Flow, Mlbm/hr	98.0
Core Inlet Subcooling, Btu/lbm	21.6*
Moderator Temperature, °F	546*
Channel Thickness (Corner), inch	0.100†
Channel Internal Face-to-face Dimension, inch	5.278
Fuel Assembly Pitch, inch	6.0
Wide Water Gap Thickness, inch	0.630†
Narrow Water Gap Thickness, inch	0.414†
<u>Control Rod Data</u>‡	
Absorber Material	B_4C
Total Blade Span, inch	9.810
Total Blade Support Span, inch	1.580
Blade Thickness, inch	0.312
Absorber Rods Per Blade	84
Absorber Rod OD, inch	0.188
Absorber Rod ID, inch	0.138
Absorber Density, % of theoretical	70

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* Based on actual operating experience.

† Value corresponds to the ATRIUM-9B offset fuel with advanced channel gap measured at the top and bottom of the bundle; i.e., from the 100-mil-thick channel wall.

‡ The control rod data represents original equipment control blades at Quad Cities which were modeled in the licensing analyses. Quad Cities UFSAR Section 4.6.2.1 indicates that reactivity characteristics of replacement control blades closely match original equipment blades.

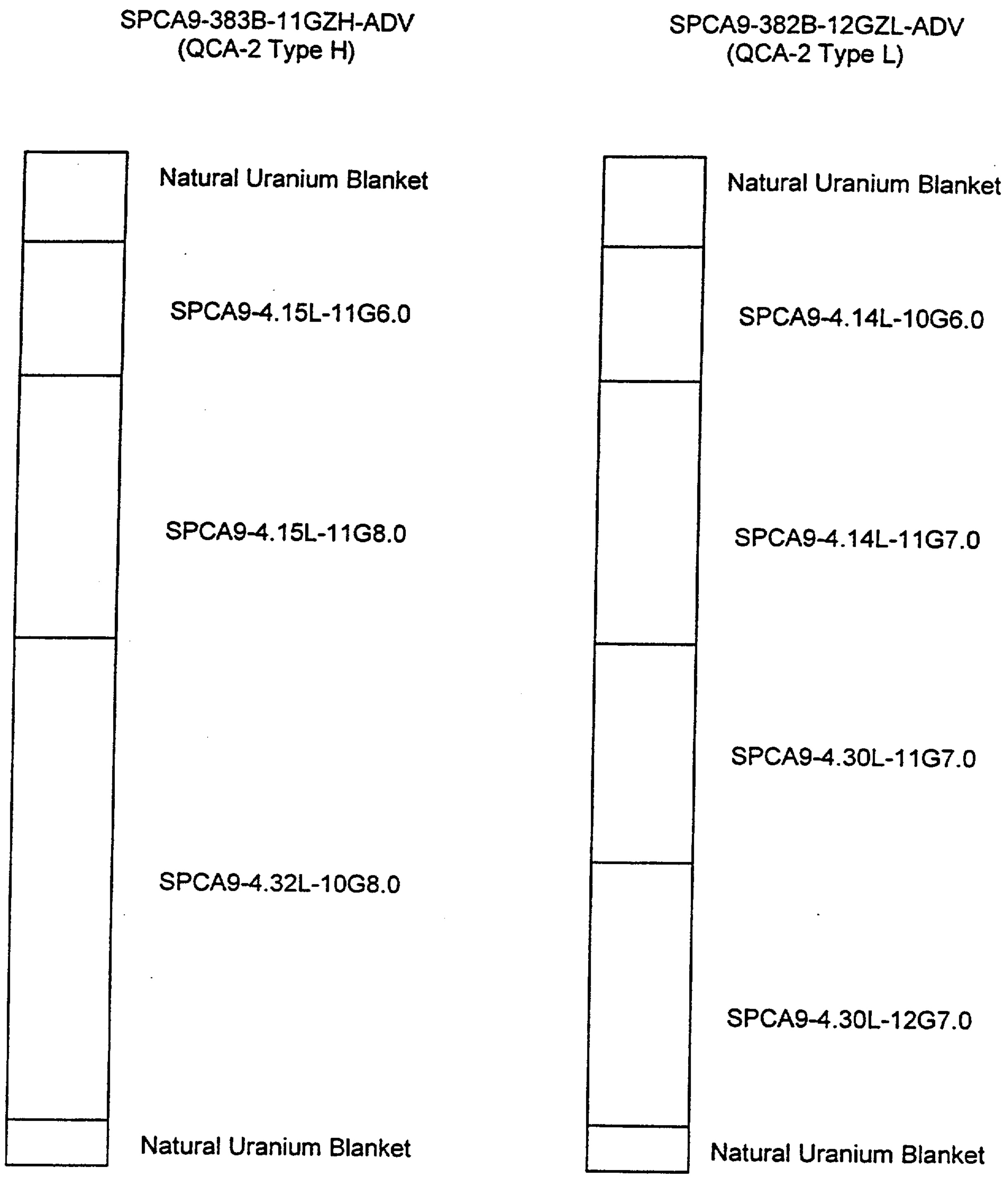

Figure 4.1 Quad Cities Unit 1 Reload Batch QCA-2 Axial Fuel Assembly Design

3	18	5	3	5	5	19	5	4	5	19	5	3	1	3
18	5	19	5	18	19	4	19	18	5	5	7	4	4	3
5	19	4	18	5	18	19	7	19	18	18	19	3	4	3
3	5	18	5	5	5	19	5	3	7	19	7	1	3	3
5	18	5	5	4	19	4	4	5	19	18	5	4	2	1
5	19	18	5	19	5	19	19	18	5	19	1	1	4	
19	4	19	19	4	19	18	7	19	19	7	3	3		
5	19	7	5	4	19	7	5	5	19	18	4	3		
4	18	19	3	5	18	19	5	3	18	3	2	3		
5	5	18	7	19	5	19	19	18	3	3	3			
19	5	18	19	18	19	7	18	3	3	3				
5	7	19	7	5	1	3	4	2	3					
3	4	3	1	4	1	3	3	3						
1	4	4	3	2	4									
3	3	3	3	1										

Fuel Type	Number of Assemblies	Bundle Description	Cycle Loaded
1	40	GE10-P8HXB311-8GZ-100M-145-CECO	14
2	16	GE10-P8HXB312-7GZ-100M-145-CECO	14
3	144	GE10-P8HXB332-8G5.0-100M-145-CECO	15
4	88	GE10-P8HXB333-4G5.0/6G4.0-100M-145-CECO	15
5	152	SPCA9-348B-11G6.5-ADV	16
7	48	SPCA9-360B-11G6.5-ADV	16
18	92	SPCA9-383B-11GZH-ADV	17
19	144	SPCA9-382B-12GZL-ADV	17

**Figure 4.2 Quad Cities Unit 1 Cycle 17 Reference Loading Map
(Quarter-Core Symmetric Loading)**

5.0 Anticipated Operational Occurrences

Applicable Generic Transient Analysis Report — References 9.6 and 9.12

5.1 *Analysis of Plant Transients at Rated Conditions*

References 9.3, 9.6 and 9.12

Limiting Transients:
- Load Rejection No Bypass (LRNB)
- Feedwater Controller Failure (FWCF)
- Loss of Feedwater Heating (LFWH)*

Event	Power (%)	Flow (%)	Maximum Heat Flux (%)	Peak Neutron Flux (%)	Maximum Pressure (psig)	ΔCPR†	Model
LRNB‡,§	100	108	130	626	1300	0.39 / 0.35	COTRANSA2
FWCF‡,§	100	108	133	610	1189	0.40 / 0.35	COTRANSA2

5.2 *Analysis for Reduced Flow Operation*

Reference 9.3

Limiting Transient: Recirculation Flow Increase Transient (Pump Run-Up Event)

5.3 *Analysis for Reduced Power Operation*

References 9.3, 9.6 and 9.12

Limiting Transient: Feedwater Controller Failure (FWCF)

---

* This data to be furnished by ComEd.

† ΔCPR results for GE10/ATRIUM-9B offset fuel.

‡ Based on Technical Specification limiting scram performance parameters.

§ The cycle specific OLMCPR penalty of 0.04 required to support operation with FFTR, FHOOS, coastdown or any combination thereof, ~~is only applied when~~ core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1. Other EOD/EOOS conditions listed in Table 1.1 require no OLMCPR penalty.

Coastdown is defined as
per SPC
9/20/00

5.4 *ASME Overpressurization Analysis*[*]

Limiting Event	MSIV Closure
Worst Single Failure	Valve Position Scram
Maximum Pressure (Lower Plenum)	1361 psig
Maximum Steam Dome Pressure[†]	1335 psig

5.5 *Control Rod Withdrawal Error*

< This analysis is the responsibility of ComEd. >

5.6 *Fuel Loading Error*

< This analysis is the responsibility of ComEd. >

5.7 *Determination of Thermal Margins*

Summary of Thermal Margin Requirements

Event	Power (%)	Flow (%)	ΔCPR[‡]	MCPR Limit[‡, §]
LRNB	100	108	0.39 / 0.35	1.50 / 1.46
FWCF	100	108	0.40 / 0.35	1.51 / 1.46

MCPR Operating Limit[‡,§,**]

Base Case Operation	1.51 / 1.46
Base Case Operation with 1 BPVOOS	1.52 / 1.47
Base Case Operation with All BPVOOS	1.56 / 1.51
EOD/EOOS Operation[††]	1.55 / 1.50

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[*] Analysis results are provided for the limiting maximum pressurization EOD/EOOS condition. Therefore, no EOD/EOOS pressure penalty is required.

[†] Analysis of the limiting maximum pressurization EOD/EOOS condition/state-point produces both the maximum vessel pressure and the maximum steam dome pressure. Based on these results, all Technical Specification vessel and steam dome pressure limits are protected.

[‡] Values for GE10/ATRIUM-9B offset fuel.

[§] Based on plant Technical Specification two-loop MCPR safety limit of 1.11 and Technical Specification limiting scram performance parameters. For operation in single-loop, the Technical Specification single-loop MCPR safety limit of 1.12 increases the MCPR operating limit by 0.01.

[**] These limits may need to be increased if ComEd analyses are more limiting.

[††] The cycle specific OLMCPR penalty of 0.04 is required to support operation with FFTR, FHOOS, coastdown or any combination thereof. ~~is only applied when~~ core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1. Other EOD/EOOS conditions listed in Table 1.1 require no OLMCPR penalty.

Coastdown is defined as
per SPC
9/20/00

MCPR Operating Limits at Off-Rated Conditions*

Reduced Flow MCPR Limits:

Manual Flow Control	Figure 5.1
Automatic Flow Control†	Figures 5.2 – 5.4

---

* Based on plant Technical Specification two-loop MCPR safety limit of 1.11 and Technical Specification limiting scram performance parameters. For operation in single-loop, the Technical Specification single-loop MCPR safety limit of 1.12 increases the MCPR operating limit by 0.01.

† Automatic flow control analyses were performed to support OLMCPRs for base case operation, base case operation with all BPVOOS and for EOD/EOOS operation (refer to Table 1.1). AFC MCPR_f limits for the 1 BPVOOS OLMCPRs can be determined from the appropriate base case and EOD/EOOS MCPR_f limits. These limits may need to be increased if ComEd analyses are more limiting.

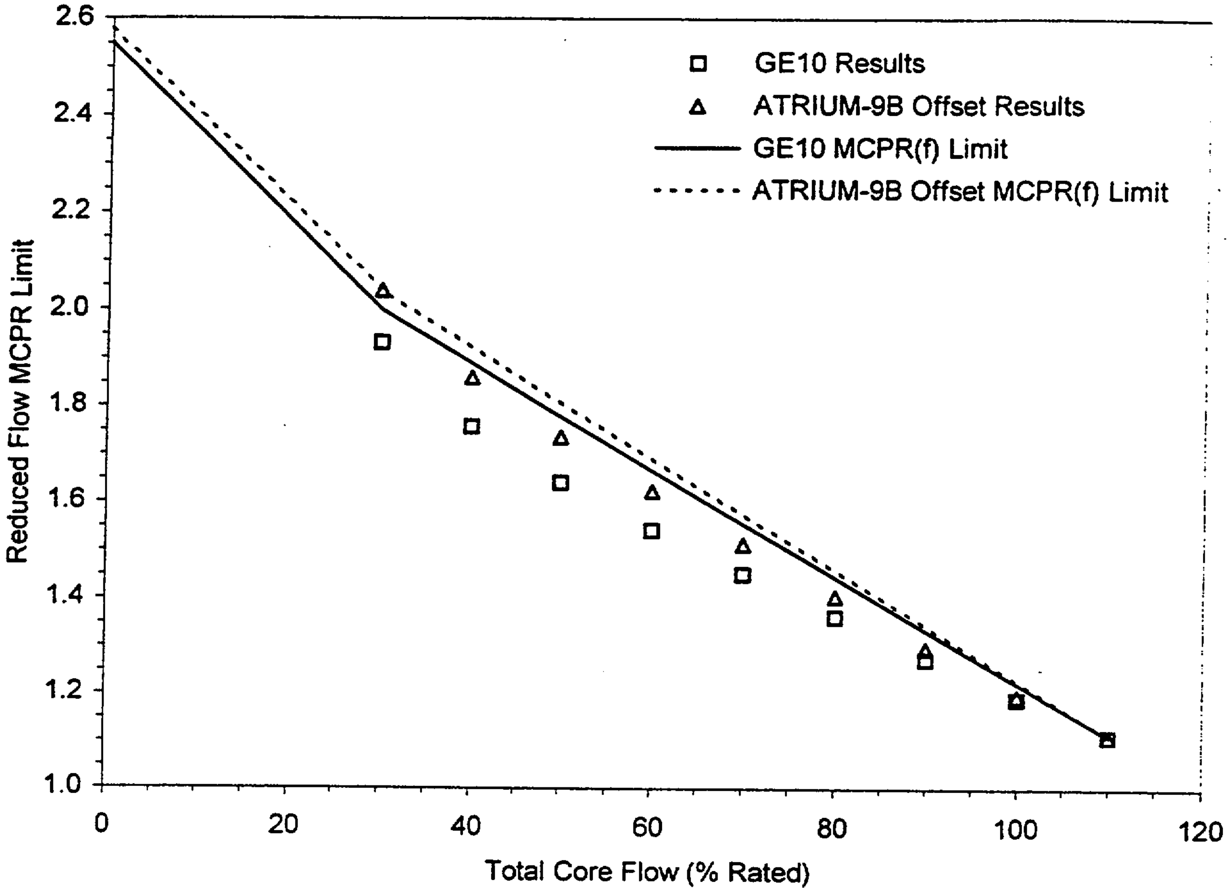


Figure 5.1 Reduced Flow MCPR Limit for Manual Flow Control (SLMCPR = 1.11)

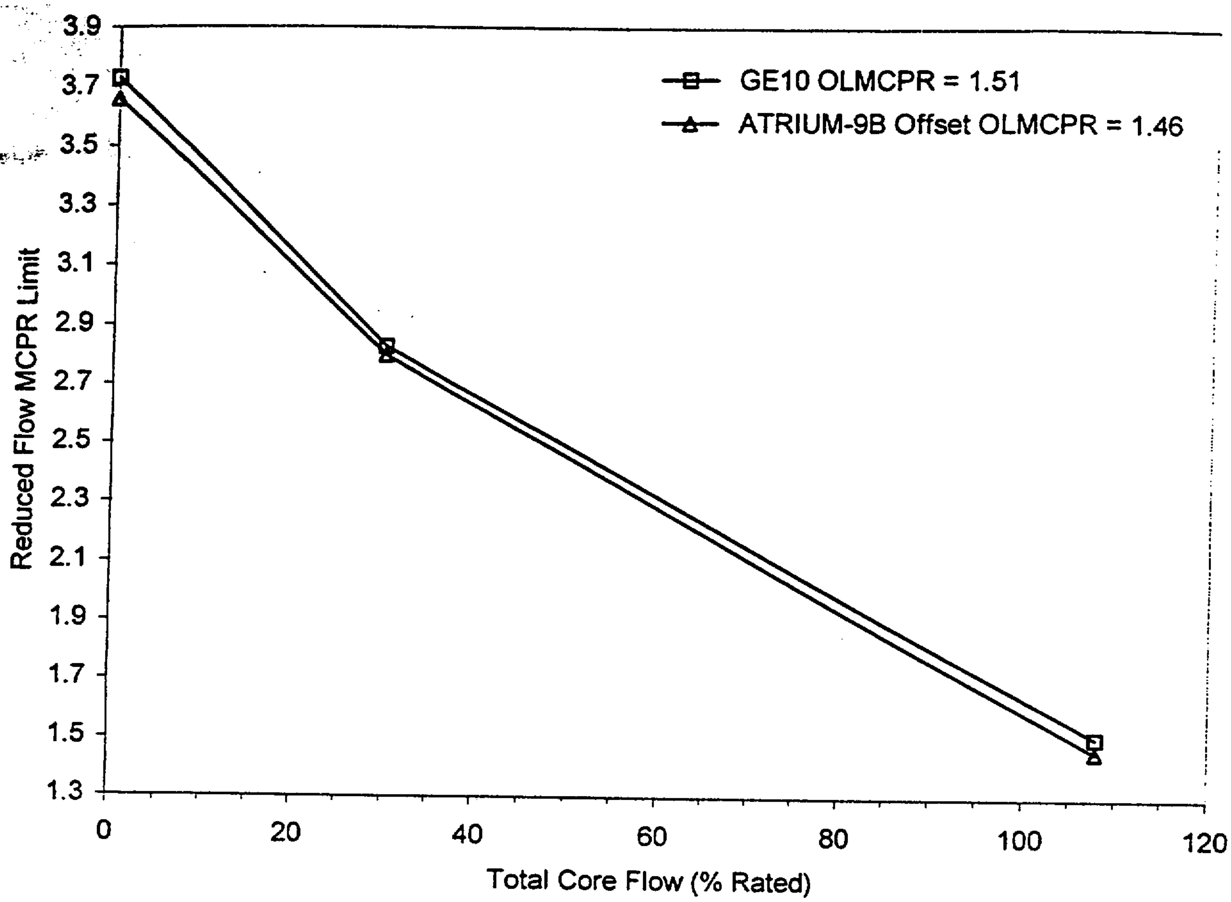


Figure 5.2 Reduced Flow MCPR Limit for Automatic Flow Control (Base OLMCPR)

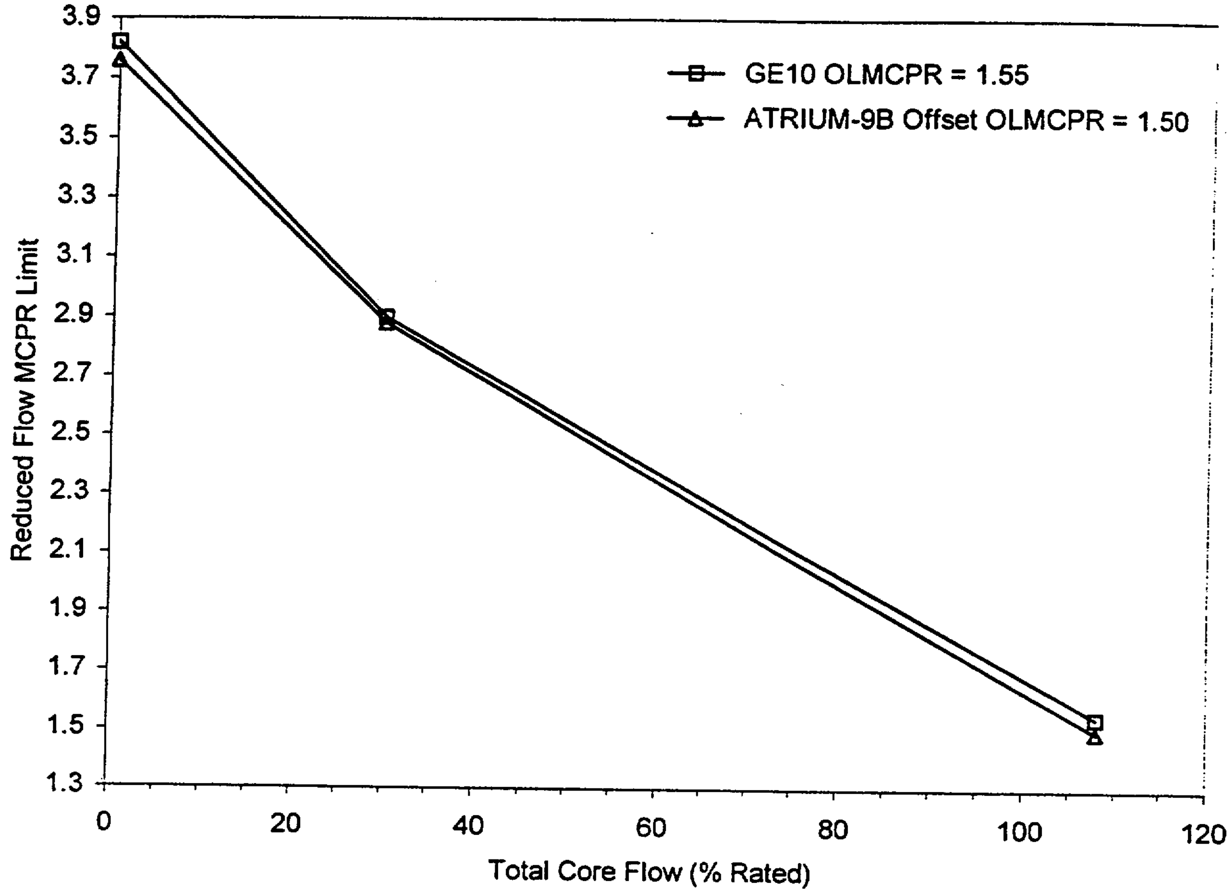


Figure 5.3 Reduced Flow MCPR Limit for Automatic Flow Control (EOD/EOOS OLMCPR)

