Appendix A

Margin to Unpiped Safety Valves

SPC performed analyses for Quad Cities Unit 1 Cycle 17 to determine the margin between peak steam line pressure and the lowest set point of the unpiped safety valves. ComEd adopts a limit of 60 psi margin for the main steam isolation valve closure - unpiped safety valve margin (MSIVC-USM) analysis. The load rejection no bypass - unpiped safety valve margin (LRNB-USM) analysis was also performed. At EOFP the limiting initial conditions for steam line pressurization occur at 100% core power and 87% core flow (100%P/87%F). For coastdown conditions of EOFP+1500 MWd/MTU, the state point 100%P/87%F is unattainable; therefore, the limiting state point for EOFP+1500 MWd/MTU is 100%P/100%F. The lowest nominal set point for a Quad Cities unpiped safety valve is 1254.7 psia.

Because the unpiped safety valve margin analyses are not licensing analyses, some of the conservatism normally assumed in COTRANSA2 analyses is relaxed. The MSIVC-USM analysis with direct scram results in a fairly mild reactor pressurization. The relief valves have sufficient capacity to depressurize the reactor once the valves actuate. The MSIVC-USM analyses with direct scram were performed with plant-specific scram insertion from Section 8.6 of Reference A.1. Technical specification relief valve (RV) opening times and delays were used with nominal RV set points for the MSIVC-USM analyses. Analyses were performed with the safety/relief valve (SRV) not credited (SRVOOS). Analyses were performed at 100%P/108%F, 100%P/100%F and 100%P/87%F for EOFP and at 100%P/108%F and 100%P/100%F for EOFP+1500 MWd/MTU to cover coastdown operation (Reference A.2). For the MSIVC-USM transient, the calculated peak steam line pressure is 1129.7 psia. This results in a calculated margin of 125.0 psi to the lowest unpiped safety valve set point as shown in Table A.1. The required 60 psi margin is met.

For the LRNB-USM analysis, nominal RV set points, opening times and delays are used. All relief valves are assumed to be operable. A best-estimate RV opening delay time of 1.25 seconds and an opening time of 0.20 second were used in the analyses based on values from Reference A.1. Analyses are performed with and without credit for the SRV. Scram insertion is based on plant-specific data provided in Section 8.6 of Reference A.1.

The results of the LRNB-USM analyses are presented in Table A.1. Analyses were performed at 100%P/108%F, 100%P/100%F and 100%P/87%F for EOFP and at 100%P/108%F and 100%P/100%F for EOFP+1500 MWd/MTU to cover coastdown operation (Reference A.2).

Quad Cities analyses indicate that a 1% decrease in rated core power increases pressure margin approximately 4 psi (Reference A.3).

Table A.1 Margin to Opening Unpiped Safety Valve Results

Transient	Exposure	Power/Flow	Maximum SRV Pressure (psia)	Margin (psi)
LRNB-USM	EOFP	100 / 108	1228.2	26.5
LRNB-USM	EOFP+1500 MWd/MTU	100 / 108	1232.6	22.1
LRNB-USM SRVOOS	EOFP	100 / 108	1236.2	18.5
LRNB-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 108	1240.8	13.9
LRNB-USM	EOFP	100 / 100	1229.8	24.9
LRNB-USM	EOFP+1500 MWd/MTU	100 / 100	1233.6	21.1
LRNB-USM SRVOOS	EOFP	100 / 100	1237.9	16.8
LRNB-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 100	1241.9	12.8
LRNB-USM	EOFP	100 / 87	1232.5	22.2
LRNB-USM SRVOOS	EOFP	100 / 87	1240.7	14.0
MSIVC-USM SRVOOS	EOFP	100 / 108	1129.7	125.0
MSIVC-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 108	1129.7	125.0
MSIVC-USM SRVOOS	EOFP	100 / 100	1129.7	125.0
MSIVC-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 100	1129.7	125.0
MSIVC-USM SRVOOS	EOFP	100 / 87	1129.7	125.0

A.1 References

A.1 EMF-2315 Revision 0, *Quad Cities Unit 1 Cycle 17 Principal Transient Analysis Parameters*, Siemens Power Corporation, February 2000.

A.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.

A.3 Letter, M. L. Hymas (SPC) to R. J. Chin (ComEd), "Dresden and Quad Cities Unpiped Safety Valve Margin Analyses," MLH:96:025, May 23, 1996.

Appendix B

Power Load Unbalance Out-of-Service

SPC performed analyses for Quad Cities Unit 1 Cycle 17 to determine MFLCPR multipliers that protect the safety limit MCPR (SLMCPR) when the power load unbalance (PLU) is out of service. Analyses were performed using parameters specified in Reference B.1.

If the PLU is out of service due to testing when a load rejection occurs, the following sequence of events will occur. The PLU will not sense the power load unbalance and a turbine control valve fast closure will not occur. The turbine will overspeed as a result of the imbalance. This turbine overspeed will result in a higher frequency power supply and an increased speed for the recirculation pump that is provided power from the main generator. This will result in increased core flow and an associated increase in thermal power until a turbine trip occurs. A turbine trip is assumed on 62.4 Hz main generator overfrequency at 0.454 second into the event. Per Reference B.1, the turbine overspeed produced a linear increase in power supply frequency from 60 Hz at the initiation of the event to 62.4 Hz at 0.454 second.

The recirculation pump speed is conservatively assumed to increase proportionately to the frequency increase. After the turbine trip, the pump speed linearly decreases to the initial speed in 5 seconds. The end result is a turbine trip occurring from more limiting power and flow conditions. This event is more limiting than the base case load rejection without bypass (LRNB) event described in Section 4.3.

The analyses were performed at the limiting state point of 100%P/108%F for EOFP and EOFP+1500 MWd/MTU (coastdown). Pump overspeed was modeled as a 5% linear increase of one recirculation pump from event initiation (time zero) to a time of 0.454 second. A conservative 5% increase bounds the 60–62.4 Hz frequency excursion of the turbine. After 0.454 second, the pump speed was linearly decreased from an initial normalized pump speed of 1.05 to 1.00 during the following 5 seconds. Turbine stop valve and turbine control valve closures were initiated at 0.454 second. The analyses assumed the conservative scram delay of 0.08 second associated with TCV fast closure.

Analysis results for the GE10 and ATRIUM-9B offset fuel are summarized in Table B.1. MFLCPR multipliers were determined based on the increase in ΔCPR for the PLUOOS events and the corresponding LRNB results at EOFP and EOFP+1500 MWd/MTU. OLMCPR and MFLCPR results provided in Table B.1 are based on PLUOOS analysis results, LRNB analysis results at EOFP and EOFP+1500 MWd/MTU, the plant Technical Specification two-loop SLMCPR of 1.11, and analysis of the limiting system transient analyzed in this report. Actual MFLCPR results may be lower if analyses within ComEd's scope of responsibility result in a ΔCPR higher than those provided in Table 2.1. For single-loop operation, the Technical Specification SLO SLMCPR of 1.12 increases the OLMCPR by 0.01.

The MFLCPR multipliers provided in Table B.1 may also be applied to the reduced flow MCPR limits provided in Section 6.2 to support PLUOOS operation at reduced flow conditions.

Table B.1 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	724	134	1304 / 1270	0.41 / 0.38	0.02 / 0.03
PLUOOS Coastdown	100 / 108	776	137	1313 / 1279	0.43 / 0.42	0.02 / 0.03

MFLCPR Multipliers

Transient	Power/Flow	OLMCPR†	MFLCPR Multiplier†,§
PLUOOS EOFP	100 / 108	1.51 / 1.46	0.986 / 0.979
PLUOOS Coastdown	100 / 108	1.55 / 1.50	0.987 / 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):

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

B.1 References

B.1 EMF-2315 Revision 0, *Quad Cities Unit 1 Cycle 17 Principal Transient Analysis Parameters*, Siemens Power Corporation, February 2000.

Distribution

D. E. Garber, 38 (15)

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Attachment B

Quad Cities Unit 2

Core operating limits report cycle16

Core Operating Limits Report

for

Quad Cities Unit 2 Cycle 16

June 2001

ISSUANCE OF CHANGES SUMMARY

Affected Section	**Affected Pages**	**Summary of Changes**	**Date**
All	All	Original Issue, Cycle 16	1/2000
Special Instructions, 1, 2, 3, 4, and 5	iii, 1, 2, 3, 4, 4-1, 4-2 and 5	Updated for ITS	4/2001
All	All, including page numbering	Update for: • OLMCPR corrections due to FANP thermal conductivity error (10CFR21 Notification Event #37874). • Removal of thermal-mechanical limits from MAPLHGR and inclusion in the MFLPD limit for GE fuel.	6/2001

TABLE OF CONTENTS

SPECIAL INSTRUCTIONS

1.0 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. 50-265, Quad Cities Station, Unit 2 Facility Operating License, License No. DPR-30.

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

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

4. Letter to Dr. R.J. Chin from R.E. Parr, "QC1/QC2 SAFER/GESTR Single Loop Operation Analysis with Delayed LPCI Injection", REP:93.020, February 12, 1993.

5. Lattice Dependent MAPLHGR Report for Quad Cities Nuclear Power Station, Unit 2, Reload 13 Cycle 14, 24A5161-AA Revision 0, December, 1994.

6. SPC Document, EMF-96-185(P) Rev. 4, "Quad Cities LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-9B Fuel", March, 1997, NDIT # NFM9700015 Seq. 3.

7. SPC Document 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, and Rev. 1 Supplement 1, September 1998.

8. SPC Document, EMF-2299 Rev. 0, "Quad Cities Unit 2 Cycle 16 Reload Analysis", November 1999 (COLR Attachment).

9. SPC Document EMF-2302 Rev. 0, "Quad Cities Unit 2 Cycle 16 Plant Transient Analysis", November 1999 (COLR Attachment)

10. NDIT NFM9900245 Seq. 1, "Quad Cities 2 Cycle 16 Design Basis Loading Plan (DBLP)", December 20, 1999.

11. NDIT NFM9900216 Seq. 0, "Quad Cities Unit 2 Cycle 16 Neutronics Licensing Report (NLR), December 8, 1999.

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

13. Letter to Dr. R.J. Chin from David E. Garber, "Quad Cities Unit 2 Cycle 16 Evaluation of Fuel Thermal Conductivity (Non-Proprietary Version for Exelon)," DEG:01:078, May 14, 2001 (COLR Attachment).

14. TODI NFM0000067 Seq. 0, "GE9/GE10 LHGR Improvement Program, J11-03692-LHGR, Revision 1, Class 3, February 2000," 3/30/2000.

LIST OF TABLES

1.0 CONROL 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)Wd + 56.1% **

1.3 DESCRIPTION (SLO):

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

$\leq$ (0.65)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), and
TS 3.4.1 (COLR 2.3)

2.2 DESCRIPTION:

MAPLHGR versus Average Planar Exposure for GE9B-P8DWB310-7G3.0-80M-145-T is determined from Table 2-1.

MAPLHGR versus Average Planar Exposure for GE9B-P8DWB308-10GZ1-80M-145-T is determined from Table 2-2.

MAPLHGR versus Average Planar Exposure for GE10-P8HXB316-8GZ-100M-145-T is determined from Table 2-3.

MAPLHGR versus Average Planar Exposure for GE10-P8HXB312-7GZ-100M-145-T is determined from Table 2-4.

MAPLHGR versus Average Planar Exposure for SPCA9-372B-11GZH-ADV is determined from Table 2-5.

MAPLHGR versus Average Planar Exposure for SPCA9-358B-11GZL-ADV is determined from Table 2-5.

MAPLHGR versus Average Planar Exposure for SPCA9-383B-11GZH-ADV is determined from Table 2-5.

MAPLHGR versus Average Planar Exposure for SPCA9-381B-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 operates in Single Loop.

TABLE 2-1

MAPLHGR vs. AVERAGE PLANAR EXPOSURE FOR BUNDLE TYPE : GE9B-P8DWB310-7G3.0-80M-145-T

LATTICE 731: P8DWL071-NOG-80M-T
LATTICE 1644: P8DWL334-7G3.0-80M-T
LATTICE 1645: P8DWL350-7G3.0-80M-T
LATTICE 1004: P8DWL071-7GE-80M-T

AVERAGE PLANAR EXPOSURE (GWd/ST)	MAPLHGR LIMITS (KW/FT)			
	731	1644	1645	1004
0.00	11.64	12.25	11.78	11.64
0.20	11.57	12.32	11.85	11.57
1.00	11.38	12.46	11.99	11.38
2.00	11.36	12.62	12.19	11.36
3.00	11.41	12.79	12.36	11.41
4.00	11.49	12.96	12.52	11.49
5.00	11.56	13.14	12.69	11.56
6.00	11.63	13.23	12.81	11.63
7.00	11.69	13.30	12.93	11.69
8.00	11.74	13.38	13.04	11.74
9.00	11.78	13.43	13.14	11.78
10.00	11.81	13.46	13.21	11.81
12.50	11.54	13.41	13.22	11.54
15.00	11.16	13.03	12.95	11.16
20.00	10.37	12.29	12.31	10.37
25.00	9.58	11.58	11.62	9.58
27.22	12.314	12.314	12.314	12.314
48.08	10.800	10.800	10.800	10.800
58.97	6.000	6.000	6.000	6.000

TABLE 2-2

MAPLHGR vs. AVERAGE PLANAR EXPOSURE
FOR BUNDLE TYPE : GE9B-P8DWB308-10GZ1-80M-145-T

LATTICE 731: P8DWL071-NOG-80M-T
LATTICE 1642: P8DWL332-8G4.0/2G3.0-80M-T
LATTICE 1669: P8DWL348-8G4.0/2G3.0-80M-T
LATTICE 1188: P8DWL071-10GE-80M-T

AVERAGE PLANAR EXPOSURE (GWd/ST)	MAPLHGR LIMITS (KW/FT)			
	731	1642	1669	1188
0.00	11.64	11.63	11.24	11.64
0.20	11.57	11.69	11.31	11.57
1.00	11.38	11.83	11.44	11.38
2.00	11.36	12.04	11.62	11.36
3.00	11.41	12.25	11.82	11.41
4.00	11.49	12.48	12.02	11.49
5.00	11.56	12.58	12.24	11.56
6.00	11.63	12.69	12.44	11.63
7.00	11.69	12.86	12.62	11.69
8.00	11.74	13.04	12.75	11.74
9.00	11.78	13.19	12.90	11.78
10.00	11.81	13.31	13.05	11.81
12.50	11.54	13.32	13.14	11.54
15.00	11.16	12.96	12.92	11.16
20.00	10.37	12.22	12.25	10.37
25.00	9.58	11.53	11.57	9.58
27.22	12.314	12.314	12.314	12.314
48.08	10.800	10.800	10.800	10.800
58.97	6.000	6.000	6.000	6.000

TABLE 2-3

MAPLHGR vs. AVERAGE PLANAR EXPOSURE
FOR BUNDLE TYPE : GE10-P8HXB316-8GZ-100M-145-T

LATTICE 7400: P8HXL071-NOG-100M
LATTICE 7467: P8HXL341-6G4.0/2G3.0-100M
LATTICE 7469: P8HXL357-6G4.0/2G3.0-100M
LATTICE 7468: P8HXL341-8G3.0-100M
LATTICE 7404: P8HXL071-8GE-100M

	MAPLHGR LIMITS (KW/FT)				
AVERAGE PLANAR EXPOSURE (GWd/ST)	1054 7400*	1916 7467*	1917 7469*	1918 7468*	1807 7404*
0.00	11.85	12.00	11.11	12.08	11.85
0.20	11.78	12.06	11.14	12.15	11.78
1.00	11.59	12.18	11.24	12.30	11.59
2.00	11.57	12.36	11.44	12.53	11.57
3.00	11.61	12.50	11.70	12.68	11.61
4.00	11.68	12.60	11.99	12.82	11.68
5.00	11.75	12.71	12.26	12.96	11.75
6.00	11.81	12.84	12.37	13.12	11.81
7.00	11.86	13.01	12.51	13.29	11.86
8.00	11.91	13.20	12.68	13.44	11.91
9.00	11.94	13.39	12.86	13.53	11.94
10.00	11.97	13.52	13.01	13.55	11.97
12.50	11.75	13.44	13.09	13.44	11.75
15.00	11.38	13.06	12.84	13.07	11.38
20.00	10.59	12.32	12.21	12.33	10.59
25.00	9.81	11.60	11.54	11.61	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

* Indicates Exelon lattice identifiers as opposed to the GE lattice identifiers

TABLE 2-4

MAPLHGR vs. AVERAGE PLANAR EXPOSURE
FOR BUNDLE TYPE : GE10-P8HXB312-7GZ-100M-145-T

LATTICE 7400: P8HXL071-NOG-100M
LATTICE 7405: P8HXL336-3G4.0/4G3.0-100M
LATTICE 7406: P8HXL354-1G4.0/6G3.0-100M
LATTICE 7407: P8HXL336-7G3.0-100M
LATTICE 7408: P8HXL071-7GE-100M

MAPLHGR LIMITS (KW/FT)					
AVERAVGE PLANAR EXPOSURE (GWd/ST)	1054 7400*	1808 7405*	1809 7406*	1810 7407*	1811 7408*
0.00	11.85	12.01	11.27	12.04	11.85
0.20	11.78	12.08	11.31	12.11	11.78
1.00	11.59	12.23	11.42	12.27	11.59
2.00	11.57	12.43	11.65	12.49	11.57
3.00	11.61	12.65	11.93	12.72	11.61
4.00	11.68	12.88	12.24	12.96	11.68
5.00	11.75	13.09	12.58	13.15	11.75
6.00	11.81	13.22	12.94	13.30	11.81
7.00	11.86	13.32	13.15	13.41	11.86
8.00	11.91	13.40	13.32	13.46	11.91
9.00	11.94	13.45	13.43	13.47	11.94
10.00	11.97	13.47	13.50	13.45	11.97
12.50	11.75	13.35	13.45	13.35	11.75
15.00	11.38	12.97	13.10	12.97	11.38
20.00	10.59	12.23	12.41	12.24	10.59
25.00	9.81	11.51	11.74	11.52	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

* Indicates Exelon lattice identifiers as opposed to GE lattice identifiers

TABLE 2-5

MAPLHGR vs. AVERAGE PLANAR EXPOSURE FOR BUNDLE TYPES :

SPCA9-372B-11GZH-ADV
SPCA9-358B-11GZL-ADV
SPCA9-383B-11GZH-ADV
SPCA9-381B-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. The LHGR limit for GE fuel types in the Q2C16 core are as follows:

TABLE 3-1 LHGR Limits For Bundle Type GE9B-P8DWB310-7G3.0-80M-145-T	
NODAL EXPOSURE (GWD/MTU)	**LHGR (kW/ft)**
0.00	14.40
12.87	14.40
27.47	12.31
49.65	10.80
61.61	6.00

TABLE 3-2 LHGR Limits For Bundle Type GE9B-P8DWB308-10GZ1-80M-145-T	
NODAL EXPOSURE (GWD/MTU)	**LHGR (kW/ft)**
0.00	14.40
12.50	14.40
27.21	12.31
33.07	11.88
38.58	11.38
44.09	10.92
49.31	10.80
61.30	6.00

TABLE 3-3 LHGR Limits For Bundle Type GE10-P8HXB316-8GZ-100M-145-T	
NODAL EXPOSURE (GWD/MTU)	**LHGR (kW/ft)**
0.00	14.40
12.82	14.40
27.25	12.31
49.22	10.80
60.94	6.00

TABLE 3-4 LHGR Limits For Bundle Type GE10-P8HXB312-7GZ-100M-145-T	
NODAL EXPOSURE (GWD/MTU)	**LHGR (kW/ft)**
0.00	14.40
13.00	14.40
27.27	12.31
49.01	10.80
60.70	6.00

B. The LHGR limits are provided in the table below for all of the SPC fuel types:

SPCA9-372B-11GZH-ADV
SPCA9-358B-11GZL-ADV
SPCA9-383B-11GZH-ADV
SPCA9-381B-12GZL-ADV

TABLE 3-5 LHGR Limits For SPC Fuel	
Average Planar Exposure (GWd/MTU)	**ATRIUM-9B LHGR (kW/ft)**
0.0	14.4
15.0	14.4
61.1	8.32

C. The Protection Against Power Transient (PAPT) LHGR limits are provided in the table below for all of the SPC fuel types:

SPCA9-372B-11GZH-ADV
SPCA9-358B-11GZL-ADV
SPCA9-383B-11GZH-ADV
SPCA9-381B-12GZL-ADV

TABLE 3-6 PAPT LHGR Limits For SPC Fuel	
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 Operating Limits are also based on a 15 psi reduction in steam dome pressure and Technical Specification SCRAM times.

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 or 4-3 using the appropriate operating condition and flow rate, or

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

   The Percent Rated Core Flow is based on 98 MLB/hr with 110% Maximum Flow in Manual Flow Control Mode. (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 4-4 or 4-5 using the appropriate operating conditions.

   The Percent Rated Core Flow is based on 98 MLB/hr with 108% Maximum Flow in Automatic Flow Control Mode. (Technical Requirements Manual 2.1.a.1 and Bases of TS 3.2.2)

4. During PLU Out of Service conditions a 0.967 MFLCPR administrative limit shall be used during operation up to EOFP and an administrative limit of 0.963 shall be used during coastdown for all fuel types in the core.

TABLE 4-1
Steady State MCPR Operating Limits
(Based on 1.11 Safety Limit MCPR for Dual Loop Operation)

Operating Condition	GE9	GE10	ATRIUM-9B
Normal Operation includes ICF, RVOOS, TIPOOS[2], and SRVOOS	1.53	1.50	1.46
Single Loop Operation includes RVOOS, TIPOOS[2], and SRVOOS	1.54	1.51	1.47
EOD/EOOS Dual Loop Operation includes ICF, RVOOS, TIPOOS[2], and SRVOOS <u>plus</u> FFTR, FHOOS, Coastdown, or any combination thereof	1.56	1.59	1.50
EOD/EOOS Single Loop Operation includes ICF, RVOOS, TIPOOS[2], and SRVOOS <u>plus</u> FFTR, FHOOS, Coastdown, or any combination thereof	1.57	1.60	1.51
One Main Turbine Bypass Valve Out of Service[1] includes ICF, RVOOS, TIPOOS[2], and SRVOOS	1.54	1.51	1.47
All Main Turbine Bypass Valves Out of Service[1] includes ICF, RVOOS, TIPOOS[2], and SRVOOS	1.56	1.53	1.50

[1] Operation with bypass valves out-of-service (one or all) is **<u>not</u>** supported during coastdown.
[2] 40% TIPOOS with 100% TIP strings available at startup, 50% of the LPRM's out-of-service (LPRM substitution model on or off), and 2000 EFPH LPRM calibration interval.

For core flows less than rated, reduced flow $MCPR_f$ curves for Manual Flow Control are provided in Tables 4-2 and 4-3. $MCPR_f$ values for Automatic Flow Control are provided in Tables 4-4 and 4-5. ***Percent Rated Core Flow based on 98 MLB/hr with 110% Maximum Flow in Manual Flow Control and 108% Maximum Flow in Automatic Flow Control operation (Requirements Manual TSR 2.1.a.1 and Bases of TS 3.2.2).***

TABLE 4-2
Reduced Flow $MCPR_f$ Limit for Manual Flow Control based on 1.11 SLMCPR (Two-Loop Operation)

Core Flow (% of rated)	GE 9 $MCPR_f$ Limit	GE 10 $MCPR_f$ Limit	ATRIUM-9B Offset $MCPR_f$ Limit
110	1.11	1.11	1.11
30	1.98	1.96	2.02
0	2.54	2.52	2.57

TABLE 4-3
Reduced Flow $MCPR_f$ Limit for Manual Flow Control based on 1.12 SLMCPR (Single-Loop Operation)

Core Flow (% of rated)	GE 9 $MCPR_f$ Limit	GE 10 $MCPR_f$ Limit	ATRIUM-9B Offset $MCPR_f$ Limit
110	1.12	1.12	1.12
30	1.99	1.97	2.03
0	2.55	2.53	2.58

TABLE 4-4
Reduced Flow $MCPR_f$ Limit for Automatic Flow Control based on 1.11 SLMCPR (Two-Loop Operation)

Core Flow (% of rated)	GE 9 Base Case	GE 9 EOD/EOOS	GE 10 Base Case	GE 10 EOD/EOOS	ATRIUM-9B Base Case	ATRIUM-9B EOD/EOOS
108	1.53	1.56	1.50	1.59	1.46	1.50
30	2.84	2.89	2.77	2.93	2.78	2.85
0	3.76	3.82	3.67	3.87	3.65	3.75

TABLE 4-5
Reduced Flow $MCPR_f$ Limit for Automatic Flow Control based on 1.12 SLMCPR (Single-Loop Operation)

Core Flow (% of rated)	GE 9 Base Case	GE 9 EOD/EOOS	GE 10 Base Case	GE 10 EOD/EOOS	ATRIUM-9B Base Case	ATRIUM-9B EOD/EOOS
108	1.54	1.57	1.51	1.60	1.47	1.51
30	2.85	2.90	2.78	2.94	2.79	2.86
0	3.77	3.83	3.68	3.88	3.66	3.76

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.