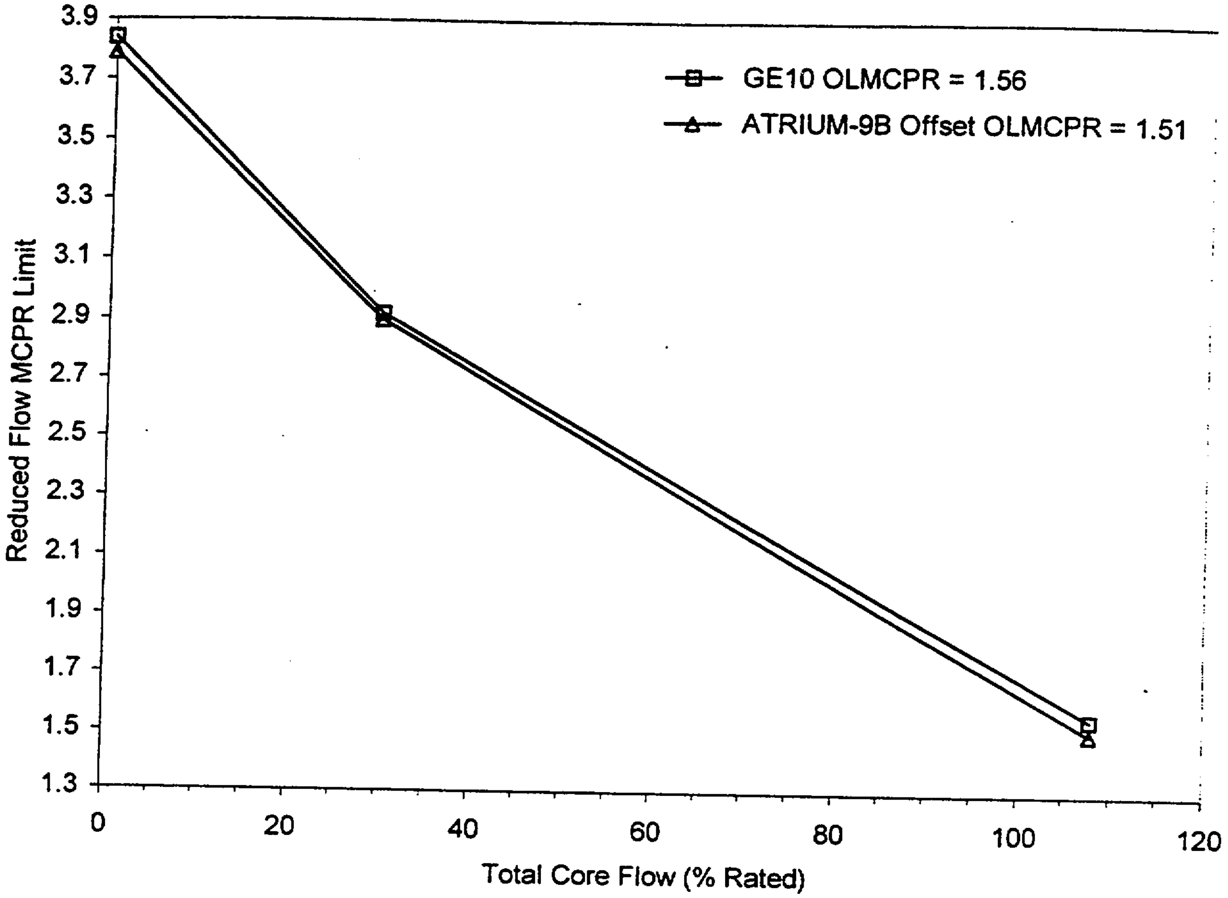


Figure 5.4 Reduced Flow MCPR Limit for Automatic Flow Control (All BPVOOS OLMCPR)

6.0 Postulated Accidents

6.1 *Loss-of-Coolant Accident*

6.1.1 Break Location Spectrum

References 9.4 and 9.5

6.1.2 Break Size Spectrum

Reference 9.5

6.1.3 MAPLHGR Analyses

Reference 9.5

The MAPLHGR limits of Reference 9.5 are valid for the Quad Cities ATRIUM-9B offset (QCA-2) fuel for Cycle 17 operation. Additional analyses were performed to extend the ATRIUM-9B offset limits to an exposure of 61.1 GWd/MTU. The exposure extension of MAPLHGR limits is consistent with existing LHGR limits.

Limiting Break: Double-Ended Guillotine Pipe Break
Recirculation Pump Suction Line
1.0 Discharge Coefficient
LPCI Injection Valve Failure

Peak cladding temperature (PCT), peak local metal-water reaction (MWR) and total core-wide MWR results for ATRIUM-9B offset fuel at Quad Cities are 1952°F, 2.23% and < 0.12%, respectively (Reference 9.5). The limiting PCT occurred at a planar exposure of 15 GWd/MTU and the peak local MWR occurred at a planar exposure of 20 GWd/MTU.

The PCT, peak local MWR and total core-wide MWR results for the Cycle 17 ATRIUM-9B offset reload fuel are 1918°F, 2.31% and < 0.12%, respectively. The limiting PCT and the peak local MWR occurred at a planar exposure of 20 GWd/MTU. Cycle 17 PCT results are bound by results provided in Reference 9.5.

6.2 *Control Rod Drop Accident*

< This analysis is the responsibility of ComEd. >

6.3 *Spent Fuel Cask Drop Accident*

The Quad Cities UFSAR analysis of record for the Spent Fuel Cask Drop Accident is not fuel-type dependent; thus, the results reported in UFSAR Section 15.7.3 are applicable for the SPC reload fuel.

7.0 Technical Specifications

7.1 *Limiting Safety System Settings*

7.1.1 MCPR Fuel Cladding Integrity Safety Limit

MCPR Safety Limit (all fuel) - Two-Loop Operation	1.11*
MCPR Safety Limit (all fuel) - Single-Loop Operation	1.12*

7.1.2 Steam Dome Pressure Safety Limit

Pressure Safety Limit	1345 psig

7.2 *Limiting Conditions for Operation*

7.2.1 Average Planar Linear Heat Generation Rate

Planar Average Exposure (GWd/MTU)	ATRIUM-9B Offset MAPLHGR (kW/ft)	GE10 MAPLHGR (kW/ft)
0	13.5	< To be furnished by ComEd. >
20	13.5	
60	8.7	
61.1	8.6	

SPC performed LOCA analyses from single-loop conditions and determined an appropriate SLO MAPLHGR multiplier of 0.9 for ATRIUM-9B offset fuel. The ECCS analysis results are presented in Reference 9.5. All calculations were performed with the NRC-approved EXEM/BWR ECCS Evaluation Model according to Appendix K of 10CFR50.

7.2.2 Minimum Critical Power Ratio

Rated Conditions MCPR Limit Based on Technical Specification Scram Times	TBD†

---

* Includes the effects of channel bow with up to 40% of the TIP strings out of service (but 100% TIP strings available at startup), a 2500 EFPH calibration interval, and up to 50% of the LPRMs out of service (LPRM substitution model on or off).

† Based on results from Section 5.7 and results from ComEd's scope of responsibility. The MCPR operating limit is based on a Technical Specification two-loop MCPR safety limit of 1.11 and the limiting ΔCPR for Cycle 17.

Off-Rated Conditions MCPR Limits:	
Manual Flow Control	Figure 5.1
Automatic Flow Control	Figures 5.2 – 5.4

7.2.3 Linear Heat Generation Rate

Figure 2.1 of Reference 9.7

Steady-State LHGR Limits

GE10		ATRIUM-9B Offset Fuel	
Planar Average Exposure (GWd/MTU)	LHGR (kW/ft)	Planar Average Exposure (GWd/MTU)	LHGR (kW/ft)
< To be furnished by ComEd. >		0.0	14.4
		15.0	14.4
		61.1	8.32

The steady-state and transient linear heat generation rate curves are provided in Figure 2.1 of Reference 9.7 for ATRIUM-9B offset fuel. This figure is presented in this report as Figure 7.1 for convenience.

Composite power history curves for the FWCF and the LRNB analyses are provided in Reference 9.11. ComEd must evaluate the information provided in Reference 9.11 to ensure that the mechanical design criteria (1% plastic strain) is satisfied for the coresident GE10 fuel.

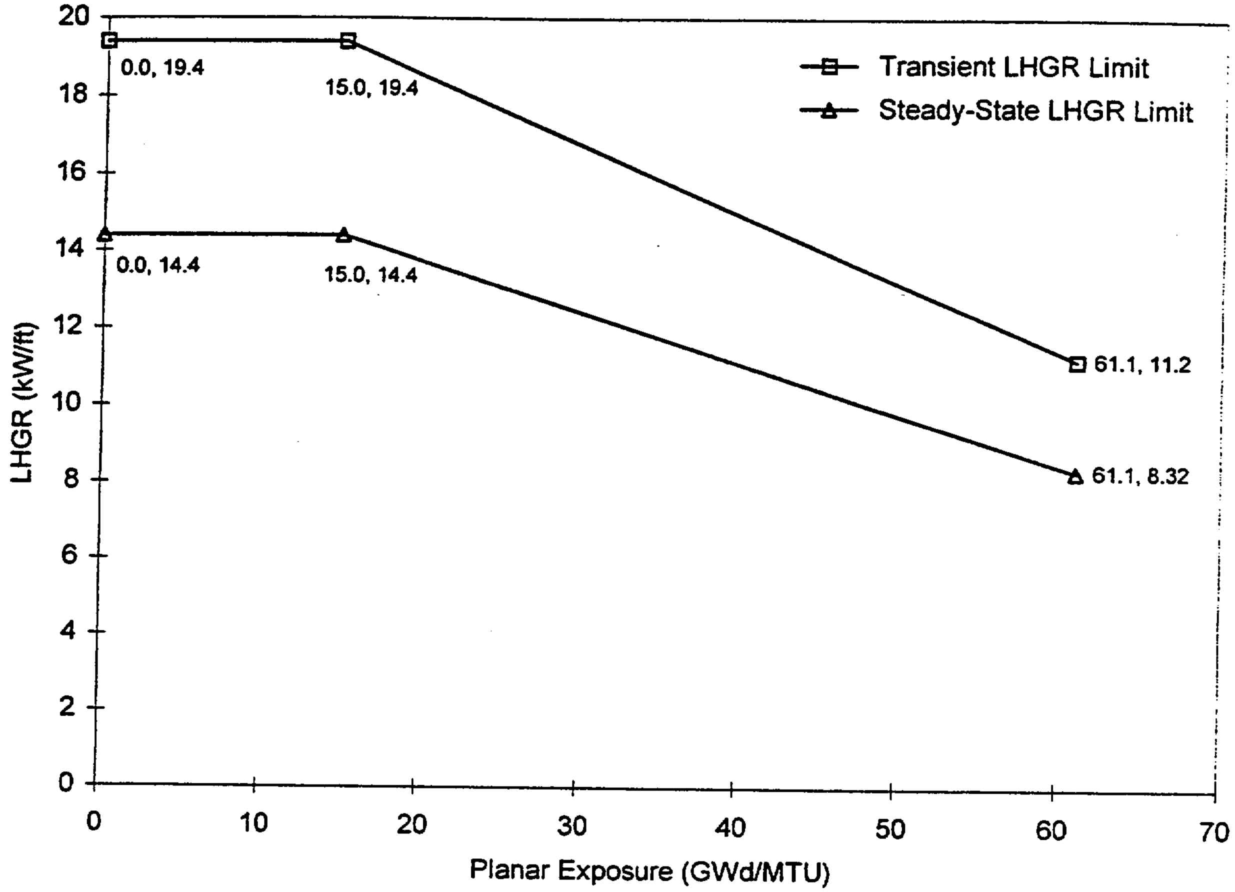


Figure 7.1 Steady-State and Protection Against Power Transient LHGR Limits for ATRIUM-9B Offset Fuel

8.0 Methodology References

See XN-NF-80-19(P)(A) Volume 4 Revision 1 for a complete bibliography.

8.1 ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.

8.2 ANF-524(P)(A) Revision 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors*, Advanced Nuclear Fuels Corporation, November 1990.

8.3 ANF-1125(P)(A) and Supplements 1 and 2, *ANFB Critical Power Correlation*, Advanced Nuclear Fuels Corporation, April 1990.

8.4 XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, *Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology*, Advanced Nuclear Fuels Corporation, November 1990.

8.5 EMF-CC-074(P)(A) Volume 1, *STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain* and Volume 2, *STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report*, Siemens Power Corporation, July 1994.

8.6 EMF-1125(P)(A), Supplement 1 Appendix C, *ANFB Critical Power Correlation Application for Co-Resident Fuel*, Siemens Power Corporation, August 1997.

8.7 ANF-1125(P)(A), Supplement 1 Appendix E Revision 0, *ANFB Critical Power Correlation Determination of ATRIUM™-9B Additive Constant Uncertainties*, Siemens Power Corporation, September 1998.

9.0 Additional References

9.1 Not used.

9.2 Not used.

9.3 EMF-2415 Revision 0, *Quad Cities Unit 1 Cycle 17 Plant Transient Analysis*, Siemens Power Corporation, August 2000.

9.4 EMF-96-184(P), *LOCA Break Spectrum Analysis for Quad Cities Units 1 and 2*, Siemens Power Corporation, December 1996.

9.5 EMF-2348(P) Revision 0, *Quad Cities LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™-9B Fuel*, Siemens Power Corporation, February 2000.

9.6 EMF-96-037(P) Revision 1, *Quad Cities Extended Operating Domain (EOD) and Equipment Out of Service (EOOS) Safety Analysis for ATRIUM™ Fuel*, Siemens Power Corporation, September 1996.

9.7 EMF-2412(P) Revision 1, *Fuel Design Report for Quad Cities Unit 1 Cycle 17 ATRIUM™-9B Fuel Assemblies*, Siemens Power Corporation, August 2000.

9.8 EMF-98-009(P) Revision 0, *Fuel Design Evaluation for Siemens Power Corporation ATRIUM-9B BWR Reload Fuel*, Siemens Power Corporation, April 1998.

9.9 EMF-96-013(P), *Criticality Safety Analysis for ATRIUM™-9B Fuel Quad Cities Units 1 and 2 Spent Fuel Storage Pools (Boraflex Racks)*, Siemens Power Corporation, June 1996.

9.10 EMF-96-148(P) Revision 1, *Criticality Safety Analysis for ATRIUM™-9B Fuel Dresden and Quad Cities New Fuel Storage Vaults*, Siemens Power Corporation, September 1996.

9.11 Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Quad Cities Unit 1 Cycle 17 Transient Power History Data for Confirming Mechanical Limits for GE Fuel," DEG:00:132, June 5, 2000.

9.12 EMF-2222(P) Revision 1, *Dresden and Quad Cities Evaluation of Changed Analytical Neutron Flux Scram and Safety Valve Set Points*, Siemens Power Corporation, June 2000.

Distribution

Controlled Distribution

D. E. Garber, 38 (9)
S. W. Jones, 34
S. A. Tylinski, 23

<u>Notification List</u>
(e-mail notification)

O. C. Brown
M. E. Garrett
K. D. Hartley
T. M. Howe
R. R. Schnepp
J. A. White
A. W. Will

Attachment 4

Quad Cities Unit 1 Cycle 17

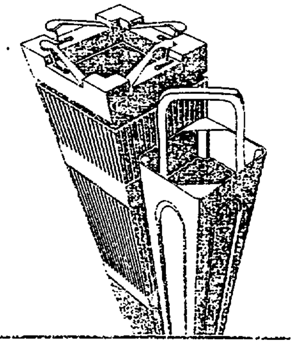
Plant Transient Analysis Report

SIEMENS

EMF-2415
Revision 0

Quad Cities Unit 1 Cycle 17 Plant Transient Analysis

August 2000



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Nuclear Division

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Quad Cities Unit 1 Cycle 17
Plant Transient Analysis

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

Contents

Tables

Figures

Nomenclature

AFC	automatic flow control
APRM	average power range monitor
ATWS	anticipated transient without scram
BPVOOS	bypass valve out-of-service
CPR	critical power ratio
EFPH	effective full power hours
ELLLA	extended load line limit analysis
EOC	end of cycle
EOD	extended operating domain
EOFP	end of full power
EOOS	equipment out-of-service
FFTR	final feedwater temperature reduction
FHOOS	feedwater heaters out-of-service
FWCF	feedwater controller failure
ICF	increased core flow
LFWH	loss-of-feedwater heating
LHGR	linear heat generation rate
LPRM	local power range monitor
LRNB	load rejection no bypass
LRNB-USM	load rejection no bypass – unpiped safety valve margin
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MFC	manual flow control
MFLCPR	maximum fraction of limiting critical power ratio
MSIV	main steam isolation valve
MSIVC-USM	main steam isolation valve closure – unpiped safety valve margin
NBR	net boiler rated steam flow
NRC	Nuclear Regulatory Commission
NSS	nominal scram speed
OLMCPR	operating limit minimum critical power ratio
PLU	power load unbalance
PLUOOS	power load unbalance out-of-service

Nomenclature
(Continued)

RPT	recirculation pump trip
RVOOS	relief valve out-of-service
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SPC	Siemens Power Corporation
SRV	safety relief valve
SRVOOS	safety relief valve out-of-service
SVOOS	safety valve out-of-service
TCV	turbine control valve
TIP	traversing incore probe
TIPOOS	traversing incore probe out-of-service
TLO	two-loop operation
TSSS	technical specification scram speed
TSV	turbine stop valve
TTNB	turbine trip no bypass
ΔCPR	change in critical power ratio

1.0 Introduction

This report describes the plant transient analyses performed by Siemens Power Corporation (SPC) in support of the reload for Quad Cities Unit 1 Cycle 17 (QC1C17). The Cycle 17 core contains 288 exposed GE10 assemblies, 200 exposed ATRIUM™-9B* offset assemblies and 236 fresh ATRIUM-9B offset assemblies. The ATRIUM-9B offset fuel assemblies use the SPC advanced channel and an offset lower tie plate. The limiting change in critical power ratio (ΔCPR) which precludes fuel damage to these fuel types in the event of anticipated plant transients during Cycle 17 operation is presented in this report. The analyzed core design is documented in Reference 1.

For QC1C17 Commonwealth Edison Company (ComEd) has responsibility for portions of the reload safety analysis. This document describes only the Cycle 17 analyses performed by SPC; ComEd analyses are described elsewhere. This document alone does not necessarily identify the limiting events or the appropriate operating limits for Cycle 17. The limiting events and operating limits must be determined in conjunction with results from ComEd analyses. The scope of the analyses performed by SPC is defined in Reference 2.

The analyses reported in this document are performed using the plant transient analysis methodology approved by the Nuclear Regulatory Commission (NRC) for generic application to BWRs (References 3 and 4). The methods employed for this analysis include the use of the COTRANSA2 system analysis methods (Reference 5), the use of safety limit methodology (Reference 6), the use of ANFB critical power correlation (References 7, 9, and 10), and the use of the CASMO-3G/MICROBURN-B code package (Reference 11). The transient analyses for Quad Cities Unit 1 Cycle 17 were performed with the parameters documented in Reference 12. This analysis supports operation in accordance with the power/flow operating map shown in Figure 1.1. The NRC technical limitations as stated in the methodology (References 3, 5, 6, 7, and 11) have been fully satisfied by this analysis. SPC has performed time step size sensitivity studies to assure that the numerical solution in the COTRANSA2 code converged. Section 6.0 describes the results of the off-rated analysis performed to demonstrate that the full power minimum critical power ratio (MCPR) operating limit, together with the reduced flow MCPR limits, protect operation throughout this map.

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* ATRIUM is a trademark of Siemens.

The ATRIUM-9B offset fuel assemblies introduced to QC1C17 have been evaluated to be hydraulically compatible with GE10 fuel resident in the reactor.

Within this report, several Quad Cities licensing reports are mentioned. In summary, the major reports are identified as:

- ***The generic extended operating domain (EOD) and equipment out-of-service (EOOS) report (Reference 13).*** Issues addressing generic EOD and EOOS documentation, penalties, trends and other generic EOD/EOOS data are referring to this report.
- ***The cycle-specific reload report (Reference 1).*** Issues addressing Cycle 17 analyses performed by SPC are referring to this report. The reload report is a summary of licensing limits.
- ***The cycle-specific plant transient report (this report).*** Issues addressing Cycle 17 thermal limits, pressure margins, and transients are referring to this report.

The structure of this report is given as:

- Section 2.0 is the summary of thermal limits and pressure margins for Cycle 17 operation.
- Section 3.0 is the Cycle 17 evaluation of the Quad Cities disposition of events and the identification of cycle-specific analyses.
- Section 4.0 is the Cycle 17 transient analyses for thermal margin.
- Section 5.0 is the Cycle 17 maximum overpressurization analyses.
- Section 6.0 is the Cycle 17 evaluation of off-rated power and flow operation.
- Section 7.0 is the Cycle 17 evaluation of cycle-specific EOD/EOOS OLMCPR penalties.

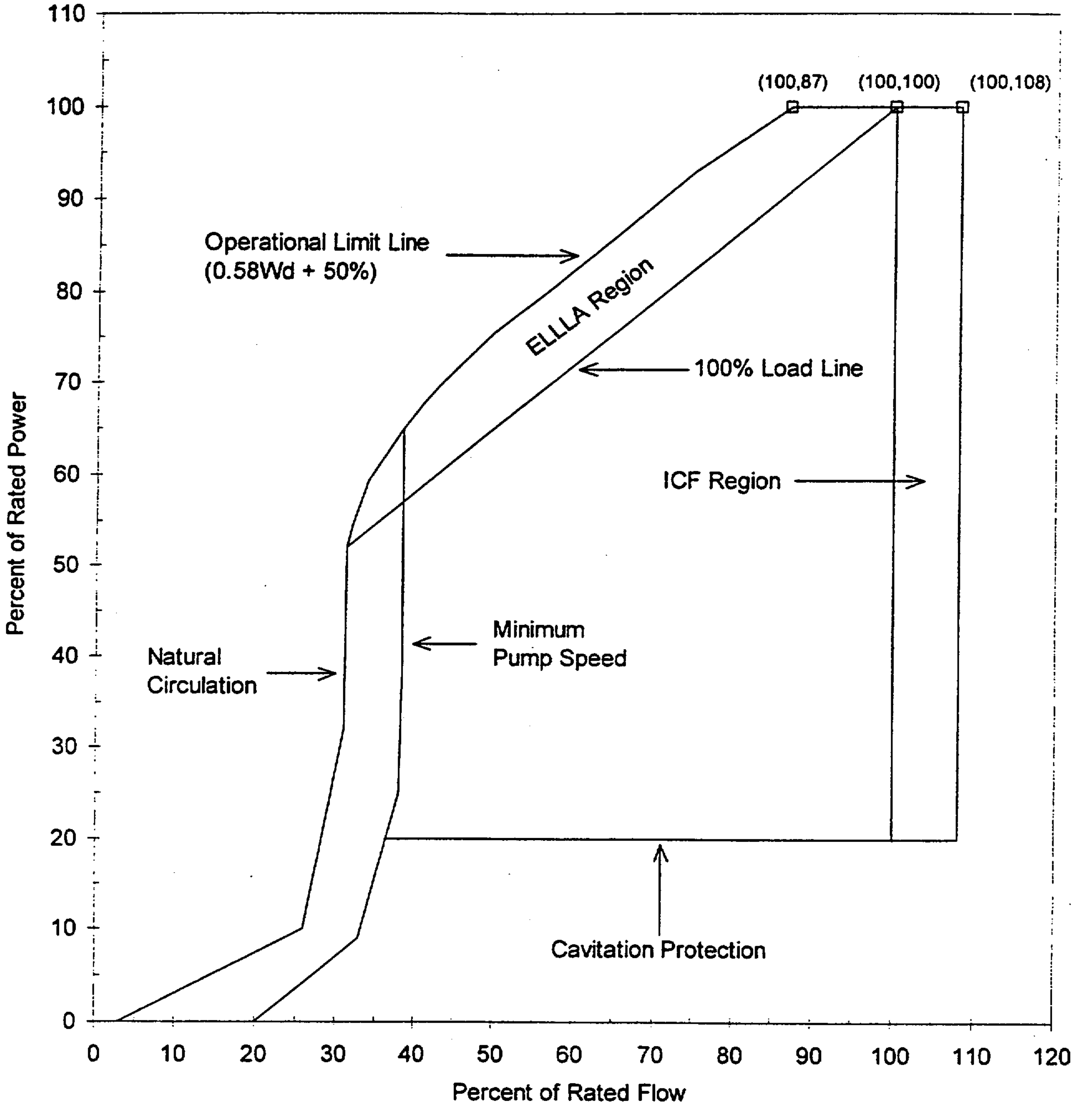

Figure 1.1 Quad Cities Unit 1 Operating Power/Flow Map

2.0 Summary

The determination of thermal margin requirements for Quad Cities Unit 1 Cycle 17 was based on the consideration of various operational transients. The most limiting transients for determination of thermal margins in Quad Cities applications in each general category of events are identified in Reference 13. Additionally, a disposition of Chapter 15 events is provided in Reference 28 for the changed analytical neutron flux scram and safety valve set points. The limiting MCPR transients determined in Reference 13 and considered in this report are the generator load rejection no bypass to the condenser (LRNB) and the feedwater controller failure (maximum demand) event (FWCF). The loss-of-feedwater heating event (LFWH) is the responsibility of ComEd for Quad Cities Unit 1 Cycle 17. Other potentially limiting MCPR transients (such as the rod withdrawal error) are either considered in the cycle reload report or are the responsibility of ComEd.

The turbine trip no bypass to the condenser (TTNB) event is nonlimiting for Cycle 17 (see Table 3.1) and is therefore not explicitly analyzed. LRNB and FWCF thermal margin analyses at 100% power/87% flow are also nonlimiting and are therefore not analyzed.

The change in critical power ratio (ΔCPR) for the base case transients is presented in Table 2.1 for Technical Specification scram speed (TSSS). The safety limit MCPR (SLMCPR) analysis for Quad Cities Unit 1 Cycle 17 supports a value of 1.11 for two-loop operation (TLO) and 1.12 for single-loop operation (SLO). These values support all normal and EOD/EOOS conditions and apply to all fuel types (GE10 and ATRIUM-9B offset) in the core for Cycle 17 and include the effects of channel bow and up to 40% TIP strings out-of-service (TIPOOS). Therefore, the SLMCPRs of 1.11/1.12 given in the Technical Specifications for TLO/SLO are applicable.

The MCPR operating limits (OLMCPRs) based on transients considered in this report are contained in Table 2.2. Base case OLMCPRs are obtained by adding the limiting ΔCPR (Table 2.1) for each fuel type to the plant Technical Specification two-loop SLMCPR of 1.11. EOD/EOOS and BPVOOS OLMCPRs presented in Table 2.2 are obtained by adding the appropriate OLMCPR penalties to the base case OLMCPRs. OLMCPRs are provided for all fuel types in the core for Cycle 17. Key parameters from the transient analyses are provided in Table 2.3.

The limiting system pressure for the maximum overpressurization events was calculated for the postulated closure of all main steam isolation valves (MSIVs) without credit for activation of the MSIV position scram, without pressure relief from the relief valves (RV), and without pressure relief from the safety/relief valve (SRV). All maximum overpressure analyses assume only three of the highest pressure set-point safety valves are operable. The anticipated transient without scram (ATWS) recirculation pump trip (RPT) at 1250 psig is modeled. The results of this analysis, as shown in Table 2.2, indicate that the requirements of the ASME code regarding overpressure protection are met for the limiting EOD/EOOS condition. Specifically, the peak vessel pressure limit of 1375 psig and the steam dome pressure limit of 1345 psig are protected.

The discussions and analyses in Sections 6.0 and 7.0 confirm that the full power MCPR operating limits adequately protect the core for reduced power and EOD/EOOS operation.

Analyses and limits presented in this report support base case operation up to EOFP with bypass valve(s) out-of-service and operation with various combinations of EOD and EOOS conditions. The EOD/EOOS conditions addressed in this report are identified in Table 2.4.

For Cycle 17, a cycle-specific OLMCPR penalty is applied to the base case OLMCPRs to support EOD/EOOS operation. The EOD/EOOS OLMCPR penalty for GE10 and ATRIUM-9B offset fuel is 0.04.* OLMCPR penalties are also applied to base case OLMCPRs to support base case operation up to EOFP with bypass valve(s) out-of-service. The OLMCPR penalties for base case operation with 1 BPVOOS and all BPVOOS are 0.01 and 0.05, respectively.

Of the EOD/EOOS operating conditions described in Table 2.4, maximum pressurization evaluations are performed with only coastdown and combined ICF/coastdown conditions. All other EOD/EOOS conditions are nonlimiting for maximum pressurization events. Limiting maximum pressurization conditions are explicitly evaluated and therefore, no EOD/EOOS pressure penalty is required for Cycle 17.

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* The cycle-specific OLMCPR penalty of 0.04 is required to support operation with FFTR, FHOOS, coastdown or any combination thereof. ~~is only applied when~~ core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1 of Reference 1. Other EOD/EOOS conditions listed in Table 2.4 require no OLMCPR penalty. The impact of SLO is applied to the SLMCPR.

Coastdown is defined as
per SPC
5/20/00

Base case analyses refer to analyses that do not fully support EOD/EOOS conditions and are representative of normal operation. The base case analyses include support for some EOD/EOOS conditions. In particular the base case analyses support increased core flow (ICF) and relief valve out-of-service (RVOOS). For maximum overpressurization analyses, base case conditions are nonlimiting with respect to EOD/EOOS conditions. Therefore, maximum overpressurization analyses are not performed for base case conditions.

Composite power history curves for the FWCF and the LRNB analyses are provided in Reference 14. ComEd must evaluate the information provided in Reference 14 to ensure that the mechanical design criteria (1% plastic strain) is satisfied for the coresident GE10 fuel.

Table 2.1 Quad Cities Unit 1 Cycle 17 Base Case ΔCPRs at Rated Power With TSSS Insertion Times

	ΔCPR	
Transient	GE10	ATRIUM-9B Offset
Load Rejection No Bypass		
100% power / 108% flow	0.39	0.35
100% power / 100% flow	0.37	0.31
Feedwater Flow Controller Failure		
100% power / 108% flow	0.40	0.35
100% power / 100% flow	0.38	0.33
Loss-of-Feedwater Heating (LFWH)**		

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* Analysis of the LFWH is the responsibility of ComEd for Quad Cities Unit 1 Cycle 17.

Table 2.2 Quad Cities Unit 1 Cycle 17 MCPR Operating Limit and Maximum Pressurization Summary

	MCPR Operating Limit[*], [†]	
Operating Conditions	GE10	ATRIUM-9B Offset
Base case	1.51	1.46
Base case with 1 BPVOOS	1.52	1.47
Base case with all BPVOOS	1.56	1.51
EOD/EOOS[‡]	1.55	1.50

	Maximum Pressurization (psig)		
Transient	Steam Dome	Lower Plenum	Steam Lines
MSIV closure without position scram (100% power / 100% flow, EOD/EOOS)	1335	1361	1335

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[*] Based on a plant Technical Specification two-loop SLMCPR of 1.11 and analysis of the limiting system transient analyzed in this report. The actual cycle operating limit may be higher if analyses within ComEd's scope of responsibility result in a ΔCPR higher than those in Table 2.1. For single-loop operation, the Technical Specification SLO SLMCPR of 1.12 increases the OLMCPR by 0.01. Refer to Section 6.2 for reduced flow MCPR limits.

[†] The operating limit for GE10 is based on FWCF 100% power/108% flow. The operating limit for ATRIUM-9B is based on LRNB and FWCF 100%power/108% flow. These results are shown in Table 2.1.

[‡] The cycle-specific OLMCPR penalty of 0.04 is required to support operation with FFTR, FHOOS, coastdown or any combination thereof ~~is only applied when~~ core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1 of Reference 1. Other EOD/EOOS conditions listed in Table 2.4 require no OLMCPR penalty.

Coastdown is defined as
per SPC
9/20/00

Table 2.3 Quad Cities Unit 1 Cycle 17 Results of Plant Transient Analysis With TSSS Insertion Times

Event	Maximum Neutron Flux (% of Rated)	Maximum Core Average Heat Flux (% of Rated)	Maximum Vessel*/ Dome Pressure (psig)
Load Rejection No Bypass			
100% power / 108% flow	626	130	1300 / 1266
100% power / 100% flow	579	129	1300 / 1269
Feedwater Flow Controller Failure			
100% power / 108% flow	610	133	1189 / 1154
100% power / 100% flow	563	132	1186 / 1154
MSIV Closure Without Position Scram (EOD/EOOS)			
100% power / 108% flow	330	132	1360 / 1332
100% power / 100% flow	324	130	1361 / 1335

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* Lower plenum pressure.

Table 2.4 EOD and EOOS Operating Conditions*

Extended Operating Domain Conditions
Increased core flow (ICF)†
Final feedwater temperature reduction (FFTR)
Coastdown
Combined ICF/FFTR
Combined ICF/coastdown
Combined FFTR/coastdown
Combined ICF/FFTR/coastdown

Equipment Out-of-Service Conditions‡
Feedwater heater(s) out-of-service (FHOOS)
Single-loop operation (SLO) - Recirculation loop out-of-service§
Relief valve out-of-service (RVOOS)†
Safety/relief valve safety function out-of-service (SRVOOS) for maximum overpressurization events†
Up to 40% TIP strings out-of-service (TIPOOS)**

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* Base case operation up to EOFP with bypass valve(s) out-of-service is analyzed separately and is not considered for EOD/EOOS operation.

† Base case analyses are performed with this condition.

‡ EOOS conditions, with the exception of FHOOS, are supported for both EOD conditions and standard operating conditions.

§ SLO adds 0.01 to the TLO SLMCPR.

** 40% TIPOOS with 100% TIP strings available at startup, 50% of the LPRMs OOS (LPRM substitution model on or off), and 2500 EFPH LPRM calibration interval. TIPOOS is evaluated in the SLMCPR analysis.

3.0 Disposition of Events

The initial disposition of events for Quad Cities is documented in Section 3.0 of Reference 13. Additionally, a disposition of Chapter 15 events is provided in Reference 28 for the changed analytical neutron flux scram and safety valve set points. The disposition of events for Cycle 17 is based on differences between principal transient analysis parameters used for Quad Cities Unit 1 Cycle 16 and Quad Cities Unit 1 Cycle 17. Differences between the QC1C17 plant parameters (Reference 12) and the QC1C16 plant parameters (Reference 15) are identified in Table 3.1. The differences do not change the conclusions of the disposition of events provided in References 13 and 28. The Cycle 17 analyses are identified in Reference 2.

Table 3.1 Quad Cities Unit 1 Cycle 17 Evaluation of Plant Parameter Changes on Disposition of Events

Parameter Change (From/To)	Impact	Resolution
Analyzed feedwater/steam flow rate, Mlbm/hr (9.759 to 9.9)	ΔCPR and maximum pressurization results increase slightly with higher steam flow.	Parameter change will not result in new limiting events. All limiting events are evaluated on a cycle-specific basis (see Sections 4.0, 5.0, and 7.0).
Steam flow versus feedwater temperature, normalized (340°F to 350°F at rated steam flow)	Insignificant, this data is used in heat balance calculations to determine steam flow for off-rated conditions. Higher feedwater temperatures result in slightly higher steam flow rates.	—
Low water level trip, in (511 to 503)	None. Scram is not initiated from low water level for any transient event evaluated by SPC.	—
Reactor internal repair hardware volume, ft^3 (new parameter, 24.3 ft^3)	None. This parameter has no effect on licensing analyses	—
Relief valve closing time, sec (0.25 to 10.0 (RV)) (0.15 to 10.0 (SRV))	None. The increased closing times have no impact on calculated results for thermal margin or overpressurization transients.	—
Turbine bypass valve parameters for operation with 1 valve out-of-service (new parameters)	These parameters are used for a single FWCF analysis to determine a specific EOOS penalty.	Calculations documented in Section 7 establish the OLMCPR penalty for operation with BPVOOS.
Control rod position versus scram time (NSS times omitted)	None. NSS is not supported for Cycle 17.	—
Safety valves available (TTNB, LRNB and FWCF events) (safety valves: 8 to 3)	None. The change in the number of specified safety valves will have no effect on ΔCPR results, since the valves either do not open or open only after the time of MCPR.	—

Table 3.1 Quad Cities Unit 1 Cycle 17 Evaluation of Plant Parameter Changes on Disposition of Events
(Continued)

Parameter Change (From/To)	Impact	Resolution
TCV closure for TTNB event and TSV closure for LRNB event (new parameter)	The TSV and TCV are closed for both the TTNB event and the LRNB event. Therefore, the only difference between the TTNB and LRNB events is the scram delay time, 0.07 scram delay from TSV position for TTNB and 0.08 scram delay from TCV fast closure for LRNB. ΔCPR results for the TTNB event will be bound by LRNB ΔCPR results for all analyzed conditions.	The TTNB event is no longer considered a limiting event due to the change in TCV/TSV modeling. LRNB ΔCPR results bound TTNB results.
TCV position (% open) versus steam flow (% total valve capacity) (48% open to 65% open at 100% flow)	The increased valve position at rated steam flow increases the time required for the TCV to fully close. Consequently, TCV events become slightly less severe.	Parameter change will not result in new limiting events. All limiting events are evaluated on a cycle-specific basis (see Sections 4.0, 5.0, and 7.0).
TCV closure for FWCF event (new parameter)	Due to the initial position of the TCV, the TSV and TCV reach the fully closed position at approximately the same time.	Parameter change will not result in new limiting events. FWCF events are evaluated on a cycle-specific basis (see Sections 4.0 and 7.0).
Safety valves available for maximum pressurization events (9 to 3)	Peak pressure results for limiting maximum pressurization events will increase by approximately 13 – 18 psi due to the increased SV pressure set point and decreased flow capacity.	Parameter change will not result in new limiting events. All limiting events are evaluated on a cycle-specific basis (see Section 5.0).
TCV closure for TSV maximum pressurization event and TSV closure for TCV maximum pressurization event (new parameters)	Since both valves are closed for maximum pressurization events and direct scram on valve position is disabled, the TCV and TSV maximum pressurization events are identical.	Separate TCV and TSV maximum pressurization evaluations are no longer required.

Table 3.1 Quad Cities Unit 1 Cycle 17 Evaluation of Plant Parameter Changes on Disposition of Events *(Continued)*

Parameter Change (From/To)	Impact	Resolution
Relief valve closing time (unpiped safety valve margin events), sec (0.20 to 10.0 (RV)) (0.15 to 10.0 (SRV))	None. The increased closing times have no impact on calculated results for unpiped safety valve margin evaluations.	—
Scram insertion (unpiped safety valve margin events), notch position versus time, sec (3.147 to 3.065 for full insertion)	Faster scram insertion increases pressure margin to the lowest set point safety valve.	No effect on licensing analyses. Unpiped safety valve margin analyses are performed on a cycle-specific basis (see Appendix A).
Relief valve nominal opening set points (unpiped safety valve margin events), psia (1114.7 to 1109.7 (RV) and 1134.7 to 1129.7 (SRV))	Reduced relief valve opening set points increases pressure margin to the lowest set point safety valve.	No effect on licensing analyses. Unpiped safety valve margin analyses are performed on a cycle-specific basis (see Appendix A).
Power load unbalance out-of-service (new parameters)	These parameters are used to determine cycle-specific MFLCPR multipliers that protect the SLMCPR at limiting base case and EOD/EOOS conditions.	Calculations documented in Appendix B establish the MFLCPRs for operation with PLUOOS.
Combined steam flow limiter setting, %NBR (105 to 115)	None. A higher setting would affect bypass valve operation. Bypass valves could open prior to TCV/TSV closure for FWCF events. However, SPC control system settings restrict the bypass valve from opening prior to full closure of the TCV/TSV. The increase could potentially make the pressure regulator failure – wide open event more severe. However, the event is essentially either an MSIV closure or a TSV closure at reduced power and will remain bound by the LRNB/TTNB event. The 115% limiter setting is recommended in GE SIL 502 to avoid the potential for exceeding the SLMCPR during a TCV slow closure event.	—

4.0 Transient Analysis for Thermal Margin

This section describes the analyses that were performed to determine the full power MCPR operating limits for Cycle 17 of Quad Cities Unit 1.

4.1 *Design Basis*

The plant transient analyses for Quad Cities Unit 1 Cycle 17 determined that the limiting transient initial conditions were at rated power and 108% rated core flow. Rated reactor plant parameters for the thermal margin analyses are shown in Table 4.1. The most limiting point in the cycle is when the control rods are fully withdrawn from the core. The thermal margins established for the end of full power (EOFP) capability are conservative for cases where control rods are partially inserted. The transient analyses were performed assuming the conservative conditions in Table 4.2. All transients were performed with the most limiting (lowest set-point) relief valve out-of-service (RVOOS). In addition, the relief function of the safety/relief valve (SRV) was conservatively modeled as RV (i.e., slower response time and lower flow capacity).

Observance of the OLMCPR shown in Table 2.2 will provide adequate protection against the occurrence of boiling transition during all anticipated transients considered in this section.

4.2 *Calculation Model*

COTRANSA2 (Reference 5), XCOBRA-T (Reference 16), XCOBRA (Reference 3), and CASMO-3G/MICROBURN-B (Reference 11) are the major codes used in the thermal limits analyses as described in SPC's THERMEX methodology report (Reference 3) and neutronics methodology report (Reference 11). COTRANSA2 is a system transient simulation code which includes an axial one-dimensional neutronics model used to model the axial power shifts associated with the system overpressurization in the LRNB, FWCF, and MSIV closure transients. XCOBRA-T is a transient thermal-hydraulic code used in the analysis of thermal margins of the limiting fuel assembly. XCOBRA is a steady-state thermal-hydraulic code used in the analysis of slow flow excursion events. Fuel pellet-to-cladding gap conductance values used in the analyses were based on RODEX2 (Reference 17) calculations for the Quad Cities Unit 1 Cycle 17 core configuration. The thermal margins of the fuel assemblies are evaluated in XCOBRA-T, XCOBRA, and MICROBURN-B using the ANFB critical power correlation (References 7 and 9) with the implementation of new ATRIUM-9B additive constants (Reference 10). The applicability of the ANFB critical power correlation to GE10 fuel at Quad Cities is demonstrated in References 9 and 18.

In accordance with SPC methodology, possible limiting transients are evaluated using a consistent set of bounding input. From the results of these transients, the limiting transient event for the fresh ATRIUM-9B offset fuel is identified as the FWCF at 100% power/108% flow. Table 4.2 summarizes the values used for important parameters in the analysis. Table 4.3 provides the feedwater flow, recirculating coolant flow, and pressure regulation control system settings used in the analysis.

4.3 *Anticipated Transients*

For Quad Cities Unit 1 Cycle 17, specific events have been evaluated for thermal margin as outlined in References 13 and 28. These events are the LRNB* and FWCF. The evaluation of parameter changes provided in Section 3.0 and the disposition of events provided in References 13 and 28 demonstrate that other categories of transients are either inherently self-limiting, bounded by one of these or are part of ComEd's analysis responsibility. Reference 13 provides descriptions of the transients that are considered for the cycle-specific evaluation.

In accordance with Reference 12, all transient thermal margin analyses were performed with a conservative reduction to the design basis steam dome pressure. Thermal margin analyses at rated conditions are based on a steam dome pressure of 1005 psia, representing a 15 psi reduction from the design value of 1020 psia. Limiting transients were also analyzed with the design basis steam dome pressure; the analyses showed no significant sensitivity with respect to reduced dome pressure. Maximum overpressurization analyses are based on the design basis steam dome pressure. For operation above 90% rated power, the steam dome pressure may be reduced no more than 15 psi from the values presented in Table 4.5 (Reference 19). Steam dome pressure does not need to be monitored below 90% rated core power, because below 90% power the MCPR margin gain due to reduced power will offset any increase in ΔCPR due to a maximum dome pressure decrease (Reference 19).