Attachments to the

Core Operating Limits Report

for

Quad Cities Unit 2 Cycle 16

COLR Attachments

Attachment 1. Quad Cities Unit 2 Cycle 16 Reload Analysis

Attachment 2. Quad Cities Unit 2 Cycle 16 Plant Transient Analysis

Attachment 3. Letter, "Quad Cities Unit 2 Cycle 16 Evaluation of Fuel Thermal Conductivity (Non-Proprietary Version for Exelon)," DEG:01:078, May 14, 2001

COLR ATTACHMENT 1

Quad Cities Unit 2 Cycle 16

Reload Analysis

NFM: BSS COPY
NOIT # NFM 9900249
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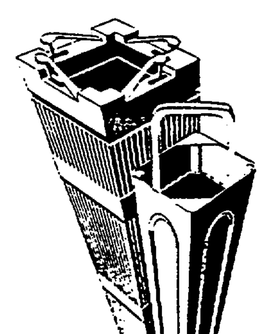
SIEMENS

EMF-2299
Revision 0

Quad Cities Unit 2 Cycle 16 Reload Analysis

November 1999



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EMF-2299
Revision 0

Quad Cities Unit 2 Cycle 16
Reload Analysis

Prepared:	S. W. Jones, Engineer BWR Neutronics	11/8/99 Date
Prepared:	AW Will A. W. Will, Engineer BWR Safety Analysis	11/8/99 Date
Concurred:	H. D. Curet, Manager Product Licensing	11/11/99 Date
Approved:	O. C. Brown, Manager BWR Neutronics	10 Nov 99 Date
Approved:	M. E. Garrett, Manager BWR Safety Analysis	11-11-99 Date
Approved:	T. M. Howe, Manager Product Mechanical Engineering	10 Nov 99 Date
Approved:	D. J. Denver, Manager Commercial Operations	11.11.99 Date

gdh

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Nature of Changes

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

Contents

Tables

Figures

Nomenclature

APRM	average power range monitor
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 16 reload for Quad Cities Unit 2. 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 2 Cycle 16, Commonwealth Edison Company (ComEd) has responsibility for portions of the reload safety analysis. This document describes only the Cycle 16 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 16. The limiting events and operating limits must be determined in conjunction with results from ComEd analyses.

The Quad Cities Unit 2 Cycle 16 core consists of a total of 724 fuel assemblies, including 240 unirradiated QCB-2 ATRIUM™-9B* offset assemblies, 216 irradiated ATRIUM-9B offset assemblies, 143 irradiated GE10 assemblies, and 125 irradiated GE9 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 2 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 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)†

* 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 2000 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 16 of Quad Cities Unit 2 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 2 Cycle 16 core have been determined in single-phase flow tests of full-scale assemblies. The hydraulic demand curves for SPC ATRIUM-9B offset, GE9, and GE10 fuel in the Quad Cities Unit 2 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*	12.8%	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)	3860 MWt
Feedwater Flow Rate (at SLMCPR)	15.2 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 2000 EFPH calibration interval, and up to 50% of the LPRMs out of service (LPRM substitution model on or off).

‡ 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
SPCA9-381B-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 16 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	0.225
23	0.455
22	1.052
21	1.294
20	1.437
19	1.499
18	1.521
17	1.520
16	1.503
15	1.477
14	1.467
13	1.417
12	1.349
11	1.263
10	1.161
9	1.044
8	0.914
7	0.784
6	0.663
5	0.563
4	0.491
3	0.437
2	0.355
Bottom 1	0.110

Licensing Axial Exposure Ratio (EOFP)
Average Bottom 8 ft/12 ft = 1.1097

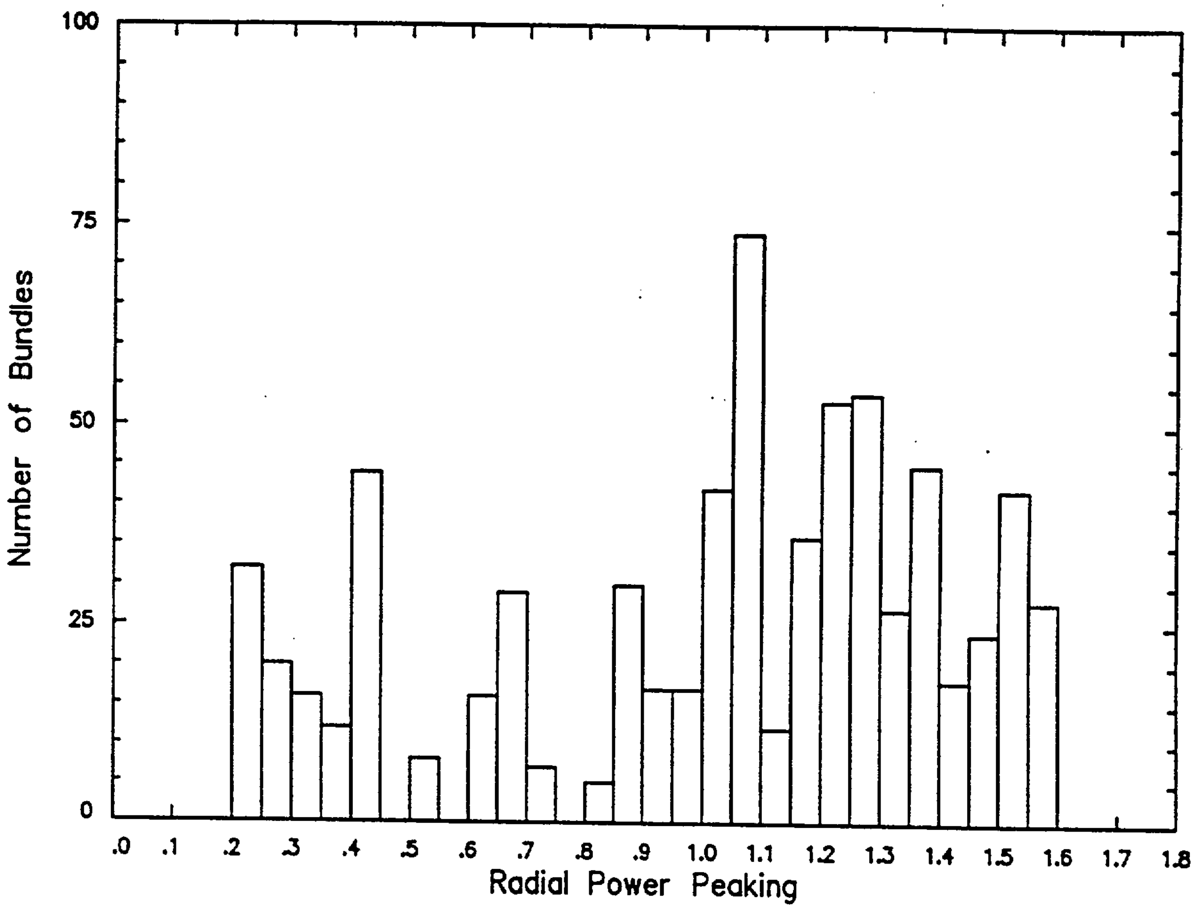

Figure 3.1 Design Basis Radial Power Distribution for SLMCPR Determination

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 2 Cycle 16 Safety Limit Local Peaking Factors With Channel Bow at Assembly Exposure of 25000 MWd/MTU (SPCA9-383B-11GZH-ADV)

Control Rod Corner

Control Rod Corner

1.006	1.021	1.040	1.047	1.115	1.101	1.100	0.994	0.979
1.021	0.983	1.053	1.030	0.960	1.067	0.931	0.998	0.972
1.040	1.053	0.977	1.121	1.126	1.082	1.026	0.876	1.052
1.047	1.030	1.121		Internal		1.038	0.966	1.007
1.115	0.960	1.126		Water		1.073	0.827	0.984
1.101	1.067	1.082		Channel		1.029	0.921	0.964
1.100	0.931	1.026	1.038	1.073	1.029	0.821	0.912	0.961
0.994	0.998	0.876	0.966	0.827	0.921	0.912	0.810	0.917
0.979	0.972	1.052	1.007	0.984	0.964	0.961	0.917	0.823

Figure 3.3 Quad Cities Unit 2 Cycle 16 Safety Limit Local Peaking Factors With Channel Bow at Assembly Exposure of 25000 MWd/MTU (SPCA9-381B-12GZL-ADV)

4.0 Nuclear Design Analysis

4.1 *Fuel Bundle Nuclear Design Analysis*

Assembly Average Enrichment

ATRIUM-9B offset	(QCB-2 Type H)	3.83 wt%
	(QCB-2 Type L)	3.81 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
SPCA9-4.13L-11G7.0	Reference 9.7
SPCA9-4.14L-12G7.0	Reference 9.7
SPCA9-4.29L-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 QCB-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 QCB-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.

* 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 EOC15A*, MWd/MTU (nominal value)	28,654
Core Exposure at BOC16, MWd/MTU (from nominal EOC15A)	16,917
Core Exposure at EOC16, MWd/MTU (licensing basis)	31,467

NOTE: Analyses in this report are applicable to a core exposure of 31,467 MWd/MTU. EOD/EOOS cycle extension analyses (References 9.3 and 9.6) are applicable for Cycle 16 provided full power capability is lost prior to reaching a core exposure of 31,467 MWd/MTU.

< Cycle 16 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 2 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.

---

* Cycle 15 is designated to have operated between June 1997 and September 1997. Cycle 15A denotes the subsequent Cycle 15 operation following the replacement of an assembly.

- 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 16 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†	0.99	(-.01)	0.90	(.11)
2.	68.5/45‡	0.79	(.01)	0.71	(.09)
3.	58.5/45§	0.58	(.05)	0.48	(.07)
4.	23/19.4**	0.41	(.05)	0.32	(.04)
5.	37/38.8††	0.30	(.06)	0.24	(.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 16 as long as operating procedures and precautions defined in the ICAs are followed.

* DR_{CY16} - DR_{CY15} 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

* 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.

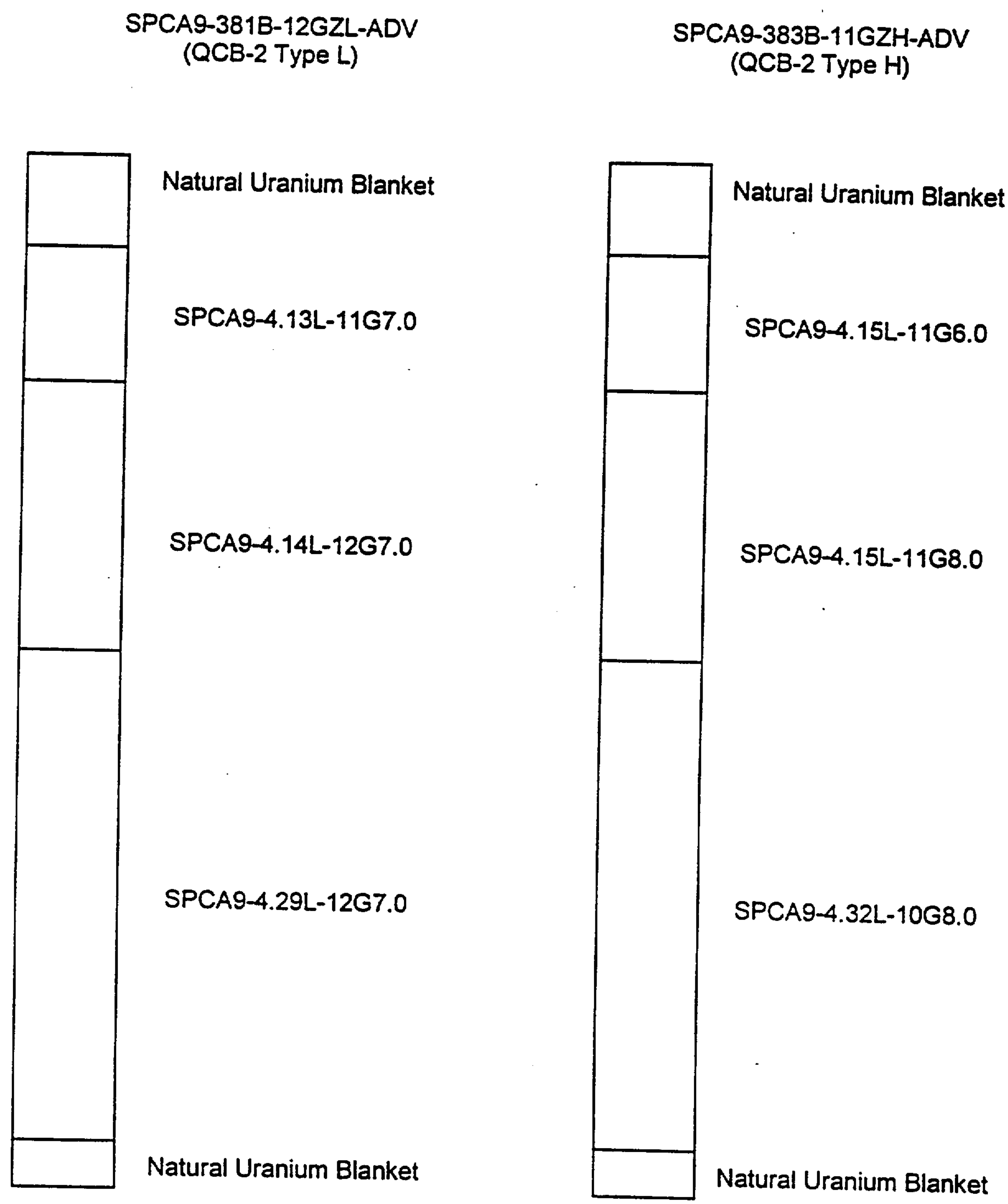

Figure 4.1 Quad Cities Unit 2 Reload Batch QCB-2 Axial Fuel Assembly Design

2	6	2	3	2	6	3	2	3	6	3	4	2	1*	16
6	3	6	4	6	4	5	3	5	3	6	5	2	2	16
2	6	3	6	3	6	2	6	3	5	5	5	4	16	15
3	4	6	2	3	3	5	2	2	3	5	3	2	16	16
2	6	3	3	2	5	4	3	4	6	5	4	2	16	16
6	4	6	3	5	3	5	6	5	5	5	3	1	15	
3	5	2	5	4	5	3	5	4	3	6	3	16		
2	3	6	2	3	6	5	2	3	5	5	2	16		
3	5	3	2	4	5	4	3	2	6	2	15	16		
6	3	5	3	6	5	3	5	6	2	16	16			
3	6	5	5	5	5	6	5	2	16	16				
4	5	5	3	4	3	3	2	15	16					
2	2	4	2	2	1	16	16	16						
1	2	16	16	16	15									
16	16	15	16	16										

Fuel Type	Number of Assemblies	Bundle Description	Cycle Loaded
1	15	GE10-P8HXB312-7GZ-100M-145-CECO	14
2	128	GE10-P8HXB316-8GZ2-100M-145-CECO	14
3	152	SPCA9-372B-11GZH-ADV	15
4	64	SPCA9-358B-11GZL-ADV	15
5	136	SPCA9-383B-11GZH-ADV	16
6	104	SPCA9-381B-12GZL-ADV	16
15	25	GE9B-P8DWB310-7G3.0-80M-145-CECO	13
16	100	GE9B-P8DWB308-10GZ-80M-145-CECO	13

Figure 4.2 Quad Cities Unit 2 Cycle 16 Reference Loading Map (Quarter-Core Symmetric Loading)

* This location is asymmetric in the lower left-hand quadrant which contains fuel type 15.

5.0 Anticipated Operational Occurrences

Applicable Generic Transient Analysis Report — References 9.6 and 9.14

5.1 *Analysis of Plant Transients at Rated Conditions* — References 9.3, 9.6 and 9.14

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	132	653	1296	0.41/0.37/0.33	COTRANSA2
FWCF‡,§	100	108	135	634	1185	0.42/0.38/0.35	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 9.14

Limiting Transient: Feedwater Controller Failure (FWCF)

---

* This data to be furnished by ComEd.

† ΔCPR results for GE9/GE10/ATRIUM-9B offset fuel.

‡ Based on Technical Specification limiting scram performance parameters.

§ Fuel dependent cycle-specific OLMCPR penalties of 0.02 (GE9), 0.10 (GE10) and 0.03 (ATRIUM-9B offset) are required to support EOD/EOOS operation with FFTR, FHOOS, coastdown, or any combination thereof. Other EOD/EOOS conditions require no OLMCPR penalty.

5.4 *ASME Overpressurization Analysis**

Limiting Event	MSIV Closure
Worst Single Failure	Valve Position Scram
Maximum Pressure (Lower Plenum)	1359 psig
Maximum Steam Dome Pressure†	1333 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	0.41/0.37/0.33	1.52/1.48/1.44
FWCF	100	108	0.42/0.38/0.35	1.53/1.49/1.46

MCPR Operating Limit**	1.53/1.49/1.46
MCPR Operating Limit With EOD/EOOS Penalty**,††	1.55/1.59/1.49
MCPR Operating Limits at Off-Rated Conditions§	
Reduced Flow MCPR Limits:	
Manual Flow Control	Figure 5.1
Automatic Flow Control**	Figures 5.2 and 5.3

* 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 GE9/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.

†† Fuel dependent cycle-specific OLMCPR penalties of 0.02 (GE9), 0.10 (GE10) and 0.03 (ATRIUM-9B offset) are required to support EOD/EOOS operation with FFTR, FHOOS, coastdown, or any combination thereof. Other EOD/EOOS conditions require no OLMCPR penalty.

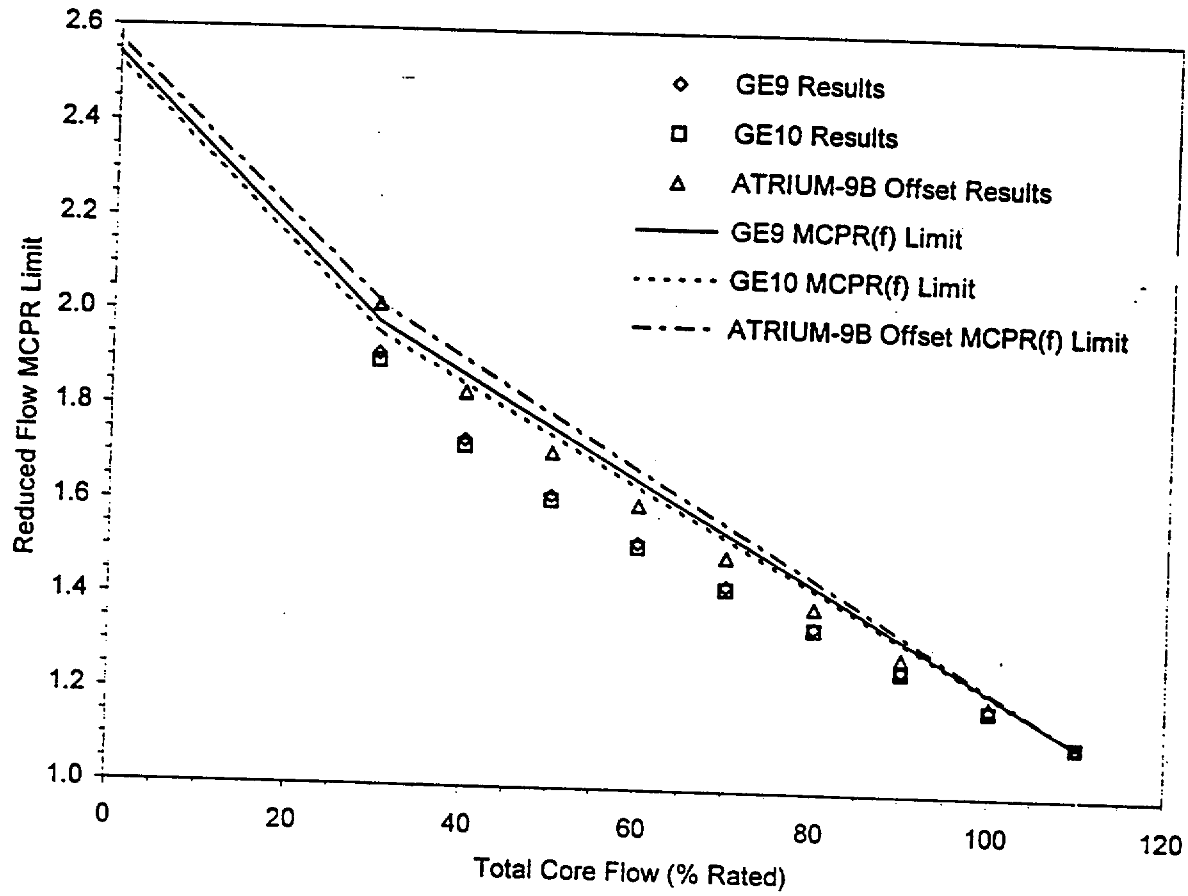

Figure 5.1 Reduced Flow MCPR Limit for Manual Flow Control (SLMCPR = 1.11)

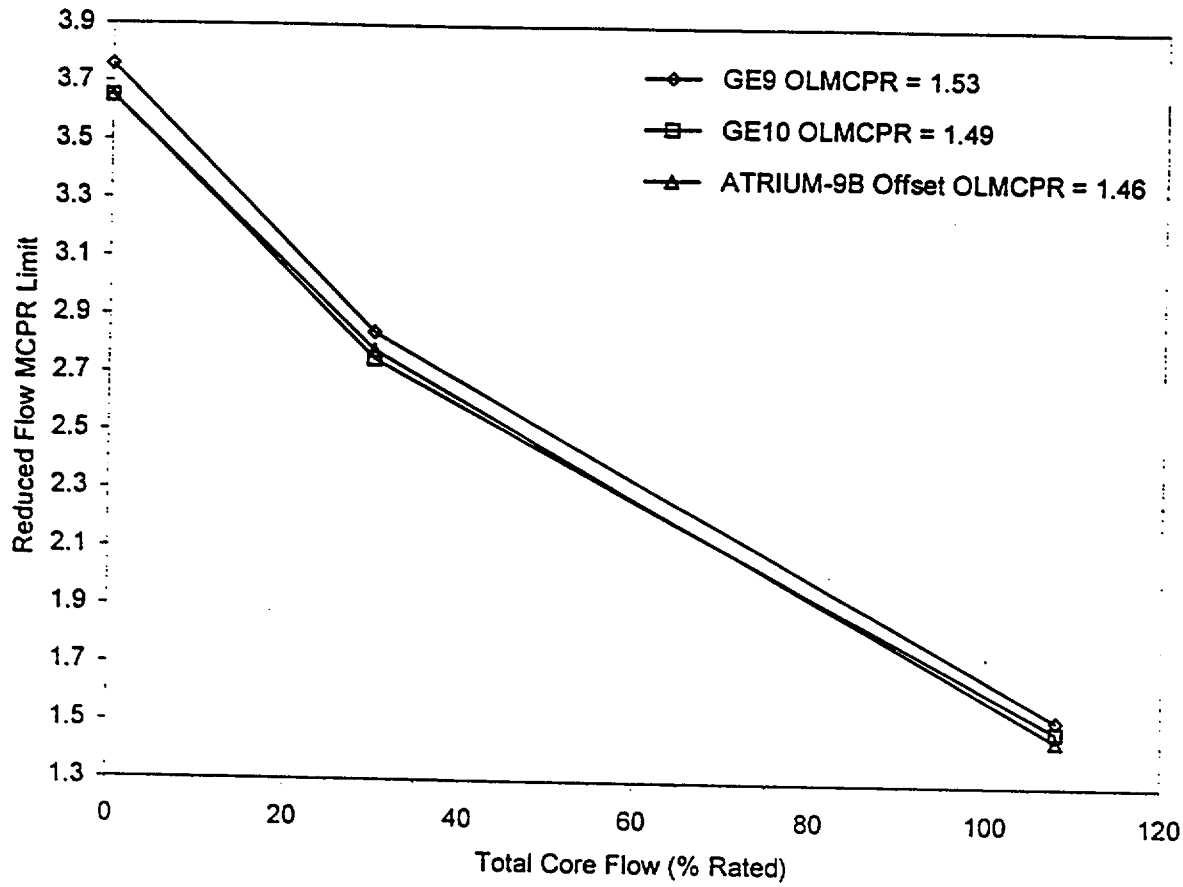

Figure 5.2 Reduced Flow MCPR Limit for Automatic Flow Control (Base OLMCPR)