Thermal margin results for the equilibrium ATRIUM-9B offset core (Reference 13), the initial ATRIUM-9B offset reload core for Quad Cities Unit 2 Cycle 15 (Reference 20) and the initial ATRIUM-9B offset reload core for Quad Cities Unit 1 Cycle 16 (Reference 21) provide sufficient evidence that the 100% power/87% flow state point is nonlimiting for all possible operating

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* Based on parameter changes described in Table 3.1, the TTNB event is no longer considered a limiting event as it is bound by the LRNB event.

domains, including standard operation and all EOD/EOOS combinations. Similarly, maximum overpressurization results provided in References 13, 20 and 21 provide sufficient evidence that base case conditions are nonlimiting relative to EOD/EOOS conditions. Therefore, as indicated in Reference 2, thermal margin evaluations are not performed at 100% power/87% flow conditions and maximum overpressurization events are not performed for base case conditions.

4.3.1 Load Rejection No Bypass

The LRNB event is more limiting than the TTNB event. Transient input parameters documented in Reference 12 specify closure of both the TCV and TSV for the LRNB and TTNB events. Consequently, the only difference in the system analysis of the TTNB and LRNB events is the scram delay time, 0.07 scram delay from TSV position for TTNB and 0.08 scram delay from TCV fast closure for LRNB. The longer scram delay for the LRNB event provides conservative results for all possible operating conditions.

In the load rejection transient, steam flow is interrupted by an abrupt closure of the TCV and coincident closure of the TSV. The resulting pressure increase causes a decrease in the void volume in the core, which in turn creates a power excursion. This excursion is mitigated in part by Doppler broadening and pressure relief, but the primary mechanisms for termination of the event are control rod insertion and regeneration of voids. A turbine trip is similar to the load rejection transient, the difference is that steam flow is interrupted by an abrupt closure of the TSV with coincident closure of the TCV.

The important parameters for these transients include the power transient (integral power) determined by the void reactivity, which affects the initial power excursion rate and is part of the intrinsic shutdown mechanism, and the control rod worth, which determines the value of the scram reactivity. Other important inputs include the control rod movement parameters (scram delay and insertion speed), which determine the event characteristics following the initial mitigation of the power excursion. From Table 2.1, the largest calculated limiting ΔCPR for the LRNB event was at 100% power/108% flow conditions.

Figures 4.1–4.4 illustrate the behavior of major system variables during the LRNB event at 100% power and 108% flow for TSSS insertion times. MCPR occurs at approximately 1.84 seconds for the ATRIUM-9B offset fuel.

4.3.2 Feedwater Controller Failure

The FWCF to maximum demand leads to an increase in feedwater flow into the reactor vessel. The excessive feedwater flow increases the subcooling in the recirculating water returning to the reactor core. This reduction in moderator temperature will result in the core power increasing to a higher equilibrium power level if no other actions occur. Eventually, the level of water in the downcomer region will rise until the high water level trip set point (L8) is reached. A turbine trip initiated on high water level results in the rapid closure of the TSV to prevent the transmission of liquid water to the turbine. The rapid closure of the TSV and coincident fast closure of the TCV produces a compression wave in the steam line which results in core void collapse and increased core reactivity. The stop valve closure initiates a scram signal at 10% TSV closure (modeled as a 0.01 second delay) and the resulting control rod insertion terminates the power increase.

In the analysis, the bypass valves do not operate before the turbine trip signal due to conservative control system assumptions (maximum combined flow limiter and bypass valve opening bias settings prevent bypass valve operation). However, the bypass valves do open as a result of the closure of the TSV. The bypass valves are assumed in the model to start opening 0.15 second after the start of TSV motion. The start of bypass valve opening corresponds to the time when the stop valves become fully closed plus a delay of 0.05 second. Although a longer TSV stroke time would result in a longer delay in bypass valve opening, a fast TSV closure results in a more severe event even though the bypass valve opens earlier. The reactor pressure increase produced by the rapid stop valve closure is mitigated by the opening of the bypass valves. The bypass valve opening time assumed in the analysis is given in Table 4.2.

FWCF analysis results are provided in Section 2.0. Figures 4.5–4.8 illustrate the behavior of major system variables during the FWCF transient at 100% power/108% flow for TSSS insertion times. MCPR occurs at 59.5 seconds for the ATRIUM-9B offset fuel. The TSV and TCV become fully closed at approximately 58.9 seconds.

4.3.3 Loss-of-Feedwater Heating

For the Quad Cities Unit 1 Cycle 17 reload, the analysis of the LFWH transient is the responsibility of ComEd.

4.4 *Safety Limit MCPR*

The safety limit MCPR (SLMCPR) for Quad Cities Unit 1 Cycle 17 operation was determined using the methodology described in Reference 6. The main input parameters and uncertainties used in the safety limit analysis are listed in Table 4.4. The radial power uncertainty includes the effects of up to 40% TIP strings out-of-service (TIPOOS) with 100% TIP strings available at startup, up to 50% of the local power range monitors (LPRM) out-of-service, and an LPRM calibration interval of 2500 effective full power hours (EFPH) as discussed in References 22 and 29.

The determination of the SLMCPR explicitly includes the effects of channel bow and relies on the following assumptions:

- Cycle 17 will not use channels for more than one fuel bundle lifetime. The GE10 fuel uses the GE advanced channel, and the ATRIUM-9B offset fuel uses the SPC advanced channel.
- Channel exposures will not exceed 50,000 MWd/MTU for GE10 fuel and 48,000 MWd/MTU for ATRIUM-9B offset fuel, based on the maximum bundle exposures at the end of Cycle 17.
- The GE advanced channel bow data for the GE10 fuel is provided in References 23 and 24 and is valid as long as Quad Cities is loaded as a control cell core, the fresh fuel loaded into Quad Cities is offset into the wide wide gap, and no new GE10 channels are inserted into the core.
- The effects of channel bow were determined using a 2x2 array with a conservative exposure configuration.

Analyses were performed with input parameters (including the radial power and local peaking factor distributions) for each exposure step in the design basis step-through including an EOFP+1500 MWd/MTU extension to cover coastdown operation. The analysis that produced the highest number of rods in boiling transition corresponds to Cycle 17 exposure of 15,935 MWd/MTU. The radial power distribution corresponding to this exposure is shown in Figure 4.9.

The limiting local power distribution for the Cycle 17 SPC fuel types with channel bow is shown in Figures 4.10 and 4.11.

The results of the analysis support a TLO SLMCPR of 1.11 for all fuel types residing in the core. Protection of this limit will assure that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition during normal operation and anticipated operational occurrences. In addition, analyses were explicitly performed to support the EOD conditions of ICF and SLO.

The TLO limit of 1.11 and an SLO limit of 1.12 support all normal and EOD/EOOS conditions identified in Table 2.4. The Quad Cities Technical Specification SLMCPR of 1.11 for TLO and 1.12 for SLO are applicable. For operation with uncalibrated LPRMs at startup, these limits are supported for cycle exposures up to 500.0 MWd/MTU (Reference 24).

4.5 *Nuclear Instrumentation Response*

The impact of loading ATRIUM-9B offset fuel into the Quad Cities core will not affect the nuclear instrumentation response. The neutron lifetime is an important parameter affecting the time response of the incore detectors. The neutron lifetime is a function of the nuclear and mechanical design of the fuel assembly, the in-channel void fraction, and the fuel exposure. The neutron lifetimes are similar for the SPC and GE Quad Cities fuel with typical values of 39×10^{-6} to 40×10^{-6} seconds for the ATRIUM-9B offset lattices and 41×10^{-6} to 43×10^{-6} seconds for the GE lattices as calculated with the CASMO-3G code at core average void exposure conditions. Therefore, the neutron lifetimes for a full core of ATRIUM-9B offset fuel, a mixed core of ATRIUM-9B offset and GE fuel, and a full core of GE fuel are essentially equivalent.

Table 4.1 Quad Cities Unit 1 Cycle 17 Design Reactor and Plant Conditions

	Thermal Margin Analysis	Maximum Overpressurization Analysis
Reactor thermal power	2511 MWt	2511 MWt
Total core flow	98.0 Mlb/hr	98.0 Mlb/hr
Core active flow	85.6 Mlb/hr	85.7 Mlb/hr
Core bypass flow*	12.4 Mlb/hr	12.3 Mlb/hr
Core inlet enthalpy†	521.6 Btu/lbm	523.6 Btu/lbm
Vessel pressures		
Steam dome	1005 psia	1020 psia
Core exit (upper plenum)	1015 psia	1030 psia
Lower plenum†	1039 psia	1054 psia
Turbine pressure	949 psia	965 psia
Feedwater/steam flow	9.9 Mlb/hr	9.9 Mlb/hr
Feedwater enthalpy†	327.1 Btu/lbm	326.6 Btu/lbm
Recirculating pump flow (per pump)	16.7 Mlb/hr	16.7 Mlb/hr

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* Includes water rod/internal water channel flow.

† These parameters vary slightly due to cycle variations (core configuration and power distribution) and to minor differences in heat balance calculations between computer codes. Differences are not significant.

Table 4.2 Quad Cities Unit 1
Significant Parameter Values Used in Analysis

High-neutron flux trip	3138.75 MWt
Time to deenergize pilot scram solenoid valves	200 msec
Time to sense fast turbine control valve closure	80 msec*
Time from high-neutron flux trip to control rod motion	290 msec†
Turbine stop valve stroke time	100 msec
Turbine stop valve position trip	90% open
Turbine control valve stroke time (total)	150 msec
Core average fuel/cladding gap‡ conductance (cycle-specific value)	1129 Btu/hr-ft^2-°F

---

* Includes a 50-msec delay for RPS logic transfer and a 30-msec delay until signal is received by RPS logic.

† Includes a 90-msec delay for signal to reach solenoid valves and a 200-msec delay for pilot scram solenoid valves to deenergize.

‡ Calculated by SPC for the Cycle 17 core using RODEX2 at rated conditions.

Table 4.2 Quad Cities Unit 1
Significant Parameter Values Used in Analysis
(Continued)

Safety/relief valve performance settings[*]	
Safety/relief valve (1 valve) Capacity per valve (relief) Capacity per valve (safety)	 155.0 lbm/sec at 1120 psig[†] 166.1 lbm/sec at 1112.4 psig[‡]
Relief valves capacity (4 valves)[§] Capacity per valve	 155.0 lbm/sec at 1120 psig
Safety valves capacity (3 valves) Capacity per valve	 179.04 lbm/sec at 1277.2 psig
Safety/relief valve[†] Opening delay Closing delay Opening time Closing time	 1.85 sec 4.0 sec 250 msec 10.0 sec
Relief valve Opening delay Closing delay Opening time Closing time	 1.85 sec 4.0 sec 250 msec 10.0 sec
MSIV stroke time	3.0 sec
MSIV position trip set point	90% open
Condenser bypass valve performance Total capacity Delay to opening (from the start of TSV motion) Opening time	 1084 lbm/sec 150 msec 0.11 sec (5% open), 0.25 sec (80% open), 0.7 sec (100% open)
Fraction of energy generated in fuel	0.965[**]
Vessel water level (above separator skirt) Normal Range of 0operation (lower bound) High-level trip	 30 in 20 in 60 in
Maximum feedwater runout flow (2 pumps)	3307 lbm/sec
Recirculating pump trip set point	1250 psig (steam dome pressure)

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[*] Valve set points are given in Reference 12.

[†] The relief valve mode of the SRV is conservatively modeled with RV flow capacity and response time.

[‡] For maximum overpressurization events, SRV function is not credited.

[§] 1 relief valve at the lowest set point is not credited.

[**] Reference 25.

Table 4.2 Quad Cities Unit 1
Significant Parameter Values Used in Analysis
(Continued)

Control Rod Insertion Time	
Position (notch)	TSSS Time (sec)
48	0.000
48	0.200
5% Inserted	0.375
45	0.419
39	0.856
20% Inserted	0.900
25	1.924
50% Inserted	2.000
5	3.484
90% Inserted	3.500
0	3.875

Table 4.3 Control System Characteristics*

Sensor time constants	
Pressure	0.50 sec
Steam flow	1.00 sec
Feedwater flow	0.25 sec
Level	1.00 sec
Feedwater control mode	Single element†
Water level controller	
Proportional gain	20%/ft
Pressure regulator settings	
Lead	3.0 sec
Large lag	7.5 sec
Small lag	0.50 sec
Gain	3.33%/psid
Bypass flow signal bias	2.5%
Combined steam flow limiter setting	115% NBR
Turbine maximum steam flow	2816.67 lbm/sec
Recirculation flow control mode	Manual

---

* The transients considered in cycle-specific analyses are mitigated by reactor scram which has a response that is faster than the feedwater control system response. The inclusion of the control system in the analysis model results in a more realistic calculated plant response. The representative parameters have an insignificant effect on pressure and thermal margins.

† Quad Cities licensing analyses are insensitive to the feedwater control system algorithms or settings. Single-element mode provides slightly more conservative results compared to manual or three-element control mode for all events based on the Dresden study in Reference 26.

Table 4.4 Input for Safety Limit MCPR Analysis

Fuel-Related Uncertainties

Parameter	Source Document	Statistical Treatment
ANFB correlation* GE10 ATRIUM-9B offset	 Reference 9 Reference 10	Convoluted
Radial peaking factor	References 22 and 29[†]	Convoluted
Local peaking factor	Reference 11	Convoluted
Assembly flow rate	Reference 6	Convoluted
Channel bow local peaking factor[‡]	Reference 6	Convoluted

Plant Measurement Uncertainties

Parameter	Unit	Value	Uncertainty (%) (Reference 12)	Statistical Treatment
Feedwater flow rate	Mlbm/hr	14.9[§]	2.62	Convoluted
Feedwater temperature	°F	352.7**	0.76	Convoluted
Core exit pressure	psia	1030	0.50	Convoluted
Total core flow	Mlbm/hr	98.0	2.50	Convoluted
Core power	MWt	3796[§]	---	Allowed to vary with heat balance

---

* Additive constant uncertainty values are used.

[†] Radial peaking factor uncertainty includes allowances for up to 40% of the TIP strings out-of-service (with POWERPLEX®-II CMSS SUBTIP methodology) with 100% TIPs available at startup, LPRM recalibration interval up to 2500 EFPH, and LPRM failures up to 50% with POWERPLEX®-II CMSS bypass methodology on or off.

[‡] Function of nominal and bowed local peaking and standard deviation of bow data.

[§] Feedwater flow rate and core power were increased above design values to attain desired core MCPR for safety limit evaluation, consistent with Reference 6 methodology.

** As determined by SPC heat balance calculations.

Table 4.5 Quad Cities Unit 1 Steam Dome Pressure - Analysis Basis

Core Power (% Rated)	Dome Pressure (psia)
100	1020
95	1012
90	1005

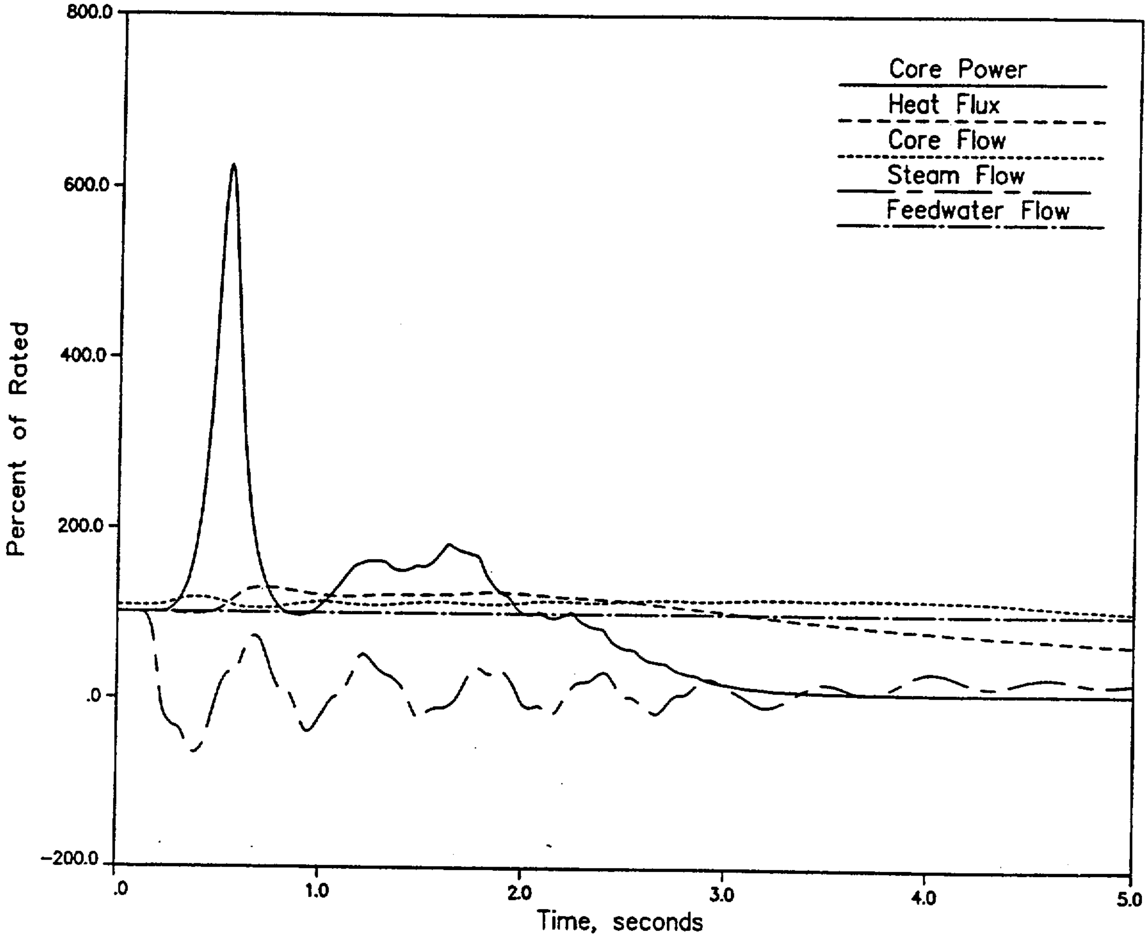


Figure 4.1 Load Rejection No Bypass at 100/108 - Key Parameters

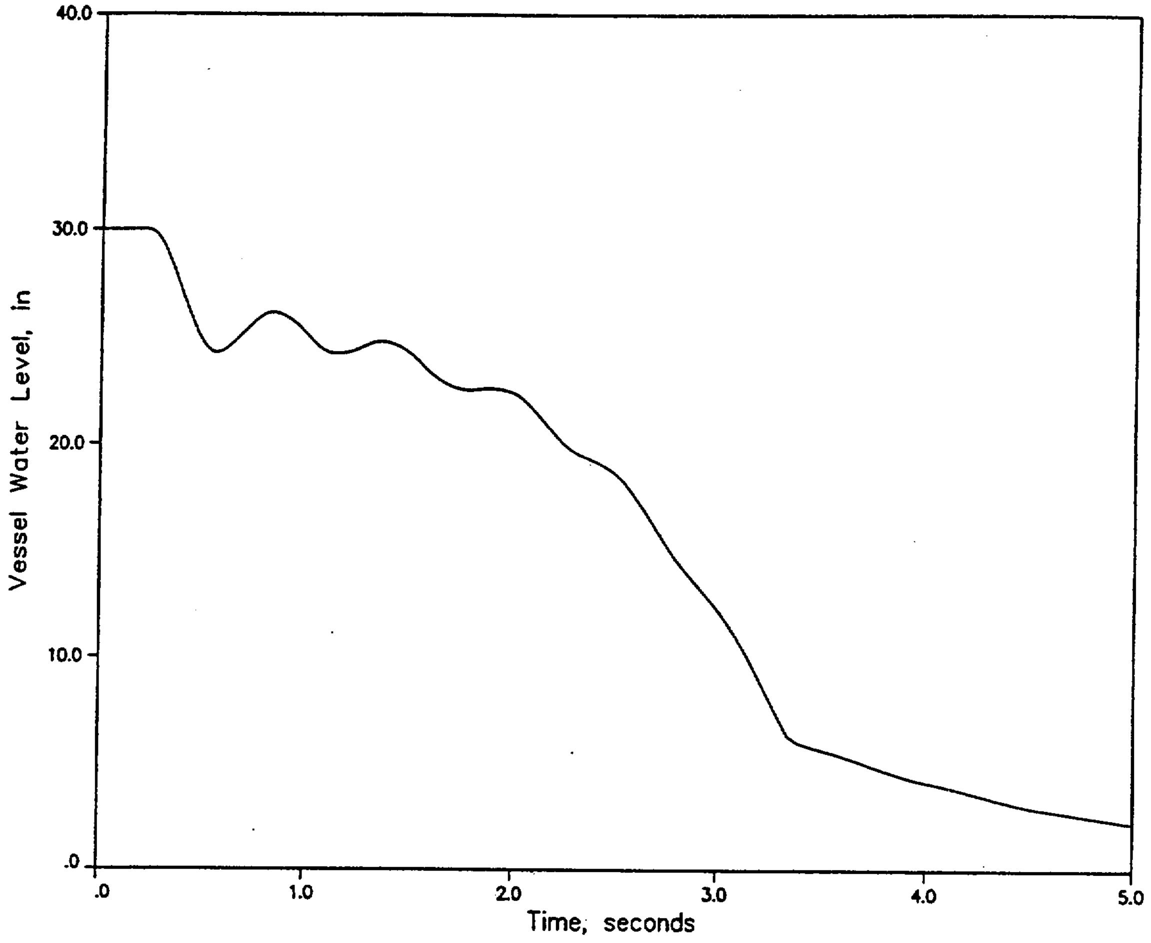


Figure 4.2 Load Rejection No Bypass at 100/108 - Vessel Water Level
(Referenced to Instrument Zero)