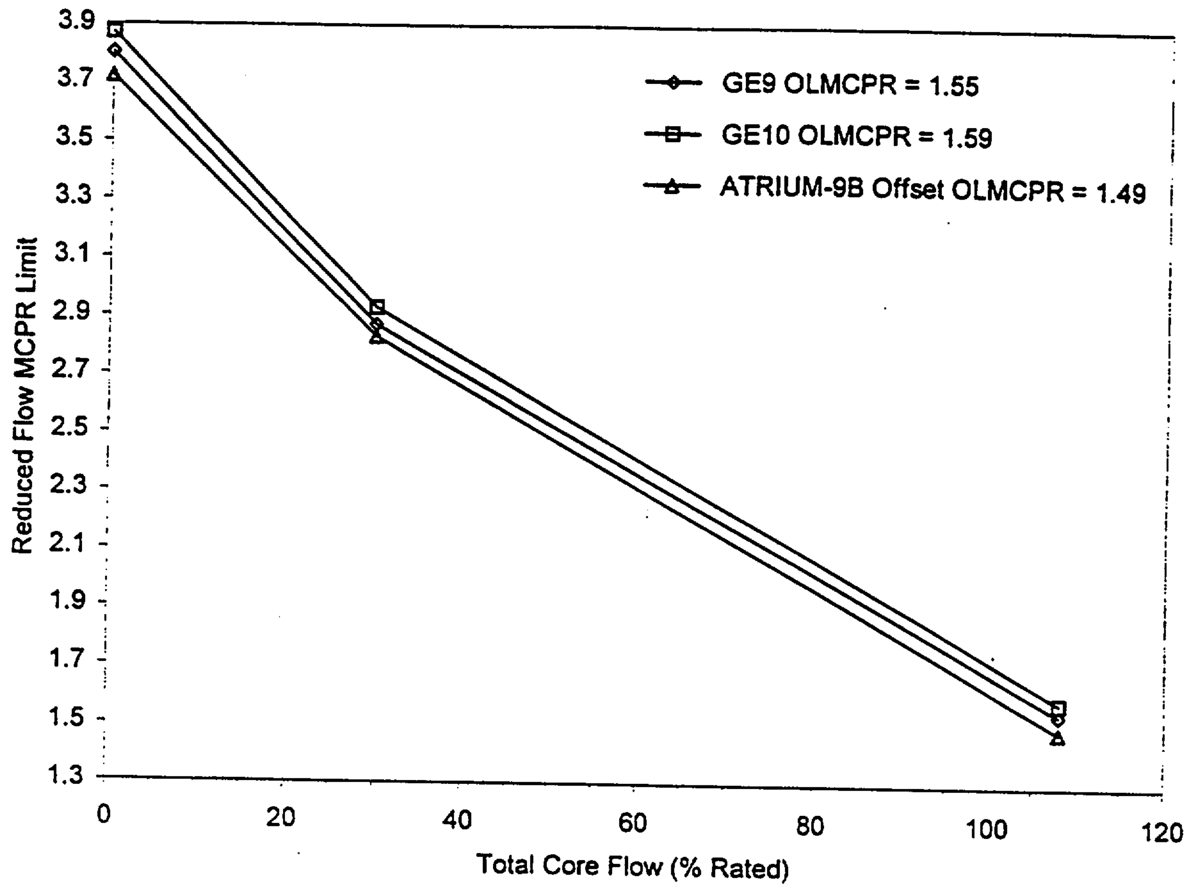

Figure 5.3 Reduced Flow MCPR Limit for Automatic Flow Control (EOD/EOOS OLMCPR)

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 (QCB-2) fuel for Cycle 16 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

The peak cladding temperature (PCT) for ATRIUM-9B offset fuel at Quad Cities was determined from 10 CFR 50.46 reporting estimates to be 1961°F (Reference 9.12). The limiting PCT occurred at a planar exposure of 15 GWd/MTU. Corresponding peak local metal-water reaction (MWR) and total core-wide MWR results are 2.33% and < 0.12%, respectively. These results are based on the condition report evaluation (Reference 9.13) to determine the impact of loading ATRIUM-9B offset fuel next to non-offset fuel (Item 13 of Reference 9.12). The analysis also explicitly accounted for the first ten items identified in Reference 9.12. Items 11 and 12 of Reference 9.12 were accounted for by adding 20°F to the PCT and increasing MWR results by 10%. These results continue to support the MAPLHGR limits reported in Reference 9.5.

The PCT, peak local MWR and total core-wide MWR results for the Cycle 16 ATRIUM-9B offset reload fuel are 1910°F, 2.42% and < 0.12%, respectively. Cycle 16 analyses explicitly accounted for the first ten items identified in Reference 9.12. Applying the estimated PCT increase (20°F), due to Items 11 and 12, results in a PCT of 1930°F. The 52°F PCT increase from Item 13 of Reference 9.12 is not applied to Cycle 16 fuel because there is no fresh ATRIUM-9B offset fuel loaded next to non-offset fuel in the Cycle 16 core. Cycle 16 PCT results are bound by results provided in Reference 9.12.

6.2 *Control Rod Drop Accident*

A disposition of the increased analytical neutron flux scram set point for SPC control rod drop analysis methodology is provided in Reference 9.14. Analysis of the control rod drop accident 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)	GE9 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†

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* 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 2000 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 16.

Off-Rated Conditions MCPR Limits:

Manual Flow Control	Figure 5.1
Automatic Flow Control	Figures 5.2 and 5.3

7.2.3 Linear Heat Generation Rate — Figure 2.1 of Reference 9.7

Steady-State LHGR Limits

GE9 and 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 GE9 and GE10 fuel.

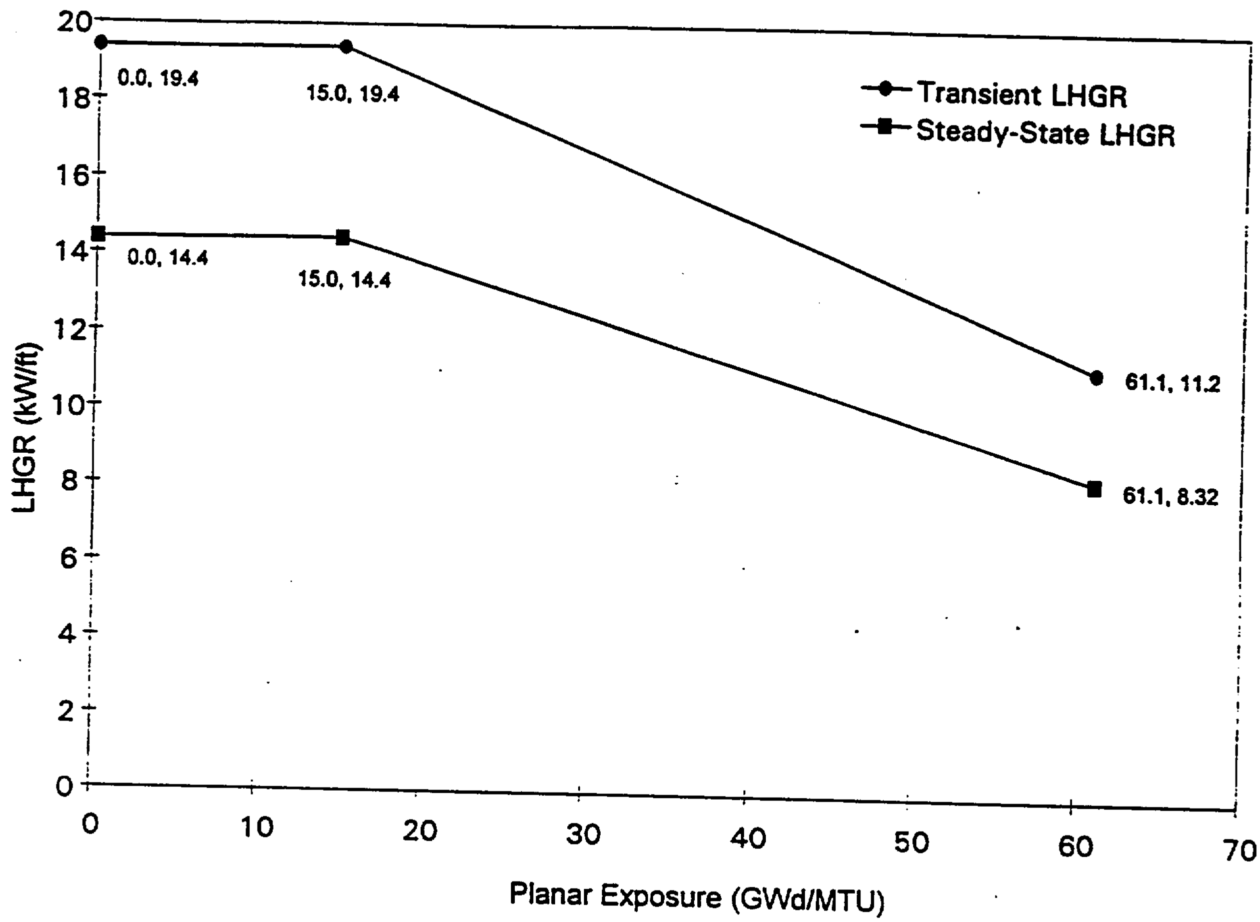


Figure 7.1 Steady-State and Protection Against Power Transient LHGR Limits for ATRIUM-9B Offset Fuel

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

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

8.8 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-2302 Revision 0, *Quad Cities Unit 2 Cycle 16 Plant Transient Analysis*, Siemens Power Corporation, October 1999.

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-96-185(P) Revision 4, *Quad Cities LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™-9B Fuel*, Siemens Power Corporation, August 1998.

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-2280(P) Revision 0, *Fuel Design Report for Quad Cities Unit 2 Cycle 16 ATRIUM™-9B Fuel Assemblies*, Siemens Power Corporation, September 1999.

9.8 ANF-89-014(P)(A) Revision 1 and Supplements 1 and 2, *Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X Reload Fuel*, Advanced Nuclear Fuels Corporation, October 1991.

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 2 Cycle 16 Transient Power History Data for Confirming Mechanical Limits for GE Fuel," DEG:99:267, October 5, 1999.

9.12 Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "10 CFR 50.46 PCT Reporting for the Quad Cities Units," DEG:99:288, October 8, 1999.

9.13 Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Transmittal of Condition Report 7849," DEG:99:220, August 10, 1999.

9.14 EMF-2222(P) Revision 0, *Dresden and Quad Cities Evaluation of Changed Analytical Neutron Flux Scram and Safety Valve Set Points*, Siemens Power Corporation, August 1999.

Distribution

Controlled Distribution

D. E. Garber, 38 (6)
S. W. Jones, 34
A. W. Will, 23

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

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

COLR ATTACHMENT 2

Quad Cities Unit 2 Cycle 16

Plant Transient Analysis

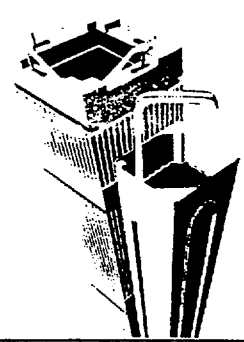
NFM: BSS Copy
NDIT# NFM9900248
seq 0 12/3/99

SIEMENS

EMF-2302
Revision 0

Quad Cities Unit 2 Cycle 16 Plant Transient Analysis

November 1999



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EMF-2302
Revision 0

Quad Cities Unit 2 Cycle 16
Plant Transient Analysis

Prepared: A W Will — 11/8/99
A. W. Will, Engineer
BWR Safety Analysis
Date

Reviewed: R. R. Schnepp — 11/9/99
R. R. Schnepp, Team Leader
BWR Safety Analysis
Date

Concurred: — 11/11/99
H. D. Curet, Manager
Product Licensing
Date

Approved: ME Garrett — 11/9/99
M. E. Garrett, Manager
BWR Safety Analysis
Date

Approved: — 10 Nov 99
O. C. Brown, Manager
BWR Neutronics
Date

Approved: ANK MEG — 11/9/99
T. M. Howe, Manager
Product Mechanical Engineering
Date

Approved: — 11 Nov 99
D. J. Denver, Manager
Commercial Operations
Date

paj

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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
ITS	improved technical specification scram
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 2 Cycle 16 (QC2C16). The Cycle 16 core contains 125 exposed GE9 assemblies, 143 exposed GE10 assemblies, 216 exposed ATRIUM™-9B* offset assemblies and 240 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 16 operation is presented in this report. The analyzed core design is documented in Reference 1.

For Quad Cities Unit 2 Cycle 16 (QC2C16), Commonwealth Edison Company (ComEd) has responsibility for portions of the reload safety analysis. This document describes only the Cycle 16 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 16. 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 and 9) with the implementation of new ATRIUM-9B additive constants (Reference 10), and the use of the CASMO-3G/MICROBURN-B code package (Reference 11). The transient analyses for Quad Cities Unit 2 Cycle 16 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 Quad Cities Unit 2 Cycle 16 have been evaluated to be hydraulically compatible with GE9/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 16 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 16 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 16 operation.
- Section 3.0 is the Cycle 16 evaluation of the Quad Cities disposition of events and the identification of cycle-specific analyses.
- Section 4.0 is the Cycle 16 transient analyses for thermal margin.
- Section 5.0 is the Cycle 16 ASME overpressurization analyses.
- Section 6.0 is the Cycle 16 evaluation of off-rated power and flow operation.
- Section 7.0 is the Cycle 16 evaluation of cycle-specific EOD/EOOS OLMCPR penalties and evaluation of limiting EOD/EOOS conditions for maximum pressurization events.

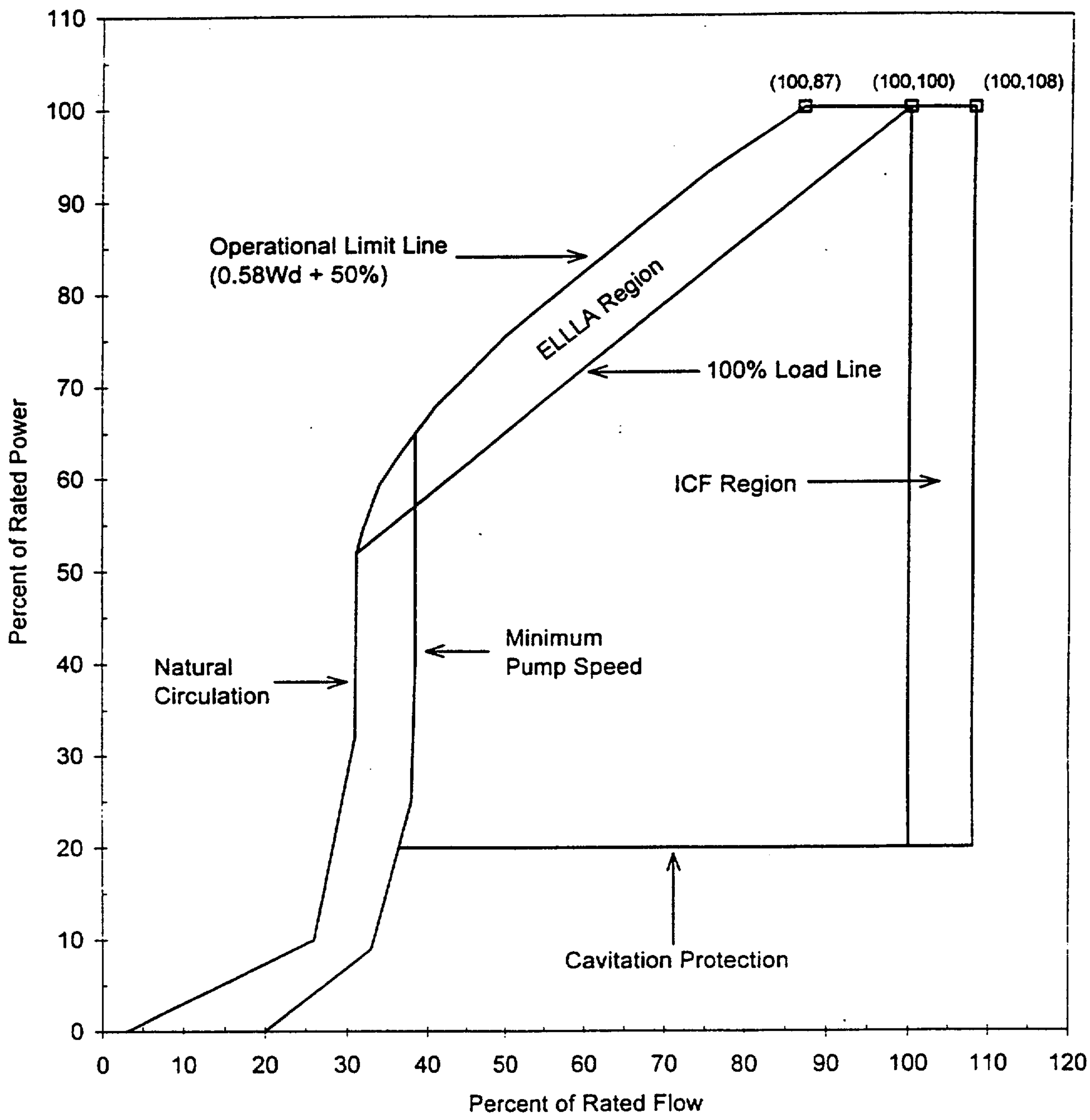

Figure 1.1 Quad Cities Unit 2
Operating Power/Flow Map

2.0 Summary

The determination of thermal margin requirements for Quad Cities Unit 2 Cycle 16 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 2 Cycle 16. 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 16 (see Table 3.1) and is therefore not explicitly analyzed. LRNB and FWCF thermal margin analyses at 100%P/87%F 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 MCPR safety limit (SLMCPR) analysis for Quad Cities Unit 2 Cycle 16 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 (GE9, GE10, and ATRIUM-9B offset) in the core for Cycle 16 and includes 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. These limits are obtained by adding the limiting ΔCPR (Table 2.1) for each fuel type to the plant Technical Specification two-loop SLMCPR safety limit of 1.11. OLMCPRs are provided for all fuel types in the core for Cycle 16. Key parameters from the transient analyses are provided in Table 2.3.

Maximum system pressure for the ASME overpressure evaluation 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 ASME 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 Quad Cities Unit 2 Cycle 16 core. 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 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 16, fuel type dependent OLMCPR penalties are applied to the base case OLMCPRs to support EOD/EOOS operation. The EOD/EOOS OLMCPR penalties for GE9, GE10 and ATRIUM-9B offset fuel are 0.02, 0.10 and 0.03, 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 16.

Base case analyses refer to analyses that do not fully support EOD/EOOS conditions and are representative of normal operation. The base case analyses do support some EOD/EOOS conditions. In particular the base case analyses support increased core flow (ICF) and relief valve out of service (RVOOS). Base case ASME overpressurization analyses support safety/relief valve out of service (SRVOOS).

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* OLMCPR penalties are required for operation with final feedwater temperature reduction (FFTR), feedwater heaters out of service (FHOOS), coastdown, or any combination thereof. Other EOD/EOOS conditions require no OLMCPR penalty. The impact of SLO is applied to the SLMCPR.

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 GE9 and GE10 fuel.

Table 2.1 Quad Cities Unit 2 Cycle 16 Base Case ΔCPRs at Rated Power With TSSS Insertion Times

Transient	ΔCPR		
	GE9	GE10	ATRIUM-9B Offset
Load Rejection No Bypass			
100%P / 108%F	0.41	0.37	0.33
100%P / 100%F	0.39	0.34	0.31
Feedwater Flow Controller Failure			
100%P / 108%F	0.42	0.38	0.35
100%P / 100%F	0.39	0.36	0.33
Loss of Feedwater Heating (LFWH)	*	*	*

* Analysis of the LFWH is the responsibility of ComEd for Quad Cities Unit 2 Cycle 16.

Table 2.2 Quad Cities Unit 2 Cycle 16 MCPR Operating Limit and Maximum Pressurization Summary

MCPR Operating Limit*

Transient	OLMCPR for Base Case / EOD/EOOS†		
	GE9	GE10	ATRIUM-9B Offset
Feedwater Controller Failure (100%P / 108%F - TSSS)	1.53 / 1.55	1.49 / 1.59	1.46 / 1.49

Maximum Pressurization (psig)

Transient	Steam Dome	Lower Plenum	Steam Lines
MSIV Closure Without Position Scram (ASME) (100%P / 87%F, Base Case)	1330	1353	1329
MSIV Closure Without Position Scram (ASME) (100%P / 100%F, EOD/EOOS)	1333	1359	1333

* 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.

† Fuel-dependent cycle-specific OLMCPR penalties of 0.02 (GE9), 0.10 (GE10) and 0.03 (ATRIUM-9B offset) are added to support EOD/EOOS operation with FFTR, FHOOS, coastdown, or any combination thereof. Other EOD/EOOS conditions require no OLMCPR penalty.

Table 2.3 Quad Cities Unit 2 Cycle 16 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%P / 108%F	653	132	1296 / 1262
100%P / 100%F	604	130	1296 / 1265
Feedwater Flow Controller Failure			
100%P / 108%F	634	135	1185 / 1150
100%P / 100%F	586	134	1182 / 1150
MSIV Closure ASME Analysis			
100%P / 108%F	331	130	1351 / 1323
100%P / 100%F	327	128	1351 / 1325
100%P / 87%F	316	125	1353 / 1330

* 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 ASME Events*
- Up to 40% TIP Strings Out of Service (TIPOOS)§

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* 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 2000 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 16 is based on differences between principal transient analysis parameters used for Quad Cities Unit 2 Cycle 15 and Quad Cities Unit 2 Cycle 16. Differences between the QC2C16 plant parameters (Reference 12) and the QC2C15 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 16 analyses are identified in Reference 2.

Table 3.1 Quad Cities Unit 2 Cycle 16 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.	—
Elevation at Top of Fuel Pellets, in (ATRIUM-9B: 360.313 to 360.553) (GE9: 360.313 to 361.553)	None.	—
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	—
Main Steamline Safety/ Relief Valve Flange Length, ft and diameter, in (new parameters,1.6 ft [length], 6 in [diameter])	The addition of the RV, SV, and SRV flanges results in a slight reduction in the flow coefficients for the valves. This has no impact on thermal margin analyses and increases overpressurization analyses by approximately 0.2 psi.	Parameter change will not result in new limiting events. ASME overpressurization events are evaluated on a cycle-specific basis (see Sections 5.0 and 7.0).
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 Appendix B establish the OLMCPR penalty for operation with BPVOOS.

Table 3.1 Quad Cities Unit 2 Cycle 16 Evaluation of Plant Parameter Changes on Disposition of Events
(Continued)

Parameter Change (From/To)	Impact	Resolution
Control Rod Position Versus Scram Time (NSS times replaced with ITS times)	None. Neither NSS nor ITS are supported for Cycle 16.	—
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.	—
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).

Table 3.1 Quad Cities Unit 2 Cycle 16 Evaluation of Plant Parameter Changes on Disposition of Events
(Continued)

Parameter Change (From/To)	Impact	Resolution
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. The reduction in the number of available safety valves also leads to the limiting state point changing from 100%P/108%F to 100%P/87%F for base case operation.	Parameter change will not result in new limiting events. All limiting events are evaluated on a cycle-specific basis (see Sections 5.0 and 7.0).
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.
Turbine Bypass Valve Operation Not Credited for Maximum Pressurization Events (new parameters)	None. This assumption was included in the TCV and TSV ASME analyses for QC2C15 even though it was not explicitly stated in the parameters document. For the MSIV closure analyses, the MSIV closes before the bypass valve can operate. Therefore, this change will not impact ASME transient analyses.	—

Table 3.1 Quad Cities Unit 2 Cycle 16 Evaluation of Plant Parameter Changes on Disposition of Events
(Continued)

Parameter Change (From/To)	Impact	Resolution
Combined Steam Flow Limiter Setting, %NBR (105 to 115)	None. A higher setting would affect bypass valve operation. The bypass valve 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 a 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 which were performed to determine the full power MCPR operating limits for Cycle 16 of Quad Cities Unit 2.

4.1 *Design Basis*

The plant transient analyses for Quad Cities Unit 2 Cycle 16 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 an 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 2 Cycle 16 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 GE9/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 2 Cycle 16, 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. ASME overpressure event 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%/87%F state point is nonlimiting for all possible operating domains

* 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.

including standard operation and all EOD/EOOS combinations. Therefore, as indicated in Reference 2, thermal margin evaluations are not performed at 100%P/87%F 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 0.70 second 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 59.6 seconds for the ATRIUM-9B offset fuel. The TSV and TCV become fully closed at approximately 59.0 seconds.

4.3.3 Loss of Feedwater Heating

For the Quad Cities Unit 2 Cycle 16 reload, the analysis of the LFWH transient is the responsibility of ComEd.

4.4 *MCPR Safety Limit*

The MCPR safety limit (SLMCPR) for Quad Cities Unit 2 Cycle 16 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 2000 effective full power hours (EFPH) as discussed in Reference 22.

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

- Cycle 16 will not use channels for more than one fuel bundle lifetime. The GE9 fuel uses CarTech channels, 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 GE9 and GE10 fuel and 48,000 MWd/MTU for ATRIUM-9B offset fuel, based on the maximum bundle exposures at the end of Cycle 16.
- 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 16 exposure of 16,050 MWd/MTU. The radial power distribution corresponding to this exposure is shown in Figure 4.9.

The limiting local power distribution for the Cycle 16 SPC fuel types with channel bow are 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 a SLO limit of 1.12 supports all normal and EOD/EOOS conditions identified in Table 2.4. The Quad Cities Tecnnical Specification SLMCPR safety limit of 1.11 for TLO and 1.12 for SLO are applicable.

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 neutronic 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 GE9/GE10 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 GE9/GE10 fuel, and a full core of GE9/GE10 fuel are essentially equivalent.