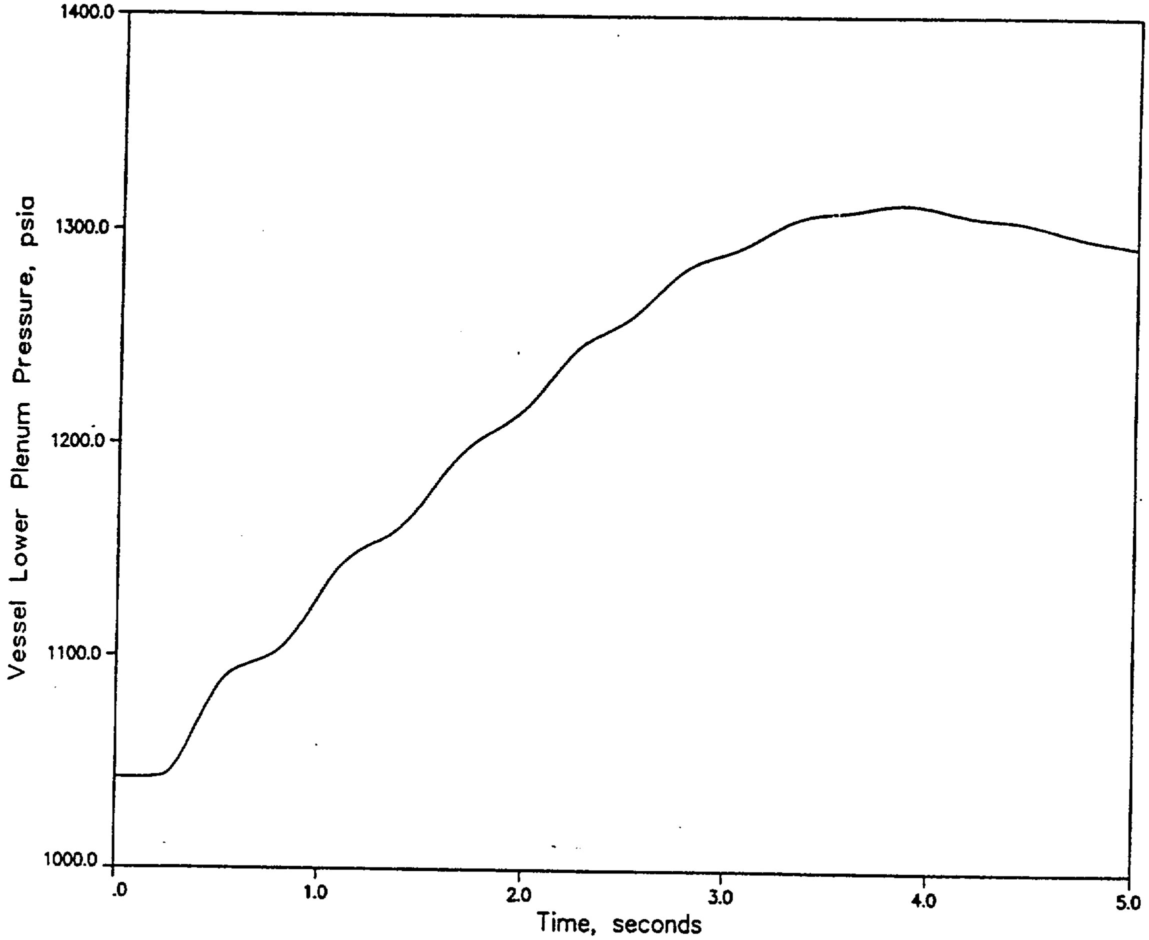


Figure 4.3 Load Rejection No Bypass at 100/108 - Vessel Pressure Response

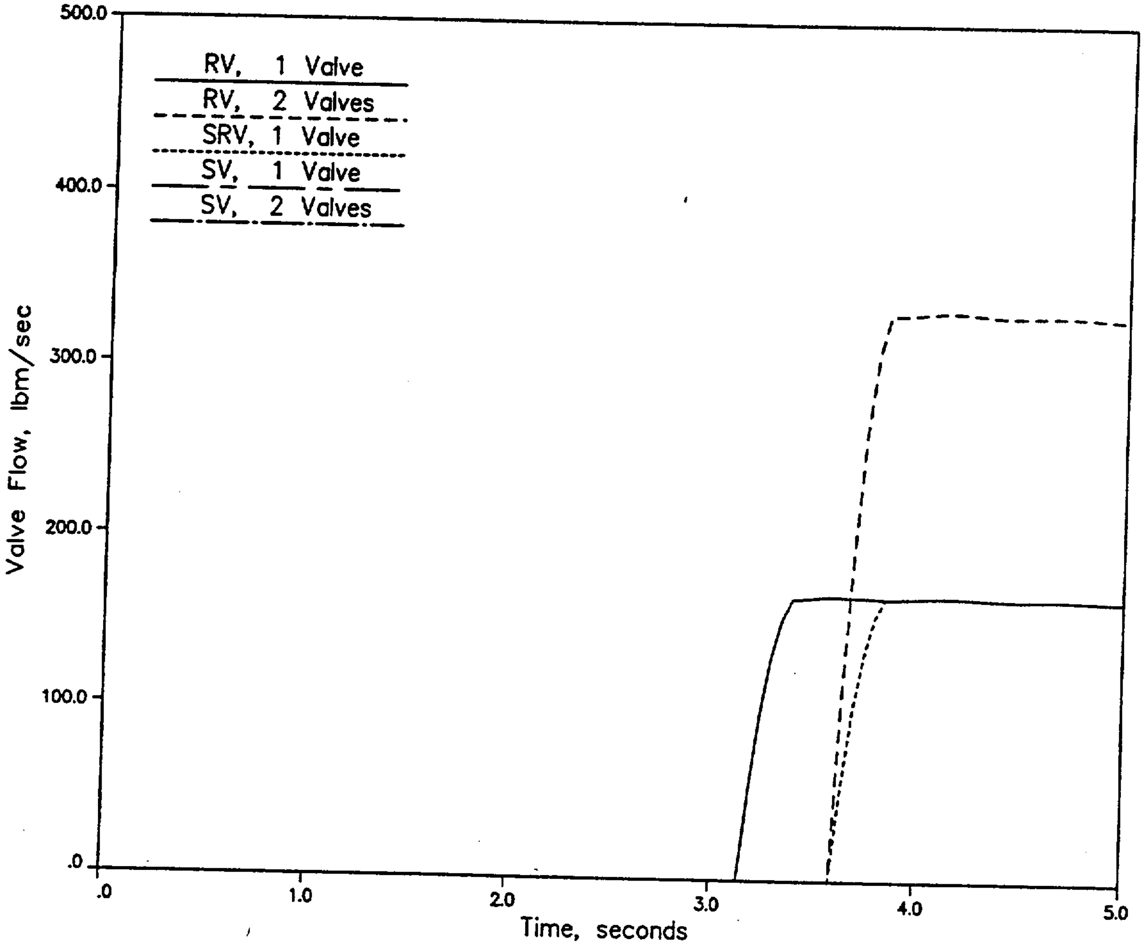
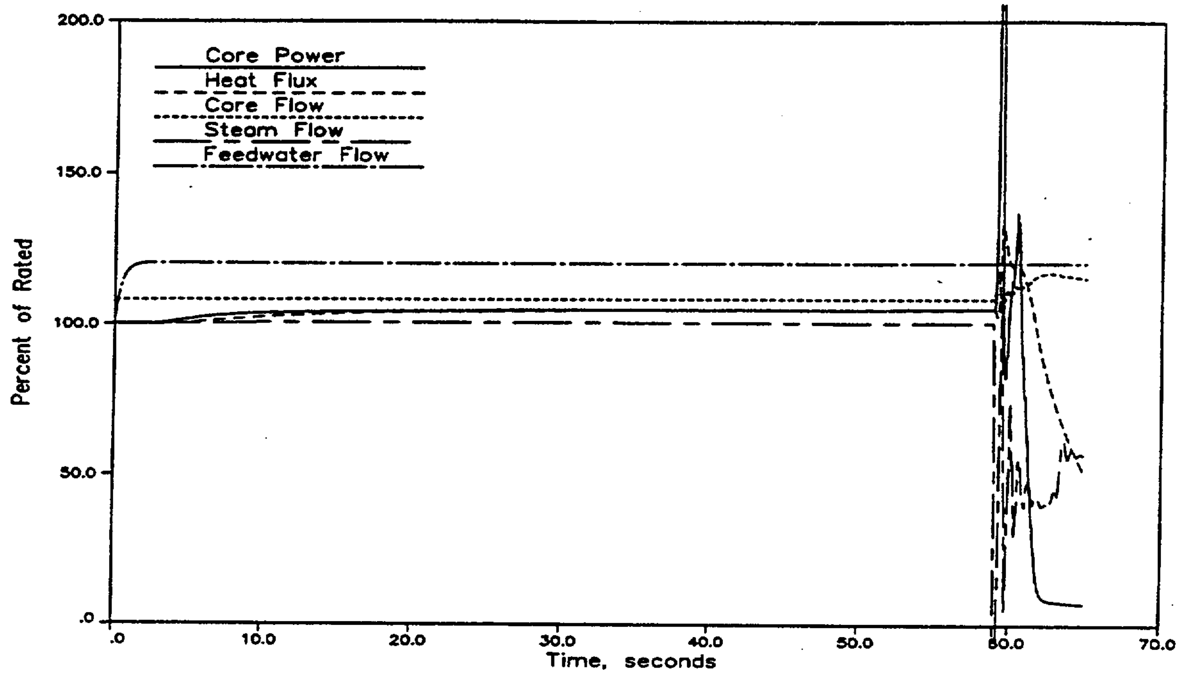


Figure 4.4 Load Rejection No Bypass at 100/108 - Safety/Relief Valve Flows



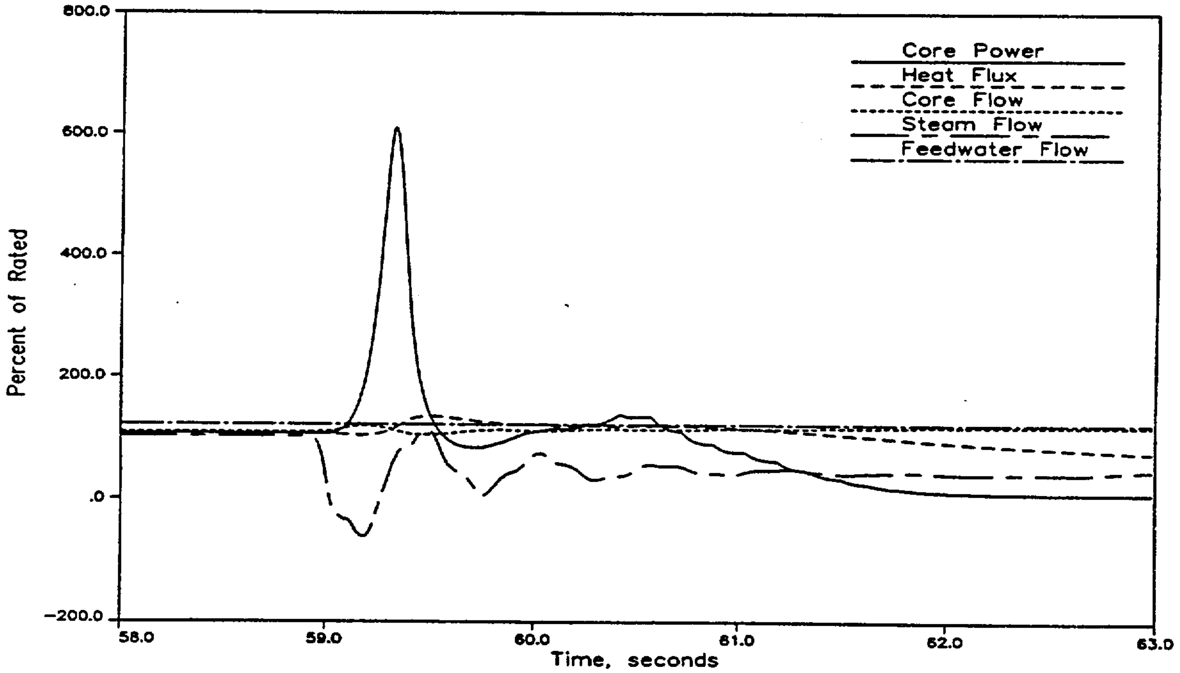


Figure 4.5 Feedwater Controller Failure at 100/108 - Key Parameters

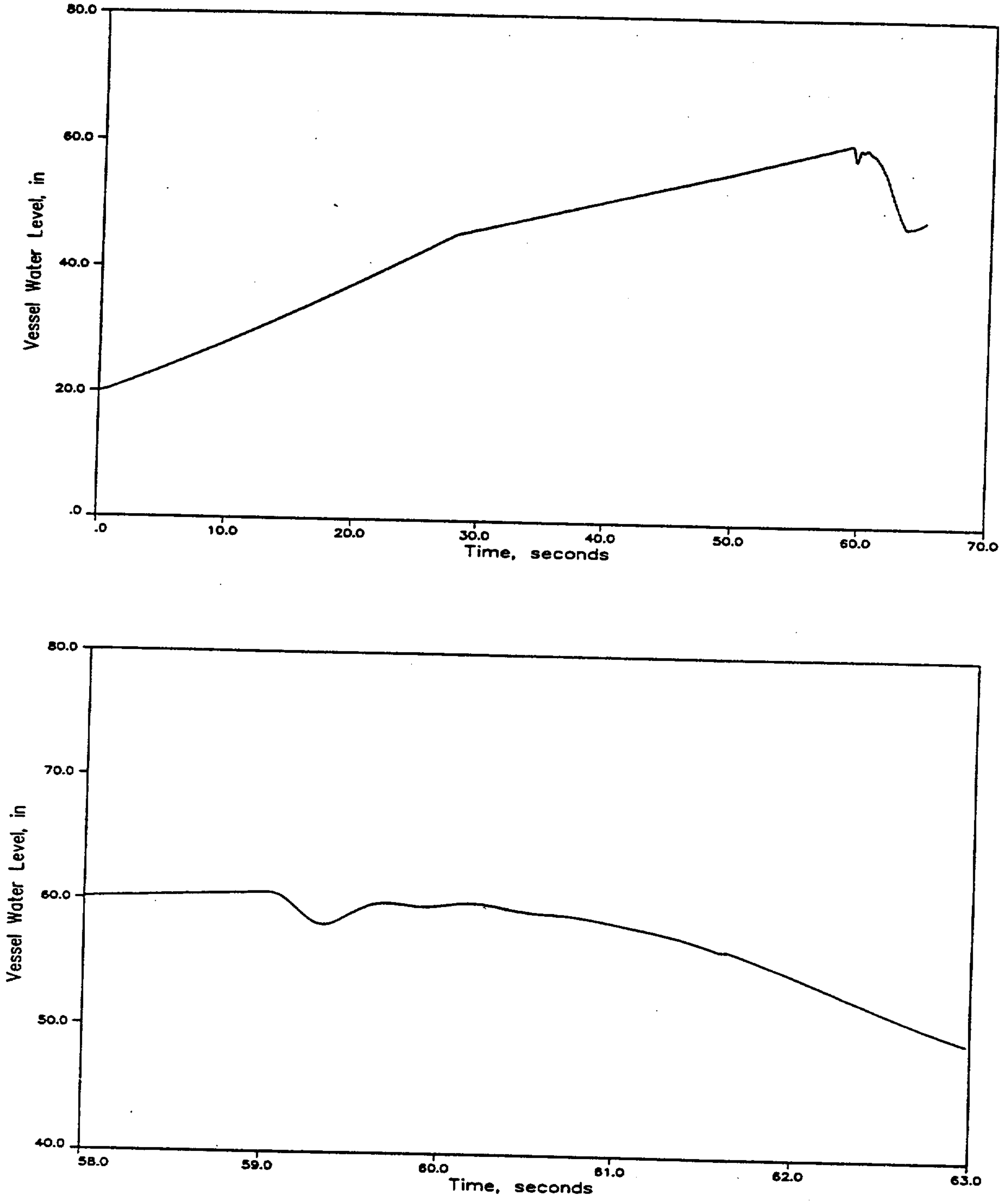


Figure 4.6 Feedwater Controller Failure at 100/108 - Vessel Water Level
(Referenced to Instrument Zero)

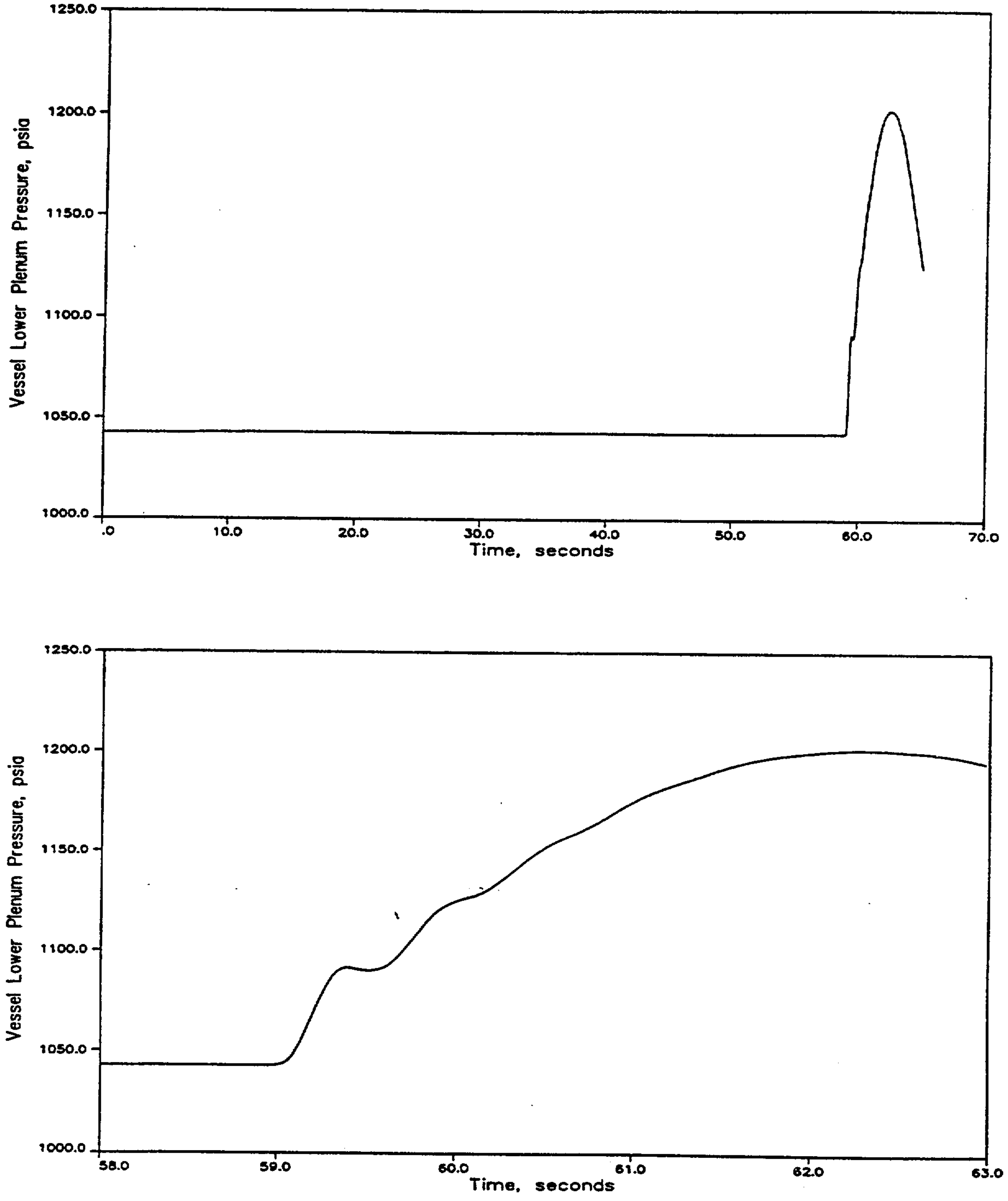
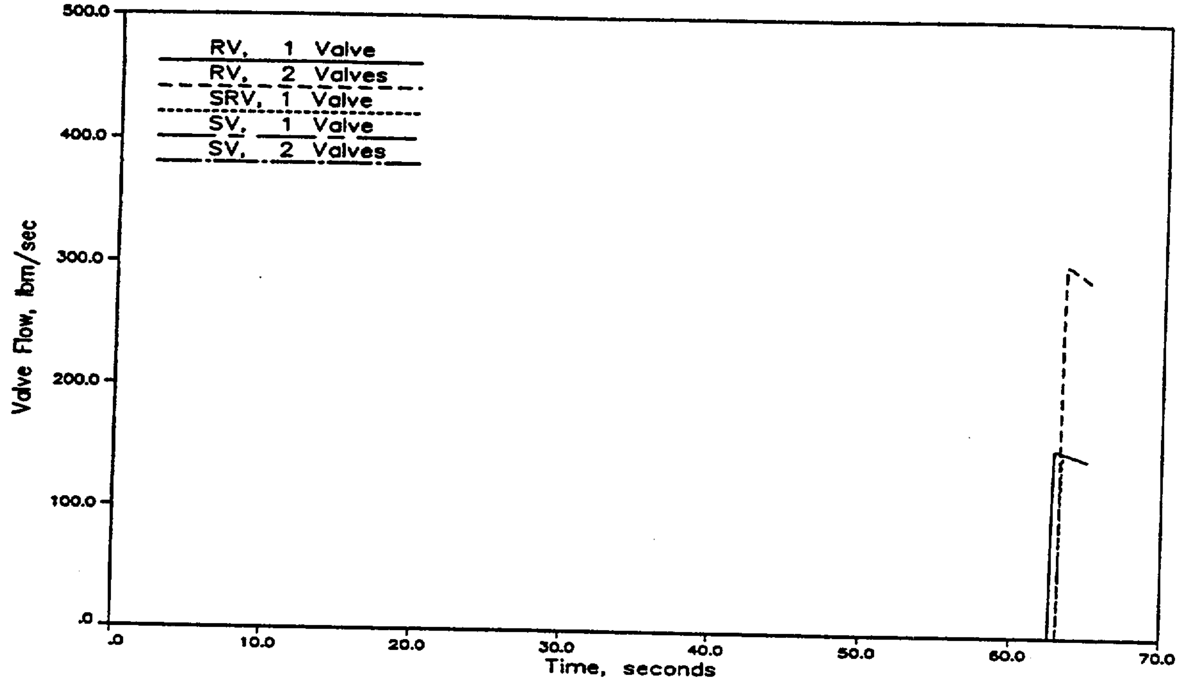