**Table 4.1 Quad Cities Unit 2 Cycle 16
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	85.5 Mlb/hr	85.6 Mlb/hr
Core Bypass Flow*	12.5 Mlb/hr	12.4 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

* 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 2
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)	1195 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 16 core using RODEX2 at rated conditions.

Table 4.2 Quad Cities Unit 2
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 Stroke	 1.85 sec 10.0 sec 250 msec
Relief Valve Opening Delay Closing Delay Stroke	 1.85 sec 10.0 sec 250 msec
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 Operation (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)

* 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 ASME overpressurization event, SRV function is not credited.
§ One relief valve at the lowest set point is not credited.
** Reference 25.

Table 4.2 Quad Cities Unit 2
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	500 msec
Steam Flow	250 msec
Feedwater Flow	250 msec
Level	1.05 sec
Feedwater Control Mode	Single Element†
Water Level Controller	
Proportional Gain	25%/ft
Pressure Regulator Settings	
Lead	470 msec
Large Lag	7.2 sec
Small Lag	594 msec
Gain	3.33%/psid
Bypass Flow Signal Bias	3%
Combined Steam Flow Limiter Setting	115% rated
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 MCPR Safety Limit Analysis

Fuel-Related Uncertainties

Parameter	Source Document	Statistical Treatment
ANFB Correlation* GE9/GE10 ATRIUM-9B Offset	 Reference 9 Reference 10	Convoluted
Radial Peaking Factor	Reference 22†	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	15.2§	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	2.50	Convoluted
Core Power	MWt	3860§	—	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 2000 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 2
Steam Dome Pressure - Analysis Basis

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

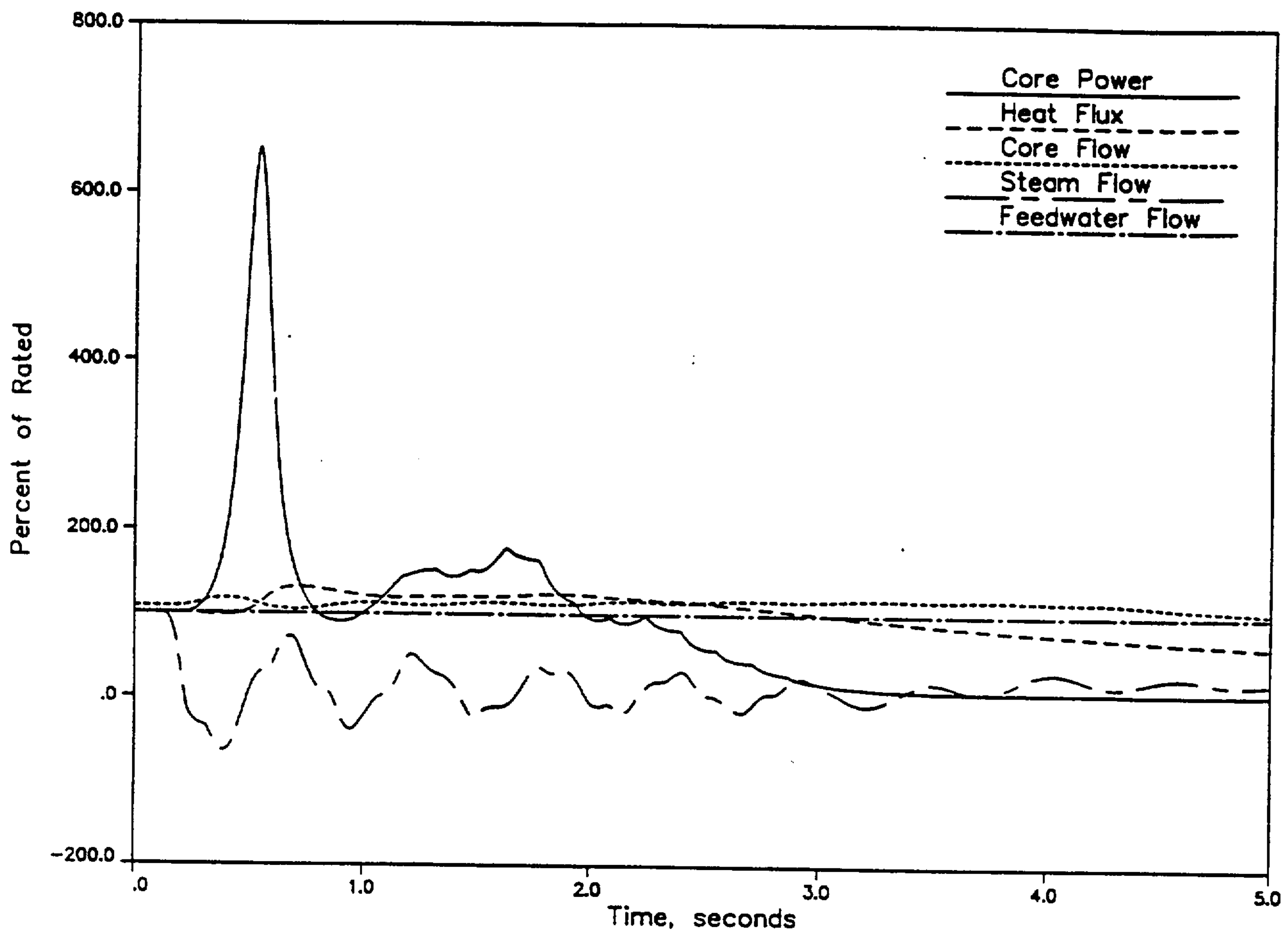


Figure 4.1 Load Rejection No Bypass at 100/108 - Key Parameters

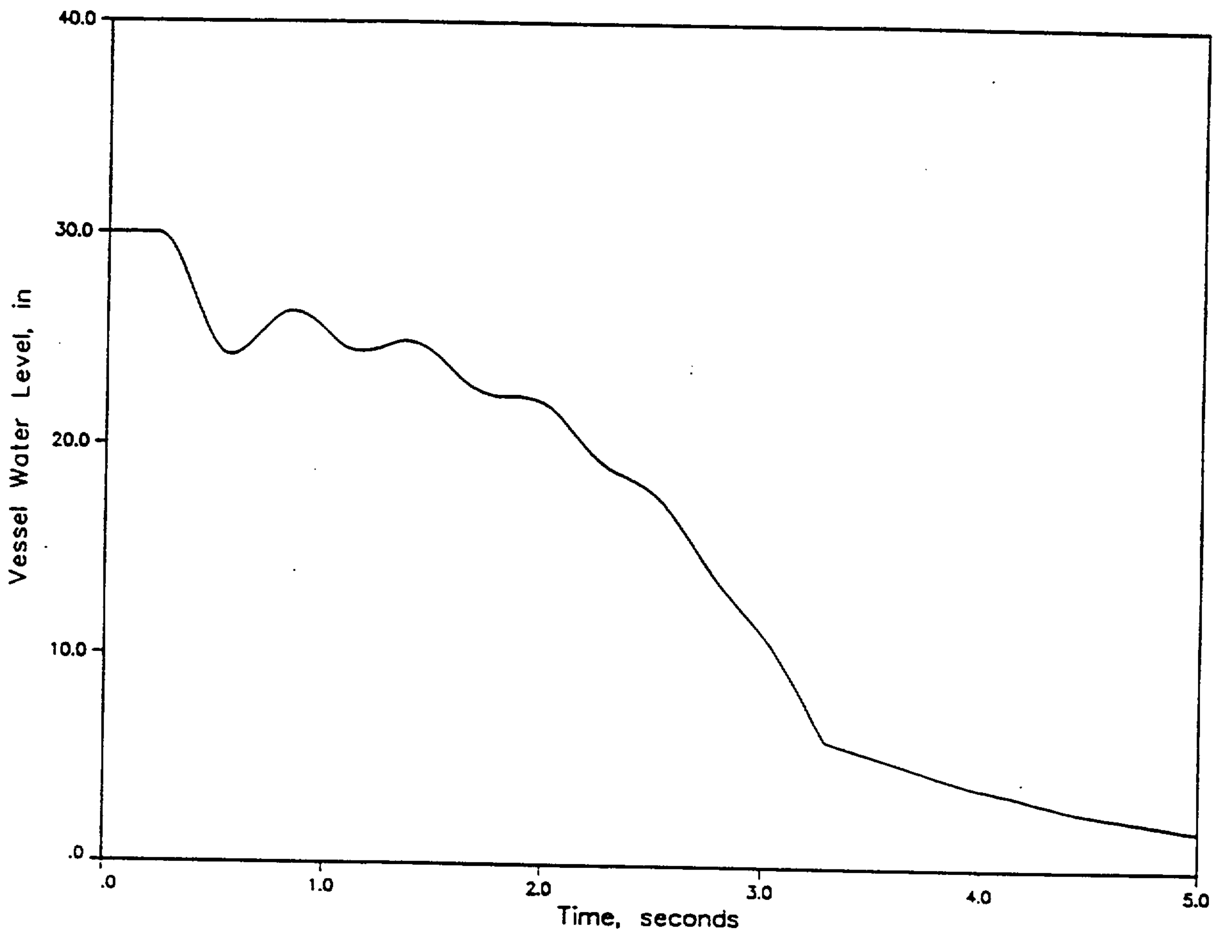


Figure 4.2 Load Rejection No Bypass at 100/108 - Vessel Water Level
(Referenced to Instrument Zero)