Figure 4.7 Feedwater Controller Failure at 100/108 - Vessel Pressure Response



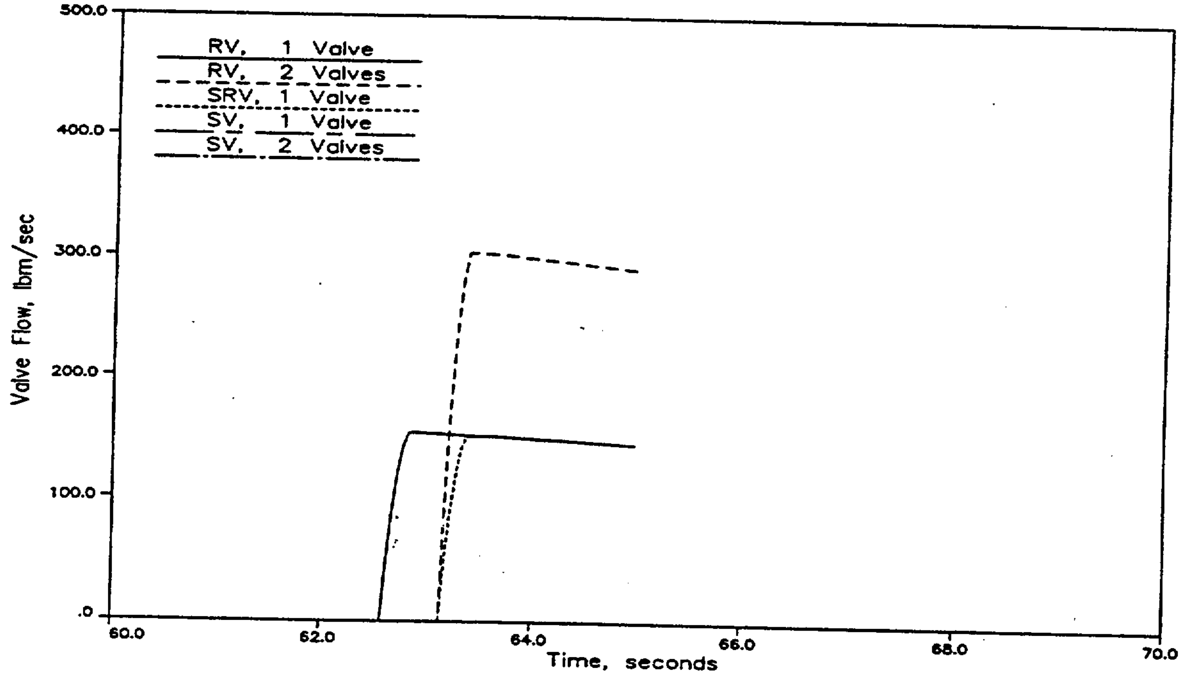


Figure 4.8 Feedwater Controller Failure at 100/108 - Safety/Relief Valves

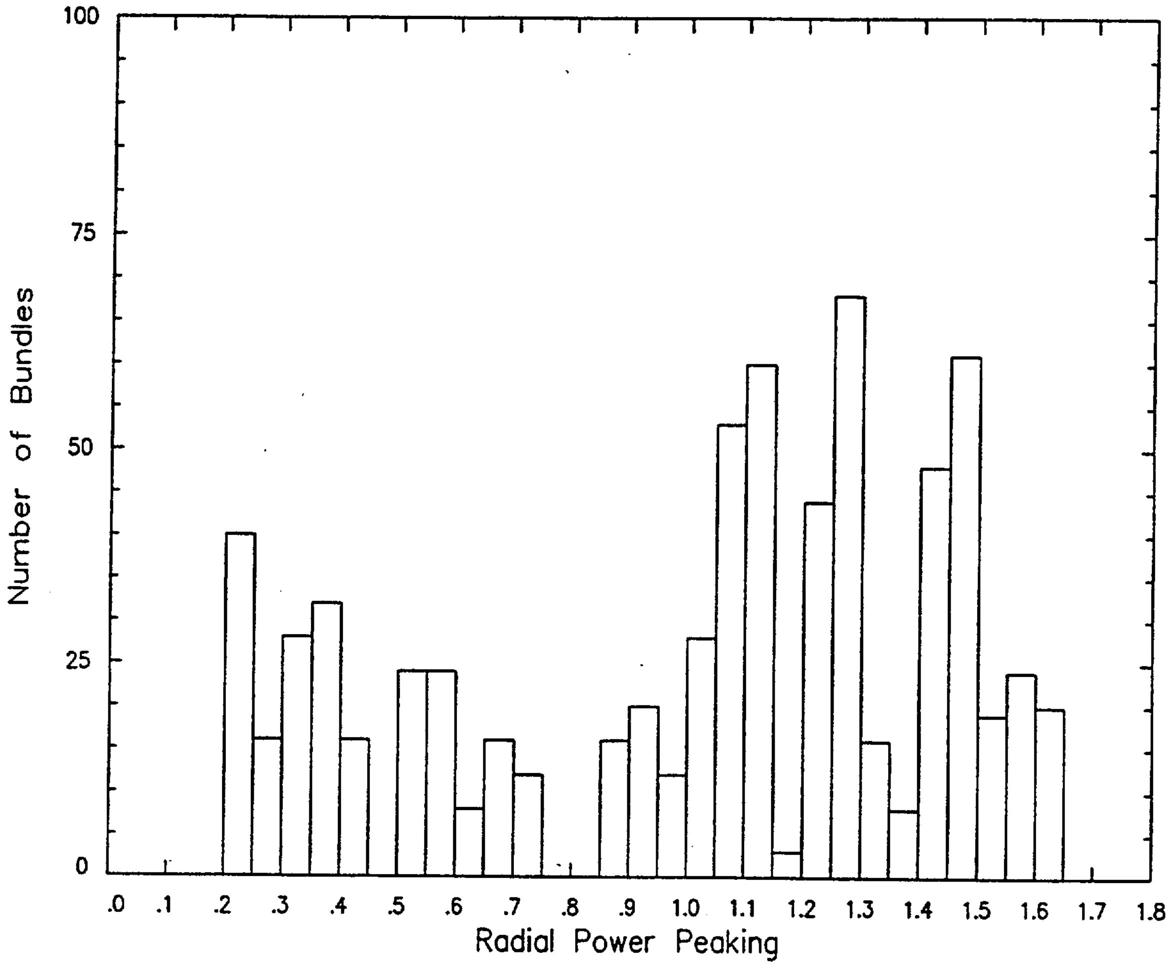


Figure 4.9 Radial Power Distribution for SLMCPR Determination

Control Rod Corner

Control Rod Corner

1.009	1.026	1.044	1.051	1.108	1.096	1.095	0.992	0.977
1.026	0.987	1.060	0.978	1.075	1.056	0.930	0.994	0.970
1.044	1.060	0.978	1.126	1.117	1.076	1.020	0.876	1.049
1.051	0.978	1.126	Internal Water Channel			1.034	0.962	1.005
1.108	1.075	1.117				1.069	0.828	0.982
1.096	1.056	1.076				1.026	0.918	0.963
1.095	0.930	1.020	1.034	1.069	1.026	0.821	0.909	0.960
0.992	0.994	0.876	0.962	0.828	0.918	0.909	0.810	0.915
0.977	0.970	1.049	1.005	0.982	0.963	0.960	0.915	0.822

Figure 4.10 Quad Cities Unit 1 Cycle 17
Safety Limit Local Peaking Factors With Channel Bow at Assembly Exposure of 25,000 MWd/MTU (SPCA9-383B-11GZH-ADV)

Control Rod Corner

Control Rod Corner

1.011	1.029	1.047	1.047	1.105	1.095	1.096	0.993	0.979
1.029	0.994	1.007	1.026	1.003	1.055	0.931	0.996	0.972
1.047	1.007	1.100	1.110	1.113	1.075	1.021	0.877	1.051
1.047	1.026	1.110	Internal Water Channel			1.035	0.964	1.006
1.105	1.003	1.113				1.070	0.829	0.983
1.095	1.055	1.075				1.027	0.920	0.964
1.096	0.931	1.021	1.035	1.070	1.027	0.822	0.911	0.961
0.993	0.996	0.877	0.964	0.829	0.920	0.911	0.811	0.917
0.979	0.972	1.051	1.006	0.983	0.964	0.961	0.917	0.823

Figure 4.11 Quad Cities Unit 1 Cycle 17
Safety Limit Local Peaking Factors With Channel Bow at Assembly Exposure of 25,000 MWd/MTU (SPCA9-382B-12GZL-ADV)

5.0 Maximum Overpressurization Analysis

This section describes the analysis of the maximum overpressurization event performed with COTRANSA2 (Reference 5) in compliance with the ASME code (ASME Boiler and Pressure Vessel Code Section III).

5.1 *Design Basis*

Rated reactor conditions for maximum overpressurization transients are summarized in Table 4.1. No credit was assumed for the operation of the 4 power-actuated relief valves as required by the ASME code. Additional conservatism was included in the analysis by assuming that the SRV (both relief and safety function) and five safety valves with the lowest set points were inoperable (only 3 safety valves were assumed to be operable). The ATWS RPT trip was modeled at 1250 psig. Failure of the most critical active component was assumed. In this instance, the most critical active component is the direct scram on valve position. A combined TCV/TSV closure event was also analyzed to verify that the closure of all MSIVs is the bounding pressurization event. Analysis assumptions provided in Reference 12 for the TCV closure and TSV closure maximum pressurization evaluations specify closure of both the TSV and TCV. Since, direct scram on valve position is not credited, the two events are identical and separate TCV closure and TSV closure maximum pressurization evaluations are no longer required.

Of the EOD/EOOS operating conditions described in Table 2.4, maximum pressurization analyses are performed with only coastdown and combined ICF/coastdown conditions. As demonstrated in References 13, 20, and 21, all other base case* and EOD/EOOS conditions are nonlimiting for maximum pressurization events. Coastdown conditions are further described in Section 7.2.

5.2 *Pressurization Transients*

The position scram, which initiates reactor shutdown almost immediately upon MSIV movement, mitigates the effects of this event to the point that it does not contribute to the determination of pressure margins. Delaying the scram until the high flux trip set point is reached results in a substantially more severe transient.

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* Base case conditions are nonlimiting with respect to coastdown and ICF/coastdown conditions. Therefore, maximum overpressurization analyses are not performed for base case conditions.

Although the closure rate of the MSIVs is substantially slower than that of the TCVs or TSVs, the compressibility of the fluid in the steam lines provides significant damping of the compression wave associated with the TCV/TSV closure event to the point that the slower MSIV closure without direct scram results in nearly as severe a compression wave.

Once the MSIVs are closed, the subsequent core power production must be contained within a smaller system volume than that associated with the TCV/TSV closure event. Table 5.1 provides analysis results for the maximum overpressurization events analyzed for Cycle 17. Cycle 17 analyses demonstrate that the MSIV closure event under these conservative assumptions results in a higher overpressure than the TCV/TSV closure event.

5.3 *Closure of All Main Steam Isolation Valves*

This calculation assumed that all four steam lines were isolated at the containment boundary within 3 seconds. The valve characteristics and steam compressibility combine to delay the arrival of the compression wave at the core until approximately 3 seconds from the initiation of the MSIV stroke. Effective shutdown is delayed until approximately 5 seconds following initiation of the MSIV stroke because control rod performance is assumed to be at the Technical Specification limits. Only TSSS insertion times were used in the analyses.

The limiting MSIV closure (highest vessel pressure) occurred at 100% power/100% flow at EOFP+1500 MWd/MTU (coastdown). The maximum vessel pressure (at the lower plenum) of 1361 psig was observed at 7.1 seconds. The maximum steam line pressure of 1335 psig and the maximum steam dome pressure of 1335 psig were observed at 7.5 seconds. The relative values of maximum pressure during the MSIV closure transient indicate that the vessel and steam lines will be protected against overpressure limits defined in the ASME code when a pressure safety limit of 1375 psig in the lower plenum is protected. In addition, based on results provided in Table 5.1, the Quad Cities Technical Specification steam dome pressure limit of 1345 psig (Reference 27) is also protected.

Figures 5.1–5.4 illustrate the performance of major system variables during the MSIV closure overpressurization event at 100% power and 100% flow at coastdown.

Maximum pressurization analysis results confirm that the limiting MSIV closure transient has approximately 14 psi margin to the vessel pressure limit and 10 psi margin to the steam dome pressure limit.

Table 5.1 Quad Cities Unit 1 Cycle 17 Results Summary of Maximum Overpressurization Analyses With TSSS Insertion Times

Transient	Maximum Pressurization (psig)	
	Steam Dome	Lower Plenum
MSIV Closure - Coastdown		
100% power / 108% flow	1332	1360
100% power / 100% flow	1335	1361
TCV/TSV Closure - Coastdown		
100% power / 108% flow	1327	1355
100% power / 100% flow	1329	1356

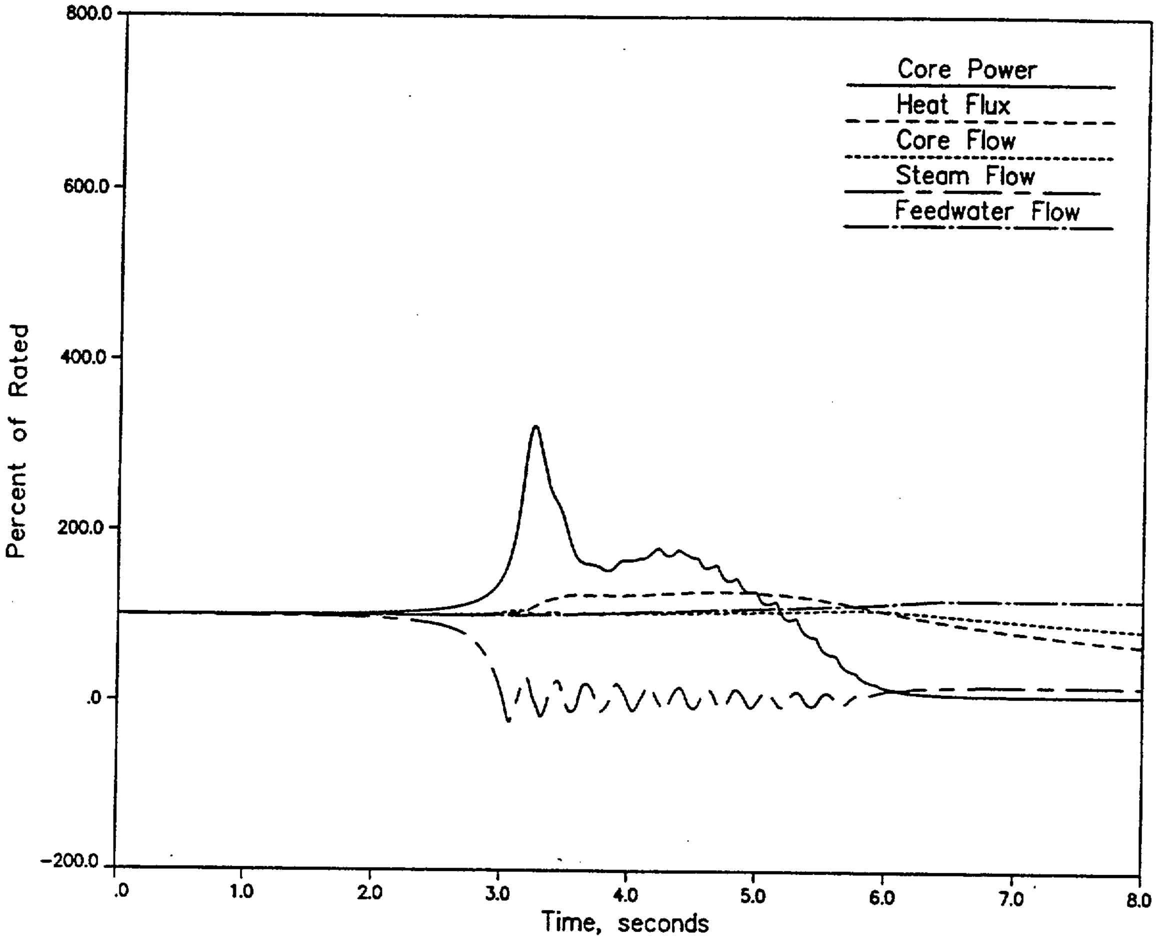

Figure 5.1 MSIV Closure at 100/100 Coastdown - Key Parameters

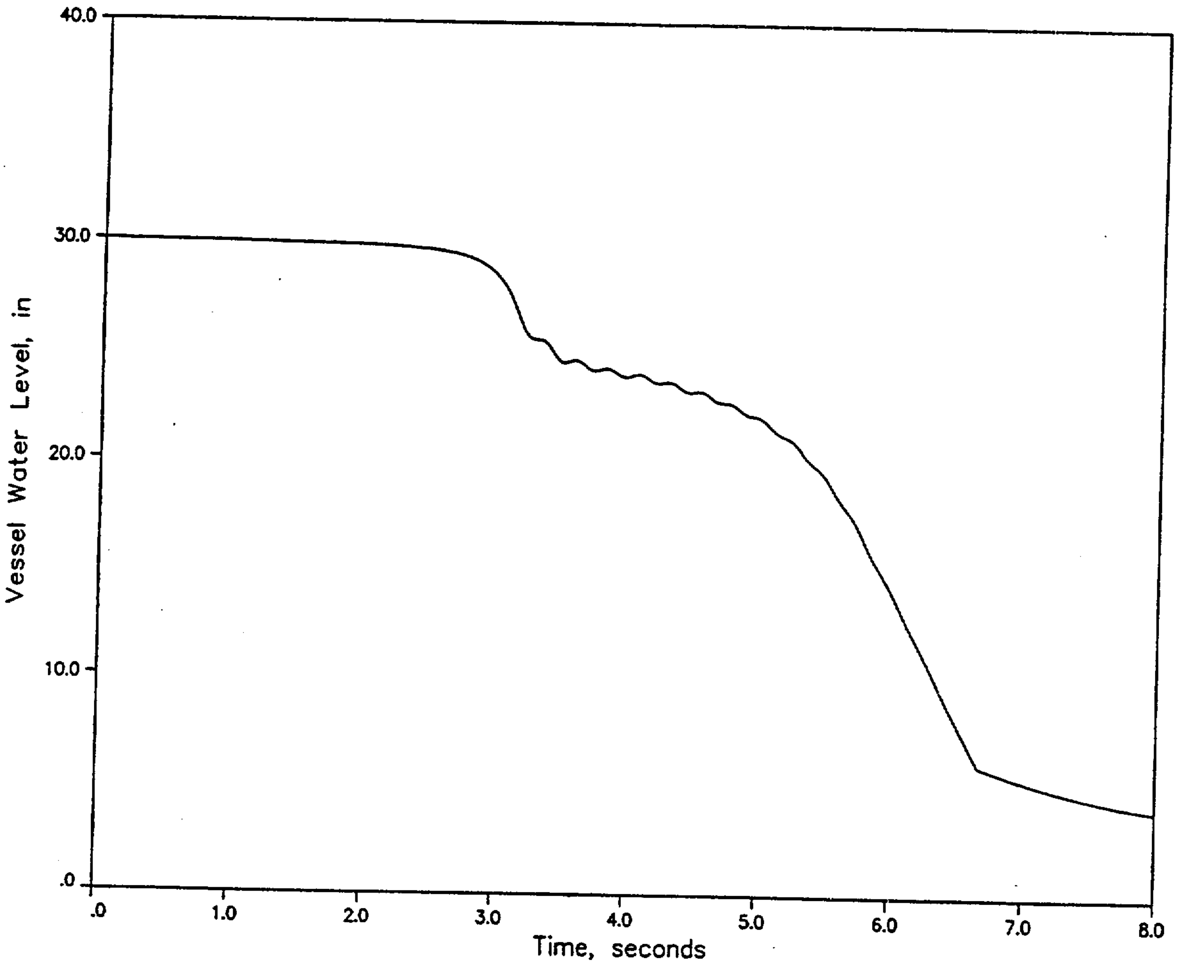

Figure 5.2 MSIV Closure at 100/100 Coastdown - Vessel Water Level
(Referenced to Instrument Zero)

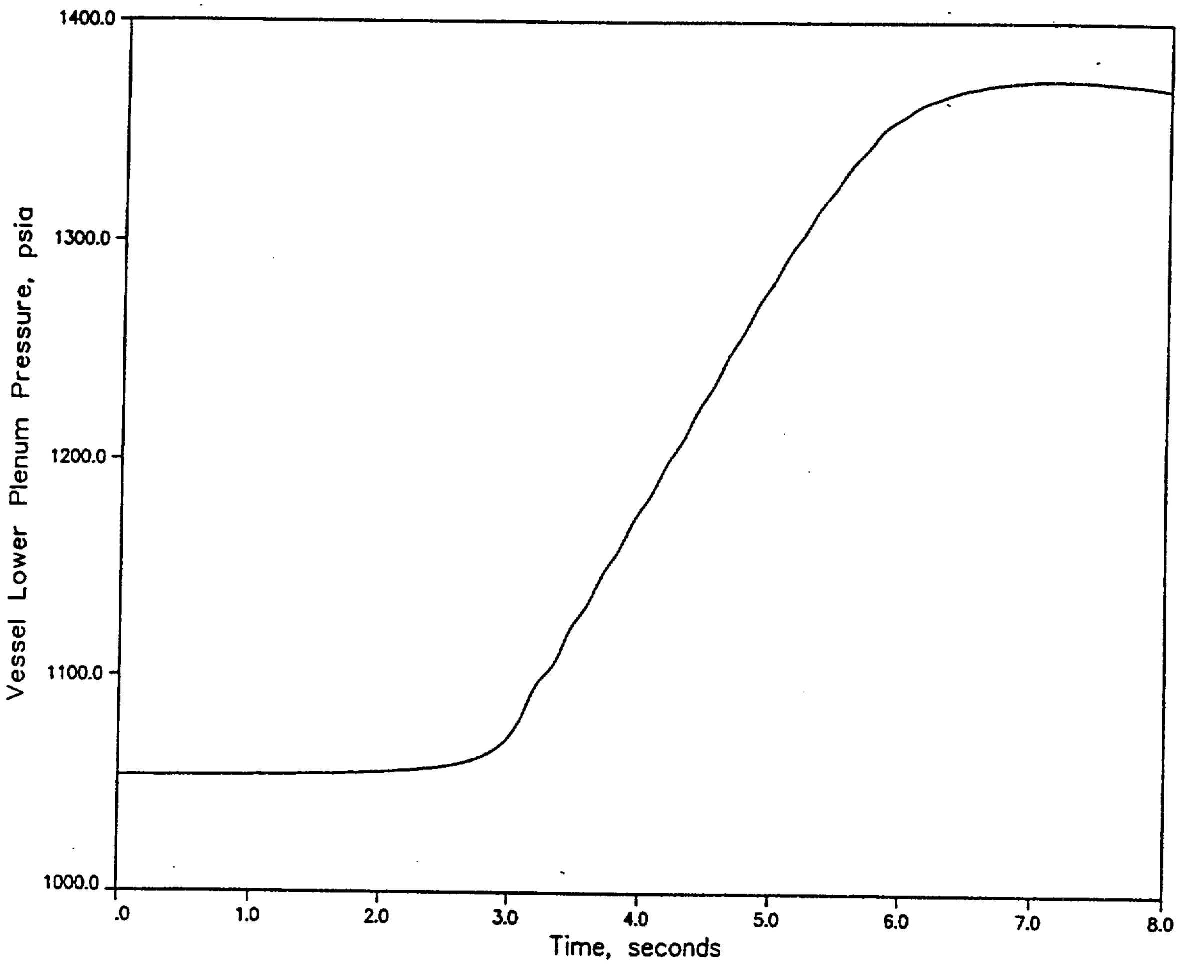

Figure 5.3 MSIV Closure at 100/100 Coastdown - Vessel Pressure Response

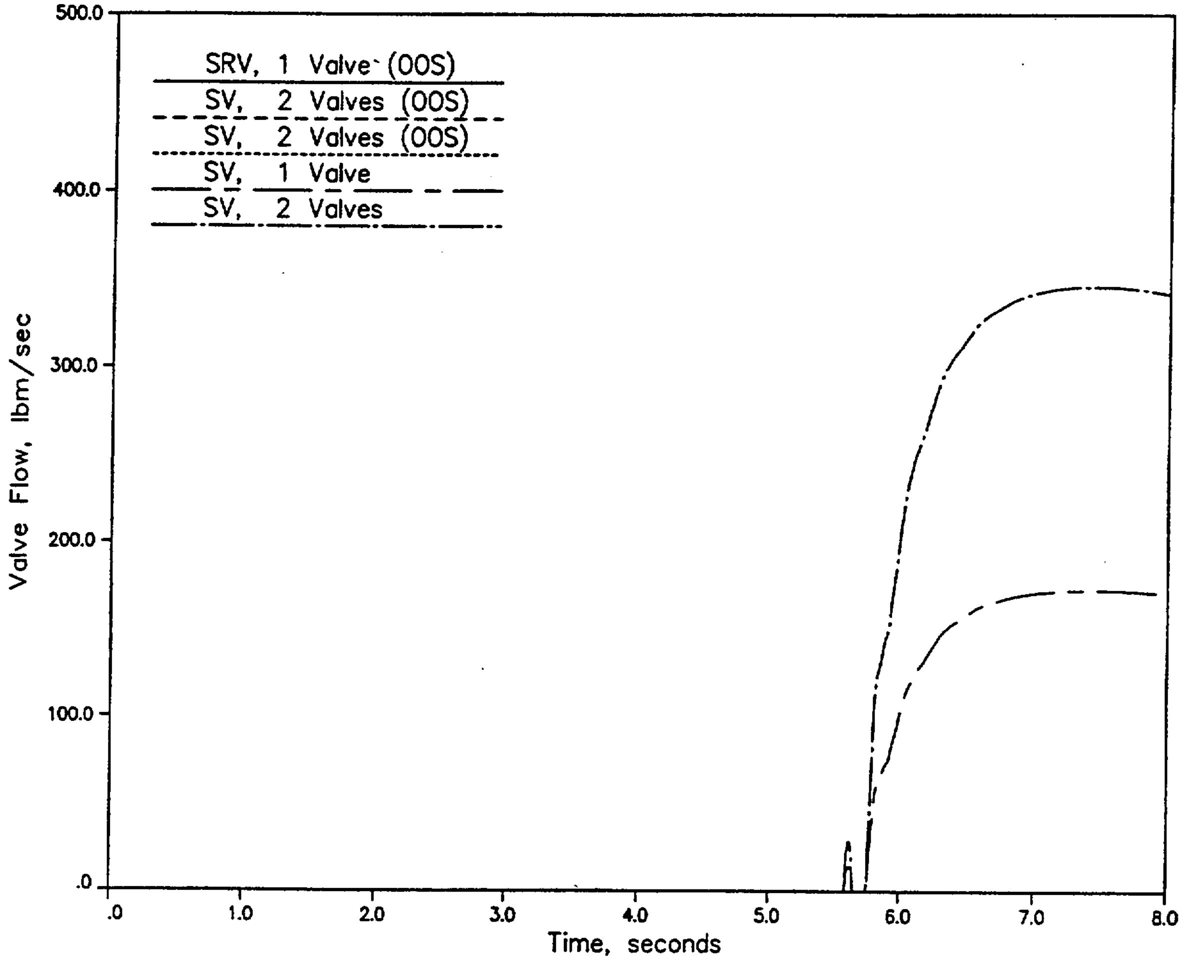

Figure 5.4 MSIV Closure at 100/100 Coastdown - Safety/Relief Valve Flows

6.0 Analysis at Off-Rated Conditions

Transient analysis of a BWR requires consideration of transients at off-rated conditions. This section describes those evaluations performed in support of Cycle 17 that are not covered in Sections 4.0 and 5.0. This section specifically addresses reduced core power and core flow. EOD/EOOS conditions are discussed in Section 7.0.

6.1 *Reduced Core Power*

The base case cycle-specific MCPR operating limits were determined using analyses performed at full power and at end-of-cycle (EOC) exposure with all control rods fully withdrawn. Off-rated analyses are not used in setting the OLMCPR limit because there is sufficient MCPR margin at off-rated conditions to ensure that the SLMCPR is not violated. The full power analysis will bound analyses at off-rated conditions. At exposures earlier in the cycle, the core could potentially be at the OLMCPR at reduced power using control rods; however, the partially inserted control rods would result in a substantial increase in scram reactivity worth and in a ΔCPR less than the full power analysis.

Transient analyses were performed with reduced power in References 13 and 28. The results of References 13 and 28 demonstrate that full power transients bound events at reduced power because of the increased margin to thermal limits. The gain in steady-state MCPR margin (the difference between the steady-state MCPR of the off-rated power case and the steady-state MCPR of the limiting full power ΔCPR case) is much greater than the increase, if any, in ΔCPR. Since changes in core configuration will not change reduced power transient trends and the power/flow map is unchanged, the conclusions of References 13 and 28 that full power transients bound events at reduced power are applicable for QC1C17.

6.2 *Reduced Core Flow*

Thermal margin results for the equilibrium ATRIUM-9B offset core (Reference 13), the initial ATRIUM-9B offset reload core for Quad Cities Unit 2 Cycle 15 (Reference 20) and the initial ATRIUM-9B offset reload core for Quad Cities Unit 1 Cycle 16 (Reference 21) provide sufficient evidence that the 100% power/87% flow state point is nonlimiting for all possible operating conditions including standard operation and all EOD/EOOS combinations. Therefore, as indicated in Reference 2, thermal margin evaluations are not performed at 100% power/87% flow conditions. Reference 13 further demonstrates that off-rated core power and core flow transients were bound by rated power transients. Since changes in core configuration will not change reduced power/flow transient trends and the power/flow map is unchanged, the conclusions of Reference 13 that full power transients bound events at reduced power and flow are applicable for QC1C17.

Limiting conditions for maximum pressurization transients occur at coastdown (EOFP+1500 MWd/MTU) where reduced core flow operation is not possible. EOFP analyses at 100% power/87% flow are bound by EOFP + 1500 MWd/MTU analyses at 100% power/100% flow and 100% power/108% flow.

Analysis for pump run-up events from operation at less than rated recirculation pump capacity indicates the need for an augmentation of the full flow OLMCPR for lower flow conditions. This is due to the potential for large reactor power increases should an uncontrolled pump flow increase occur.

The analysis establishes the reduced flow MCPR operating limits ($MCPR_f$) necessary to protect the reactor fuel against boiling transition during anticipated pump run-up events from off-rated core flow conditions for manual flow control (MFC). The analysis also establishes $MCPR_f$ limits to protect the OLMCPR for automatic flow control (AFC). The Quad Cities flow run-up analyses use steep run-up paths that bound GE10 and ATRIUM-9B offset equilibrium cores as well as transition cores from GE10 to ATRIUM-9B offset. Analyses are performed using XCOBRA (Reference 6) to calculate the change in critical power along a conservative flow run-up path from 37%power/30% flow to 100% power/108% flow for AFC (Table 6.1). The flow run-up path for MFC begins at 48% power/30% flow and ends at 125% power/110% flow (Table 6.2). Flow-dependent MCPR results for GE10 and ATRIUM-9B offset fuel are provided in Tables 6.3 and 6.4. Linear extrapolation of the 40% and 30% core flow XCOBRA analysis results is used to obtain MCPR limits below 30% of rated core flow.

$MCPR_f$ limits are shown in Figures 6.1- 6.3 for the limiting fuel in Quad Cities Unit 1 Cycle 17 for the automatic flow control event. Figure 6.4 details $MCPR_f$ limits pertaining to the manual flow control event for the limiting fuel in Quad Cities Unit 1 Cycle 17. The analysis results provide for operation up to EOFP and operation with EOD/EOOS. The cycle-specific MCPR limit for Quad Cities Unit 1 shall be the maximum of the $MCPR_f$ limit depicted in these tables for the appropriate control mode and the full flow cycle-specific OLMCPR. It is conservative to use the TLO $MCPR_f$ limit or full flow OLMCPR plus 0.01 (whichever is greater) for SLO. This method is applied for operation up to EOFP and for EOD/EOOS conditions. These limits conservatively bound all transients from single-loop conditions. The $MCPR_f$ limit protects against boiling transition during flow excursions to maximum two-pump flow; excursions to such high flows are not possible during single-loop one-pump operation. Thus, conservatively maintaining this two-loop limit assures that there is even more thermal margin under single-loop conditions than under two-loop full power/full flow conditions.

Automatic flow control analyses were performed to support OLMCPRs for base case operation, base case operation with all BPVOOS and for EOD/EOOS operation (refer to Table 2.2). AFC $MCPR_f$ limits for the 1 BPVOOS OLMCPRs can be determined from the appropriate base case and EOD/EOOS $MCPR_f$ limits.

The $MCPR_f$ penalty described in Reference 18 has been applied to the GE10 $MCPR_f$ limits shown in Figures 6.1–6.4. The penalty is a function of core flow with a value of 0.0 at 100% rated and increases linearly to 0.05 at 40% rated. The penalty is linearly extrapolated for flows less than 40% of rated. GE10 flow-dependent MCPR results provided in Table 6.3, with the addition of the penalty, are bound by the $MCPR_f$ limits of Figures 6.1–6.4.

6.2.1 Automatic Flow Control

If the reactor is operated in the AFC mode, variations in core power should not result in CPRs less than the established OLMCPR for rated conditions. If the rated condition MCPR limit is observed in a reduced flow condition, a subsequent increase in power to full power along the AFC control line may result in inadvertent degradation of fuel CPRs below this reference (full flow) OLMCPR limit. The probability of boiling transition conditions occurring during a subsequent anticipated event may increase beyond acceptable levels if this were the case.

SPC has determined the required $MCPR_f$ limit for off-rated conditions to prevent the MCPR from degrading below the cycle full power OLMCPR limit during AFC operation. This was determined by evaluating the MCPR for a given reactor power distribution at varying total reactor power and flow conditions. The variations in total core power and flow were assumed to follow the expected relationship for AFC operation (Table 6.1). The power distribution chosen was such that MCPR equaled the referenced OLMCPR at 100% rated power and 108% rated flow. The expected variation of core pressure and inlet coolant subcooling with reactor power level was also considered.

Reduced flow MCPR limits for AFC are presented in Figures 6.1 – 6.3 for the Cycle 17 fuel types. The $MCPR_f$ limits provide the required protection during AFC operation for operation up to EOFP* and operation with EOD/EOOS.

6.2.2 Manual Flow Control

This section discusses pump excursions when the plant is in MFC, i.e., not in AFC operation mode. Because the power/flow increase due to a single-pump excursion is bound by that of a two-pump excursion, only a two-pump excursion is evaluated for Cycle 17. The analysis of the two-pump flow excursion indicates that the limiting event scenario is a gradual quasi-steady run-up. These results indicate that MCPR would decrease below the SLMCPR if the full flow reference MCPR was observed at initial conditions. Thus, an augmented MCPR limit is needed for partial flow operation to protect the two-pump excursion event. The manual flow control $MCPR_f$ limits are not affected by operation at reduced steam dome pressure (Reference 19).

The power/flow path used for the run-up is shown in Table 6.2 and bounds that calculated for constant xenon.

The results of the two-pump run-up analyses for manual flow control are presented in Figure 6.4 for the Cycle 17 fuel types. When in manual flow control, the cycle-specific MCPR limit for Quad Cities Unit 1 shall be the maximum of the MFC $MCPR_f$ limit or the OLMCPR. The $MCPR_f$ limits provide the required protection for operation up to EOFP and operation with EOD/EOOS.

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* AFC $MCPR_f$ limits are provided for base case operation up to EOFP with all BPVOOS. AFC $MCPR_f$ limits for base case operation with 1 BPVOOS can be determined from the appropriate base case and EOD/EOOS $MCPR_f$ limits.

Table 6.1 Automatic Flow Control Excursion Path

Total Core Flow (% of rated)	Power (% of rated)
108	100
100	94
90	86
80	78
70	69
60	61
50	53
40	45
30	37

Table 6.2 Manual Flow Control Excursion Path

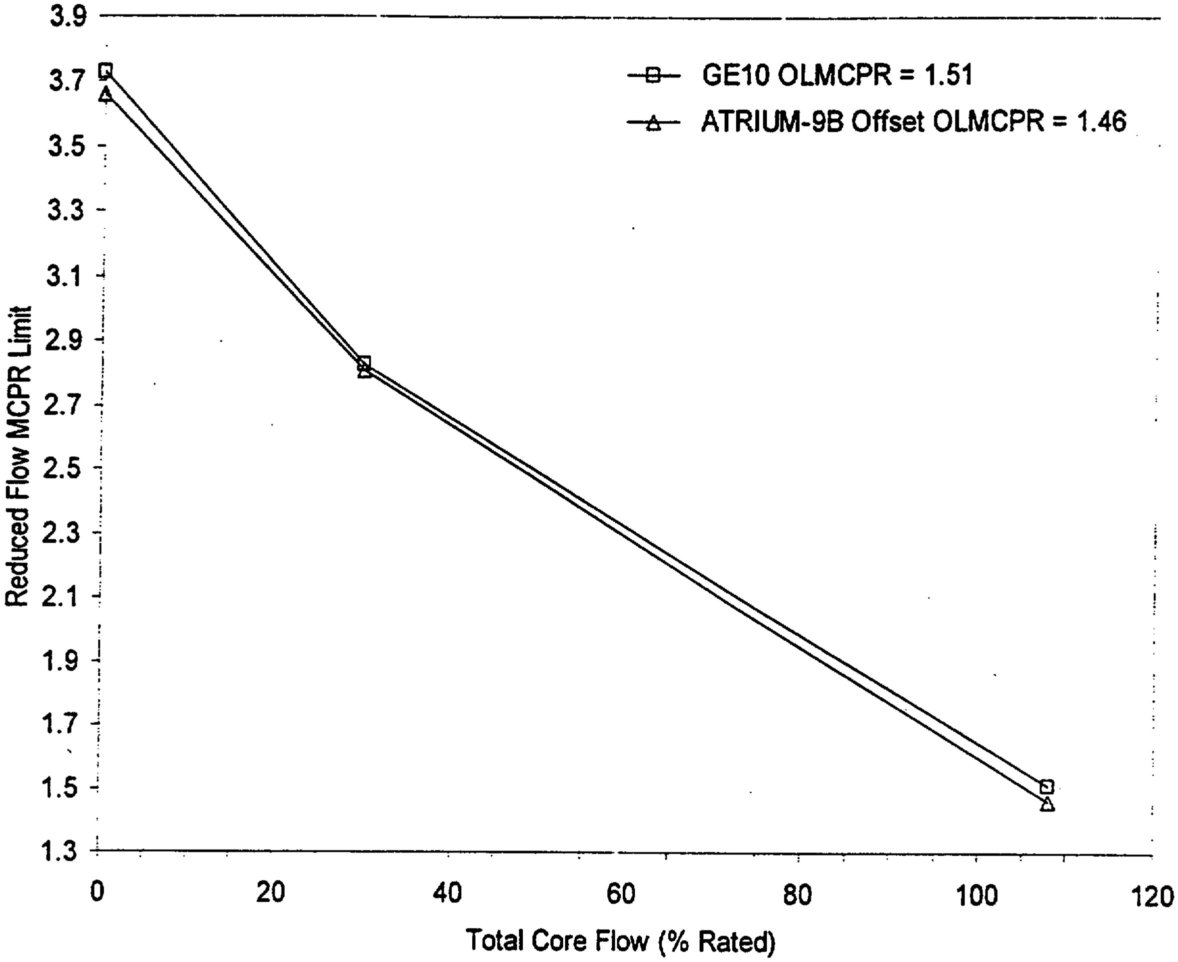
Total Core Flow (% of rated)	Power (% of rated)
110	125
100	115
90	106
80	96
70	87
60	77
50	68
40	58
30	48

Table 6.3 Flow-Dependent MCPR Results GE10 Fuel (Penalty Not Included)

Total Core Flow (% of rated)	Manual Flow Control	Automatic Flow Control MCPR		
		1.51 OLMCPR	1.55 OLMCPR	1.56 OLMCPR
110	1.110	—	—	—
108	—	1.510	1.550	1.560
100	1.192	1.604	1.647	1.658
90	1.276	1.723	1.769	1.781
80	1.366	1.852	1.903	1.916
70	1.455	1.990	2.045	2.059
60	1.546	2.134	2.195	2.210
50	1.645	2.292	2.357	2.373
40	1.760	2.476	2.545	2.563
30	1.936	2.766	2.841	2.860

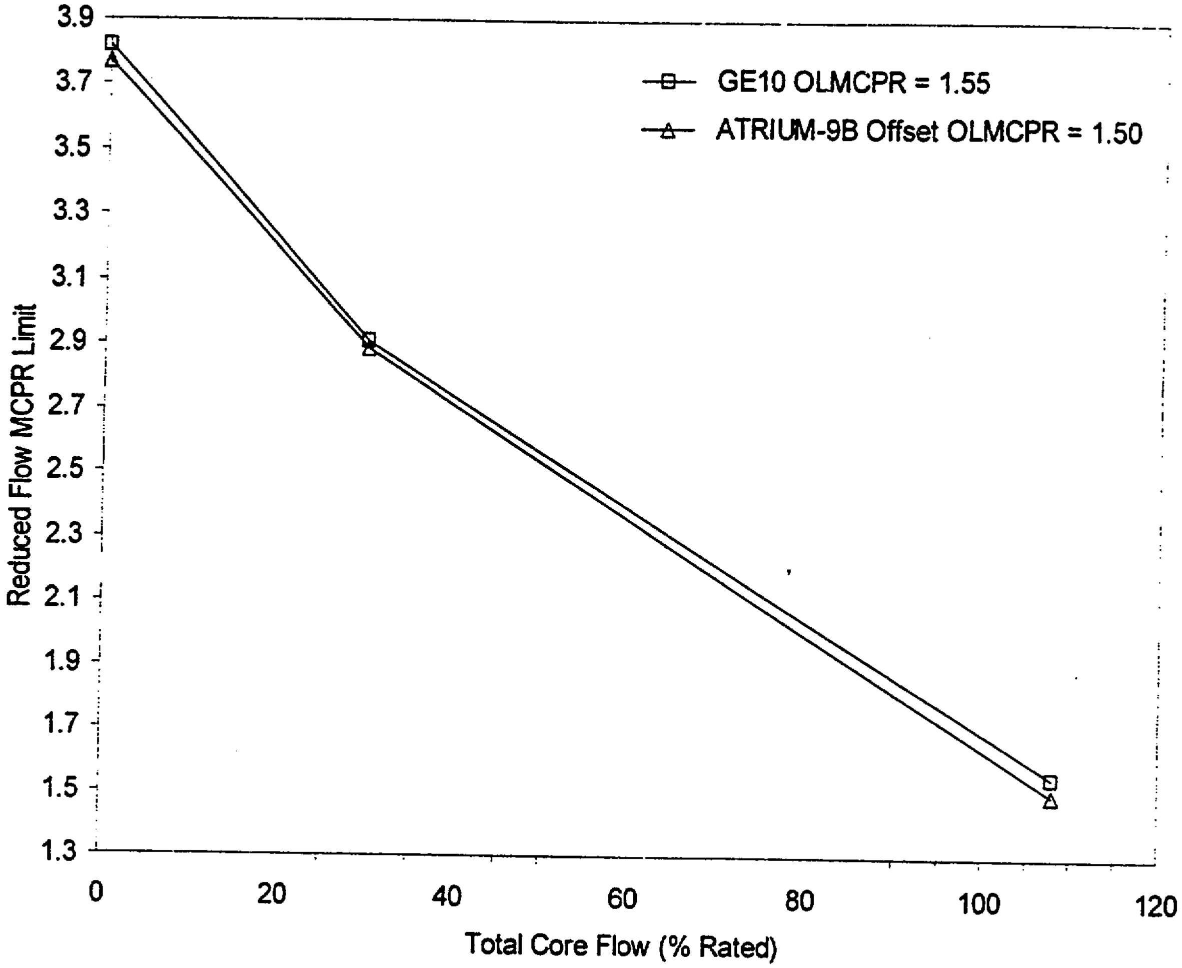
Table 6.4 Flow-Dependent MCPR Results ATRIUM-9B Offset Fuel

Total Core Flow (% of rated)	Manual Flow Control	Automatic Flow Control MCPR		
		1.46 OLMCPR	1.50 OLMCPR	1.51 OLMCPR
110	1.110	—	—	—
108	—	1.460	1.500	1.510
100	1.199	1.555	1.598	1.608
90	1.300	1.687	1.733	1.744
80	1.408	1.832	1.883	1.896
70	1.517	1.987	2.043	2.057
60	1.627	2.150	2.212	2.228
50	1.739	2.322	2.390	2.407
40	1.864	2.515	2.588	2.606
30	2.043	2.801	2.881	2.901



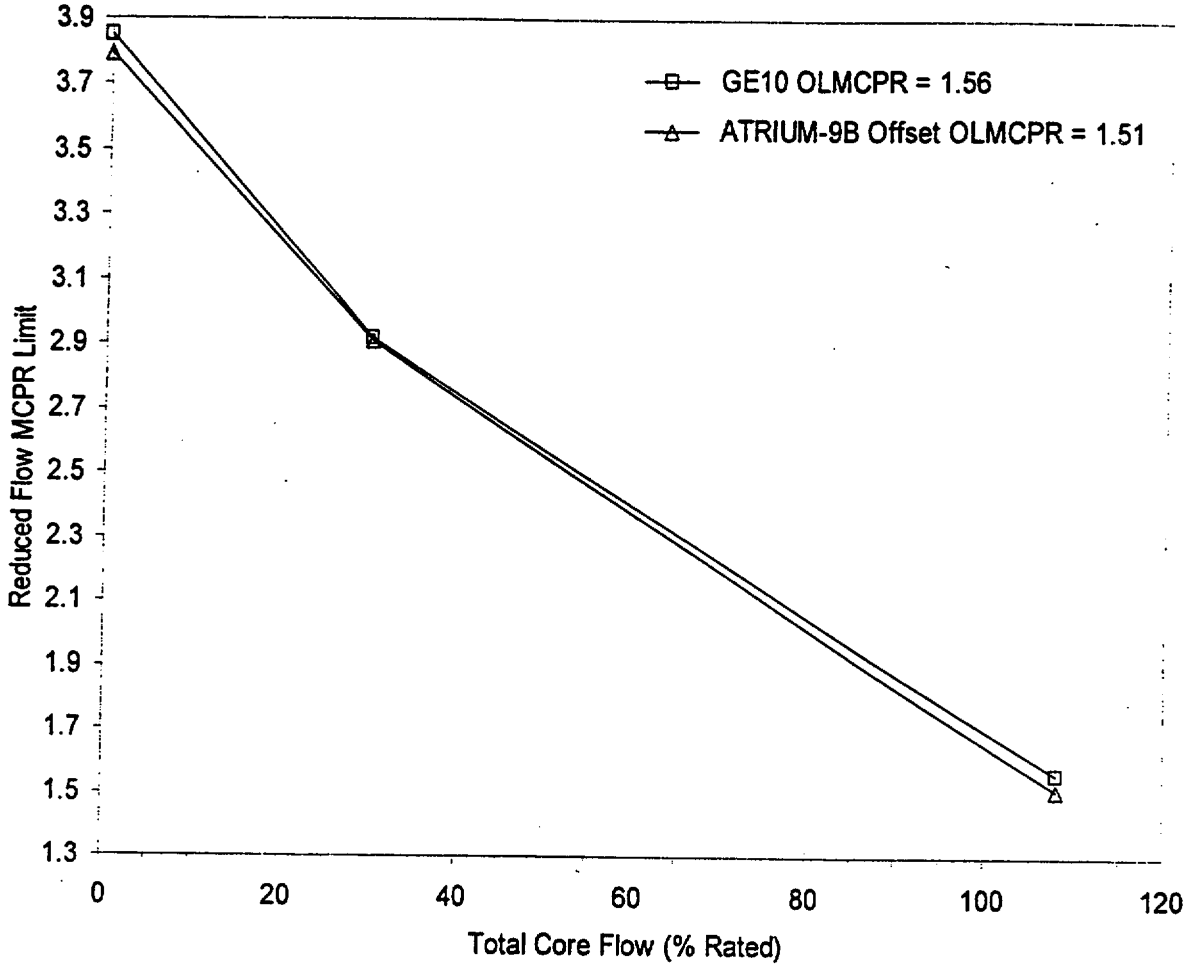
Total Core Flow (% of rated)	GE10 $MCPR_f$ Limit for OLMCPR=1.51	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR=1.46
108	1.51	1.46
30	2.83	2.81
0	3.73	3.66

Figure 6.1 Reduced Flow MCPR Limit for Automatic Flow Control (Base Case OLMCPR)



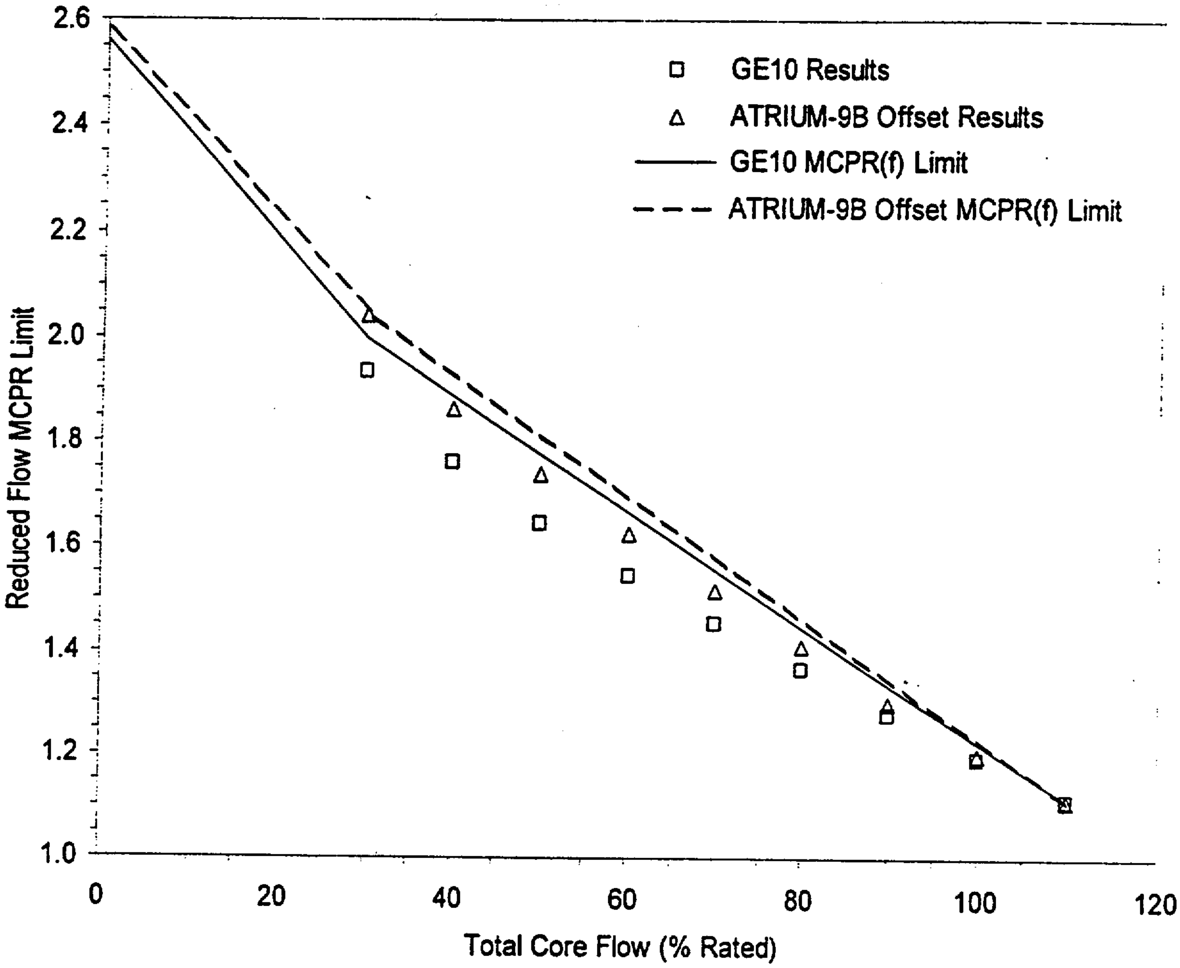
Total Core Flow (% of Rated)	GE10 MCPR$_f$ Limit for OLMCPR=1.55	ATRIUM-9B Offset MCPR$_f$ Limit for OLMCPR=1.50
108	1.55	1.50
30	2.91	2.89
0	3.82	3.77

Figure 6.2 Reduced Flow MCPR Limit for Automatic Flow Control (EOD/EOOS OLMCPR)



Total Core Flow (% of rated)	GE10 MCPR$_f$ Limit for OLMCPR=1.56	ATRIUM-9B Offset MCPR$_f$ Limit for OLMCPR=1.51
108	1.56	1.51
30	2.92	2.91
0	3.85	3.79

Figure 6.3 Reduced Flow MCPR Limit for Automatic Flow Control (All BPVOOS OLMCPR)



Total Core Flow (% of rated)	GE10 MCPR Limit	ATRIUM-9B Offset MCPR Limit
110	1.11	1.11
30	2.00	2.05
0	2.56	2.59

Figure 6.4 Reduced Flow MCPR Limit for Manual Flow Control (SLMCPR=1.11)

7.0 Evaluation of EOD/EOOS Conditions

Reference 13 provides a discussion of operation with EOD/EOOS at Quad Cities and also provides generic penalties* for EOD/EOOS operation. The specific EOD/EOOS conditions supported for Quad Cities are identified in Table 2.4. Additional analyses were performed for Cycle 17 to determine specific EOOS OLMCPR penalties for operation up to EOFP with: (a) 1 bypass valve out-of-service (BPVOOS), and (b) all bypass valves out-of-service. The limiting EOFP FWCF event at 100% power/108% flow was analyzed with all BPVOOS and with parameters specified in Reference 12 for 1 BPVOOS.

Transient analysis results provided in References 20 and 21 demonstrate that the generic OLMCPR penalty described in Reference 13 cannot be confirmed for cycle-specific applications. Therefore, thermal margin analyses were performed with the EOD/EOOS conditions identified in Table 2.4 to develop cycle-specific OLMCPR penalties for QC1C17. Of the EOD/EOOS operating conditions described in Table 2.4, maximum pressurization evaluations are performed with only coastdown and combined ICF/coastdown conditions. All other base case and EOD/EOOS conditions are nonlimiting for maximum pressurization events. Maximum overpressurization analysis results for the limiting EOD/EOOS conditions are provided in Section 5.0, no EOD/EOOS pressure penalty is required for Cycle 17.

The Cycle 17 OLMCPR penalty for operation with FFTR, FHOOS, coastdown, or any combination thereof is 0.04 for GE10 and ATRIUM-9B offset fuel. Other EOD/EOOS conditions require no OLMCPR penalty. The Cycle 17 OLMCPR penalties for base case operation up to EOFP with 1 BPVOOS and all BPVOOS are 0.01 and 0.05, respectively. OLMCPR penalties are determined by comparing all EOD/EOOS and BPVOOS state points to the limiting base case state point at EOFP (100% power/108% flow). Maximum pressurization analysis results provided in Section 5.0 confirm that the limiting EOD/EOOS MSIV closure transient (100% power/100% flow coastdown) has approximately 14 psi margin to the vessel pressure limit and 10 psi margin to the steam dome pressure limit.

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* The generic OLMCPR penalty provided in Reference 13 and the cycle-specific OLMCPR penalty for QC1C17 are both 0.04. The cycle-specific OLMCPR penalty of 0.04 is required to support operation with FFTR, FHOOS, coastdown or any combination thereof, ~~is only applied when~~ core exposure ~~is~~ greater than the licensing basis core exposure at EOC17 shown in Section 4.2.1 of Reference 1. Other EOD/EOOS conditions listed in Table 2.4 require no OLMCPR penalty. The 5 psi pressure penalty provided in Reference 13 is not required for QC1C17.

Coastdown is defined as
per SPC
9/20/00

7.1 *Final Feedwater Temperature Reduction*

Final feedwater temperature reduction (FFTR) at the end of cycle can be used to extend full power operation of the cycle. Analyses were performed for a 100°F reduction in feedwater temperature.* Results for FFTR operation are presented in Table 7.1.

7.2 *Coastdown*

Coastdown operation occurs after EOFP where a gradual reduction in core power occurs as the fuel depletes. Coastdown analyses assume an additional 1500 MWd/MTU full power exposure step after EOFP to provide for operation at 15% of rated power above the equilibrium xenon coastdown power level. It is the 1500 MWd/MTU exposure extension from EOFP that forces the need to establish the coastdown penalties. As explained in Reference 13, after EOFP+1500 MWd/MTU the core power is conservatively assumed to decrease at a rate of 10% in rated power per 1000 MWd/MTU increase in exposure. Analyses at EOFP+1500 MWd/MTU bound coastdown at higher exposures. Results for coastdown are presented in Table 7.2. For the coastdown conditions analyzed, the 100% power/87% flow state point is unattainable.

7.3 *Combined Final Feedwater Temperature Reduction/Coastdown*

Results for combined FFTR/coastdown are presented in Table 7.3.

7.4 *Feedwater Heater(s) Out-of-Service*

The feedwater heater out-of-service (FHOOS) scenario assumes a 100°F reduction in the feedwater temperature.* Operation with FHOOS is similar to operation with FFTR except that the reduction in feedwater temperature can occur at any time during the cycle. Results for FHOOS are presented in Table 7.4. The LRNB event is nonlimiting because the reduced feedwater temperature causes a decrease in steam flow.

7.5 *Combined Feedwater Heaters Out-of-Service/Coastdown*

Results for combined FHOOS/coastdown are presented in Table 7.5.

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* The 100°F reduction in feedwater temperature is applicable for all rated and off-rated conditions.

7.6 *Bypass Valve(s) Out-of-Service*

The limiting EOFP FWCF event at 100% power/108% flow was analyzed with all BPVOOS and with parameters specified in Reference 12 for 1 BPVOOS. Analysis results are presented in Table 7.6.

Table 7.1 Quad Cities Unit 1 Cycle 17 Final Feedwater Temperature Reduction MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	(ΔCPR)†	Change in ΔCPR From Limiting Rated Power Case†
LRNB	100 / 108	528	125	1251 / 1217	0.37 / 0.32	-0.03 /-0.03
LRNB	100 / 100	501	125	1251 / 1220	0.36 / 0.29	-0.04 /-0.06
FWCF	100 / 108	535	137	1139 / 1104	0.42 / 0.37	0.02 / 0.02
FWCF	100 / 100	508	137	1136 / 1105	0.40 / 0.36	0.00 / 0.01

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* Lower plenum.

† Values for GE10/ATRIUM-9B offset fuel.

Table 7.2 Quad Cities Unit 1 Cycle 17 Coastdown Operation MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	(ΔCPR)†	Change in ΔCPR From Limiting Rated Power Case†
LRNB	100 / 108	672	132	1309 / 1275	0.41 / 0.39	0.01 / 0.04
LRNB	100 / 100	604	131	1308 / 1277	0.39 / 0.35	-0.01 / 0.00
FWCF	100 / 108	649	136	1195 / 1160	0.41 / 0.37	0.01 / 0.02
FWCF	100 / 100	578	134	1191 / 1159	0.39 / 0.35	-0.01 / 0.00

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* Lower plenum.

† Values for GE10/ATRIUM-9B offset fuel.

Table 7.3 Quad Cities Unit 1 Cycle 17 Combined FFTR/Coastdown MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	(ΔCPR)[†]	Change in ΔCPR From Limiting Rated Power Case[†]
LRNB	100 / 108	573	128	1258 / 1225	0.40 / 0.35	0.00 / 0.00
LRNB	100 / 100	524	127	1258 / 1227	0.38 / 0.31	-0.02 /-0.04
FWCF	100 / 108	569	139	1143 / 1109	0.44 / 0.39	0.04 / 0.04
FWCF	100 / 100	518	137	1140 / 1109	0.42 / 0.37	0.02 / 0.02

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* Lower plenum.

[†] Values for GE10/ATRIUM-9B offset fuel.

Table 7.4 Quad Cities Unit 1 Cycle 17 Feedwater Heater Out-of-Service MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	(ΔCPR)[†]	Change in ΔCPR From Limiting Rated Power Case[†]
FWCF	100 / 108	515	136	1136 / 1101	0.41 / 0.36	0.01 / 0.01
FWCF	100 / 100	492	136	1134 / 1102	0.40 / 0.35	0.00 / 0.00

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* Lower plenum.

[†] Values for GE10/ATRIUM-9B offset fuel.

Table 7.5 Quad Cities Unit 1 Cycle 17 Combined FHOOS/Coastdown MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	(ΔCPR)[†]	Change in ΔCPR From Limiting Rated Power Case[†]
FWCF	100 / 108	556	138	1140 / 1106	0.42 / 0.38	0.02 / 0.03
FWCF	100 / 100	518	137	1138 / 1107	0.41 / 0.37	0.01 / 0.02

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* Lower plenum.

[†] Values for GE10/ATRIUM-9B offset fuel.

Table 7.6 Quad Cities Unit 1 Cycle 17 Bypass Valve(s) Out-of-Service MCPR Results and Comparison to Limiting Rated Power Case

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	(ΔCPR)†	Change in ΔCPR From Limiting Rated Power Case†
FWCF						
1 BPVOOS	100 / 108	635	135	1203 / 1168	0.40 / 0.36	0.00 / 0.01
All BPVOOS	100 / 108	663	136	1308 / 1274	0.41 / 0.40	0.01 / 0.05

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* Lower plenum.

† Values for GE10/ATRIUM-9B offset fuel.

8.0 References

1. EMF-2416 Revision 0, *Quad Cities Unit 1 Cycle 17 Reload Analysis*, Siemens Power Corporation, August 2000.
2. Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Quad Cities Unit 1 Cycle 17 Calculation Plan - Final," DEG:00:045, February 16, 2000.
3. XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description*, Exxon Nuclear Company, January 1987.
4. XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, June 1986.
5. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.
6. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors*, Advanced Nuclear Fuels Corporation, November 1990.
7. ANF-1125(P)(A) and Supplements 1 and 2, *ANFB Critical Power Correlation*, Advanced Nuclear Fuels Corporation, April 1990.
8. Not used.
9. EMF-1125(P)(A) Supplement 1 Appendix C, *ANFB Critical Power Correlation Application for Co-Resident Fuel*, Siemens Power Corporation, August 1997.
10. ANF-1125(P)(A) Supplement 1 Appendix E, *ANFB Critical Power Correlation Determination of ATRIUM™-9B Additive Constant Uncertainties*, Siemens Power Corporation, September 1998.
11. XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, *Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology*, Advanced Nuclear Fuels Corporation, November 1990.
12. EMF-2315 Revision 0, *Quad Cities Unit 1 Cycle 17 Principal Transient Analysis Parameters*, Siemens Power Corporation, February 2000.
13. EMF-96-037(P) Revision 1, *Quad Cities Extended Operating Domain (EOD) and Equipment Out of Service (EOOS) Safety Analysis for ATRIUM™-9B Fuel*, Siemens Power Corporation, September 1996.
14. Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Quad Cities Unit 1 Cycle 17 Transient Power History Data for Confirming Mechanical Limits for GE Fuel," DEG:00:132, June 5, 2000.
15. EMF-97-048 Revision 1, *Quad Cities Unit 1 Cycle 16 Principal Transient Analysis Parameters*, Siemens Power Corporation, August 1999.

8.0 References *(Continued)*

16. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis*, Exxon Nuclear Company, February 1987.

17. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

18. EMF-95-049(P), *Application of the ANFB Critical Power Correlation to Coresident GE Fuel at the Quad Cities and LaSalle Nuclear Power Stations*, Siemens Power Corporation, October 1995.

19. Letter, J. H. Riddle (SPC) to J. E. Nevling (ComEd), "Analysis to Support Operation with Reactor Pressure Less than the Design Pressure at Quad Cities," JHR:97:203, May 27, 1997.

20. EMF-96-180 Revision 4, *Quad Cities Unit 2 Cycle 15 Plant Transient Analysis*, Siemens Power Corporation, August 1999.

21. EMF-98-051 Revision 1, *Quad Cities Unit 1 Cycle 16 Plant Transient Analysis*, Siemens Power Corporation, August 1999.

22. EMF-1903(P) Revision 3, *Impact of Failed/Bypassed LPRMs and TIPs and Extended LPRM Calibration Interval on Radial Bundle Power Uncertainty*, Siemens Power Corporation, March 2000.

23. Letter, R. J. Chin (ComEd) to J. H. Riddle (SPC), "GE10 Channel Bow Data Assumptions for Quad Cities MCPR Safety Limit Calculations," NFS:BND:96-105, September 19, 1996.

24. Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Quad Cities Unit 1 Cycle 17 Safety Limit MCPR Analysis Results - Final," DEG:00:109, April 27, 2000.

25. EMF-2362 Revision 0, *Principal Reload Fuel Design Parameters Quad Cities 1 QCA-2, Cycle 17 (ATRIUM™-9B)*, Siemens Power Corporation, February 2000.

26. Letter, J. H. Riddle (SPC) to R. J. Chin (ComEd), "Disposition of the Feedwater Control System Change in Quad Cities Unit 2 - Final Letter Report," SPCCWE:96:040/ JHR:96:048, February 14, 1996.

27. *Quad Cities Stations 1 and 2 Technical Specifications*, as amended.

28. EMF-2222(P) Revision 1, *Dresden and Quad Cities Evaluation of Changed Analytical Neutron Flux Scram and Safety Valve Set Points*, Siemens Power Corporation, June 2000.

29. Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Extension of LPRM Calibration Interval to 2500 EFPH," DEG:00:112, May 2, 2000.