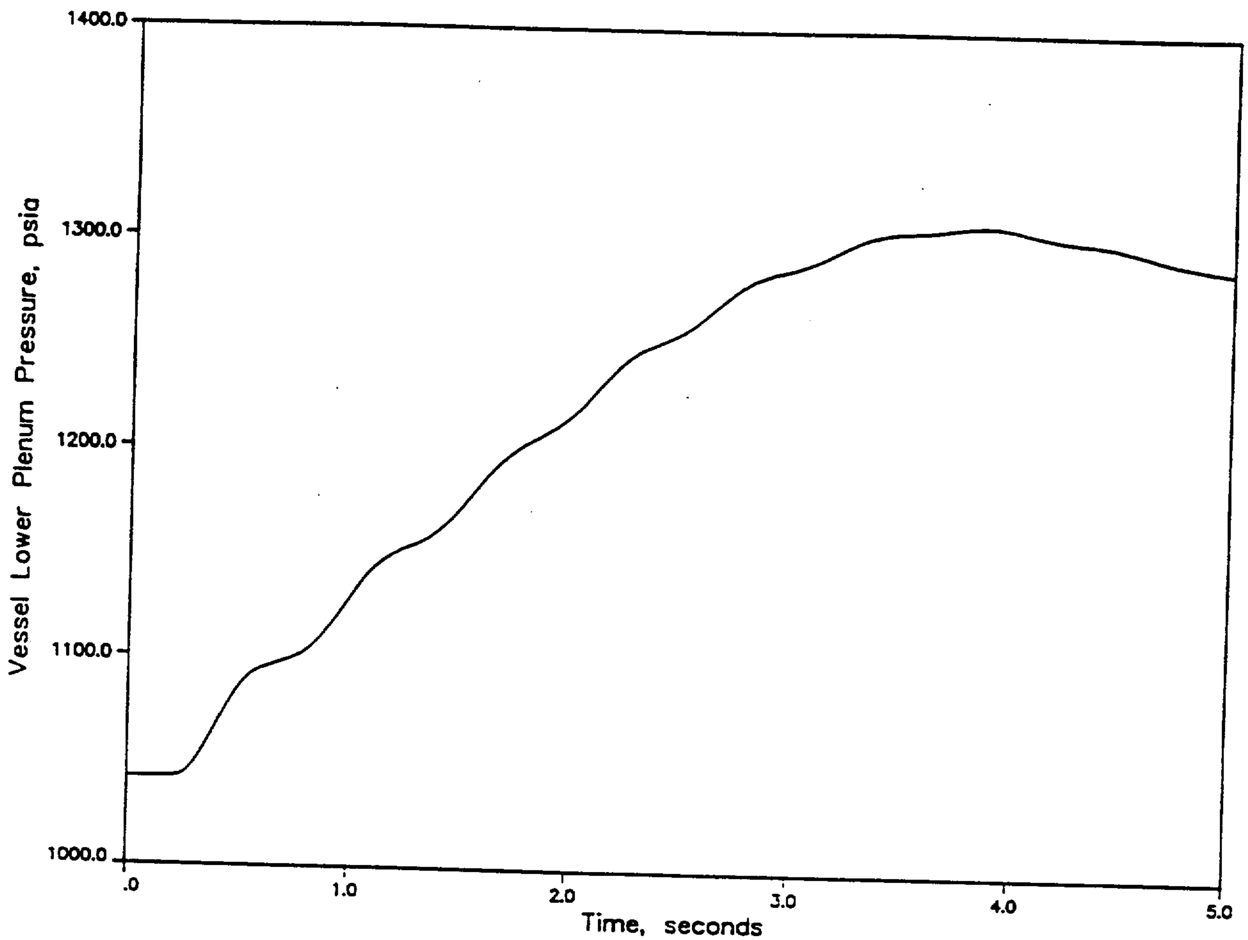


Figure 4.3 Load Rejection No Bypass at 100/108 - Vessel Pressure Response

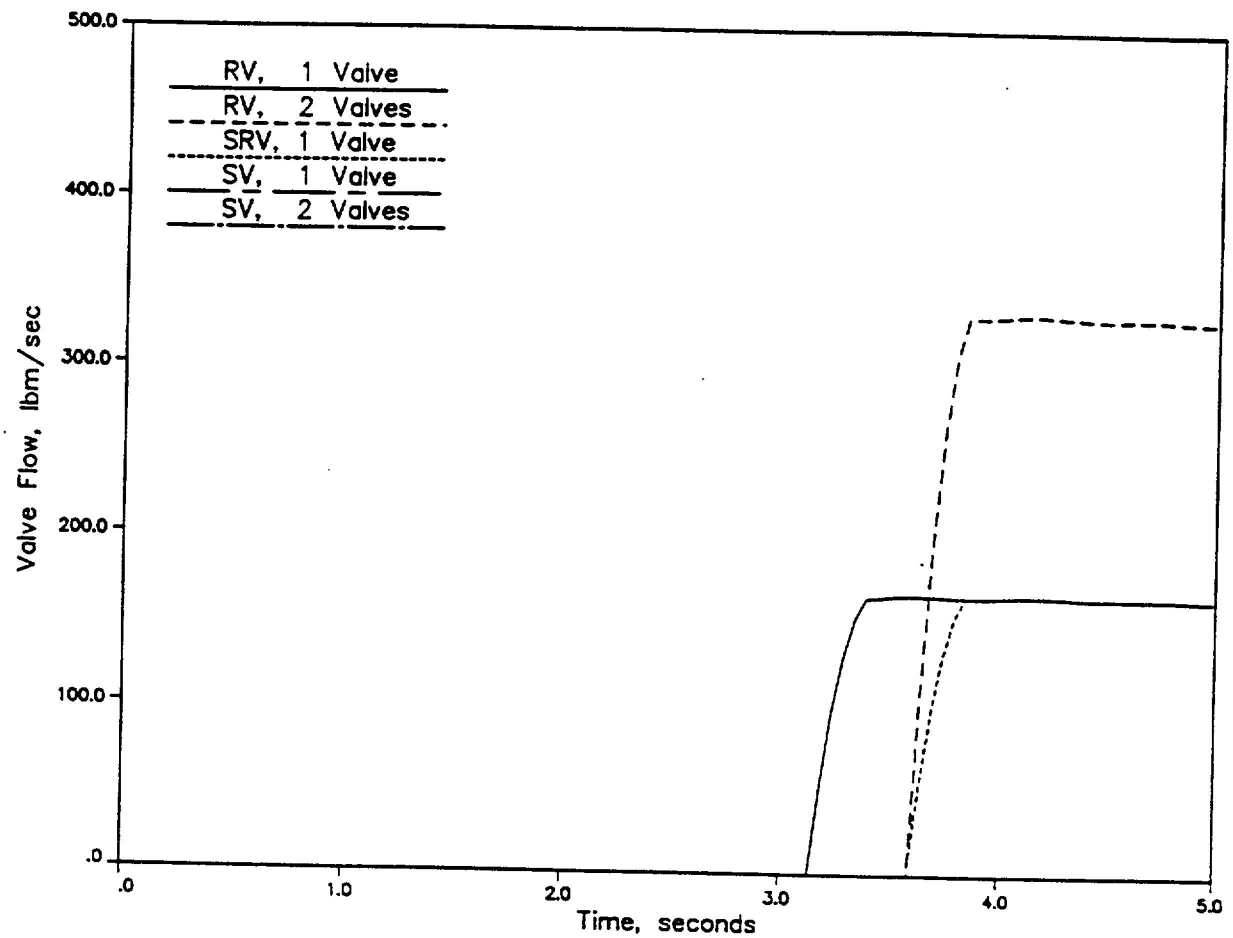
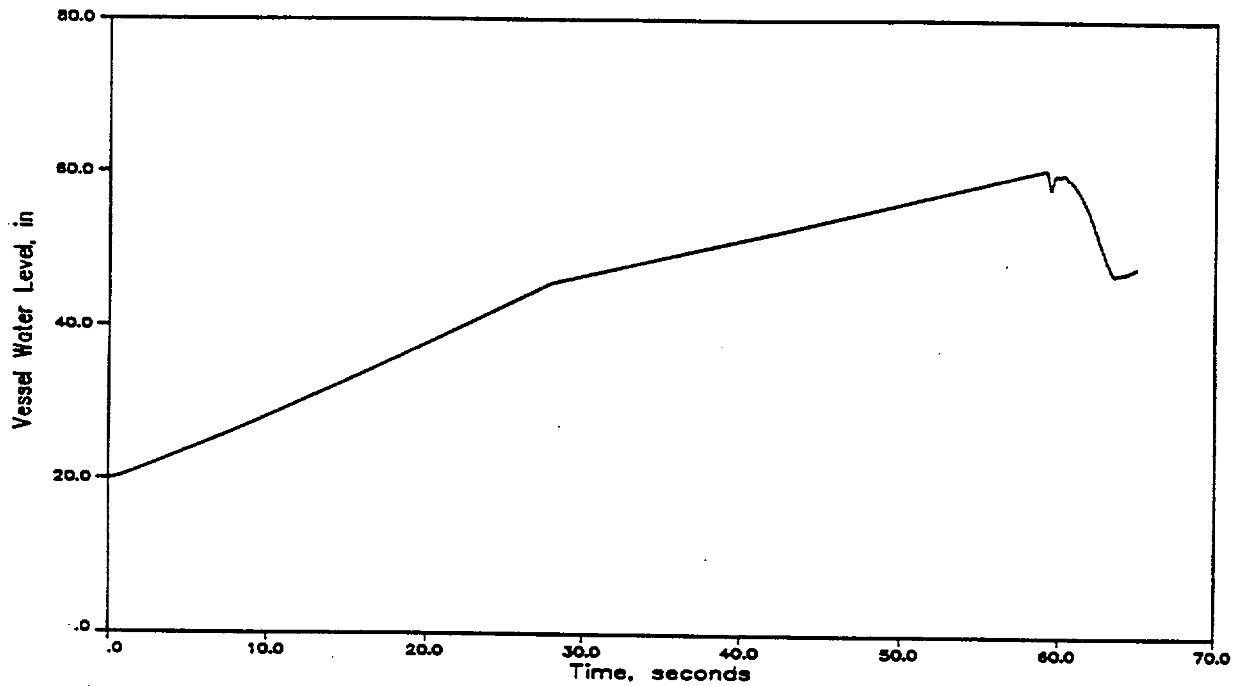


Figure 4.4 Load Rejection No Bypass at 100/108 - Safety/Relief Valve Flows



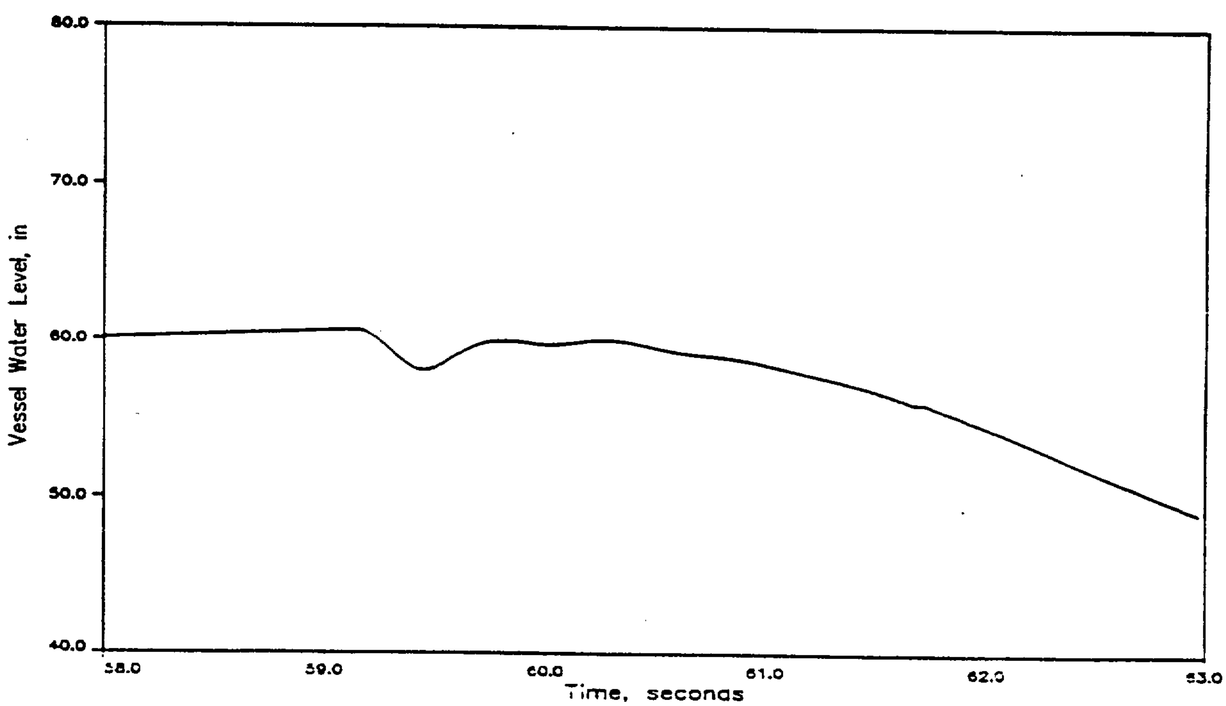
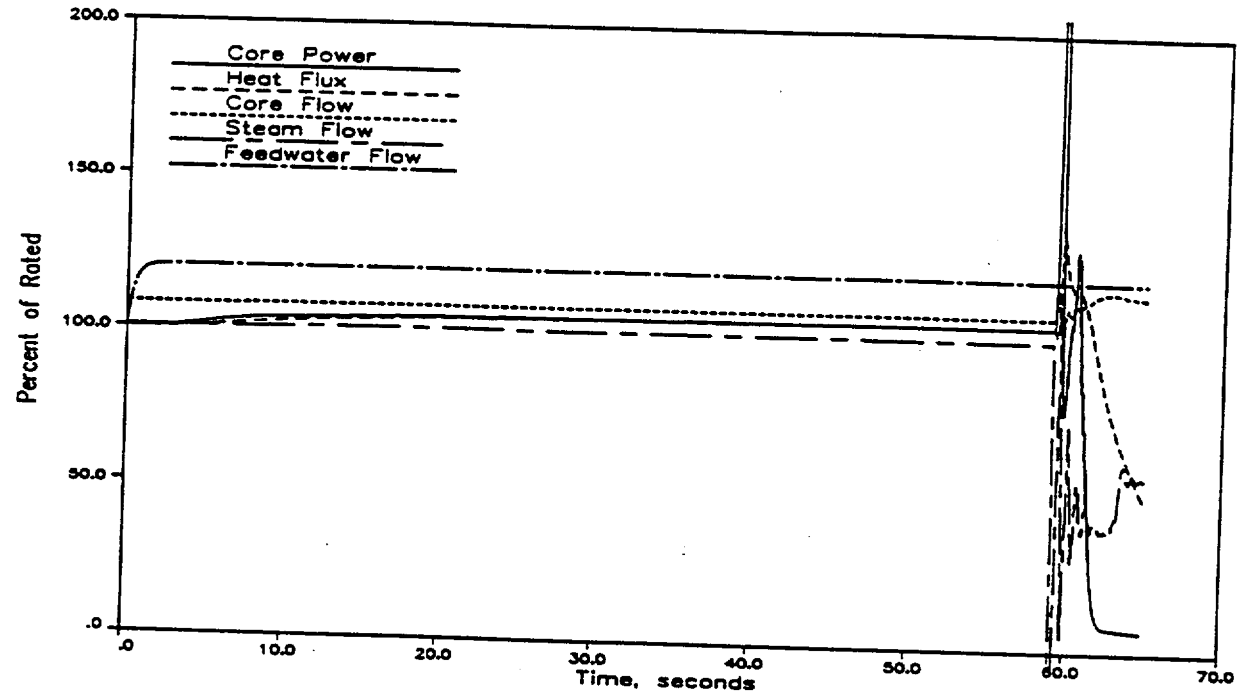


Figure 4.6 Feedwater Controller Failure at 100/108 - Vessel Water Level
(Referenced to Instrument Zero)



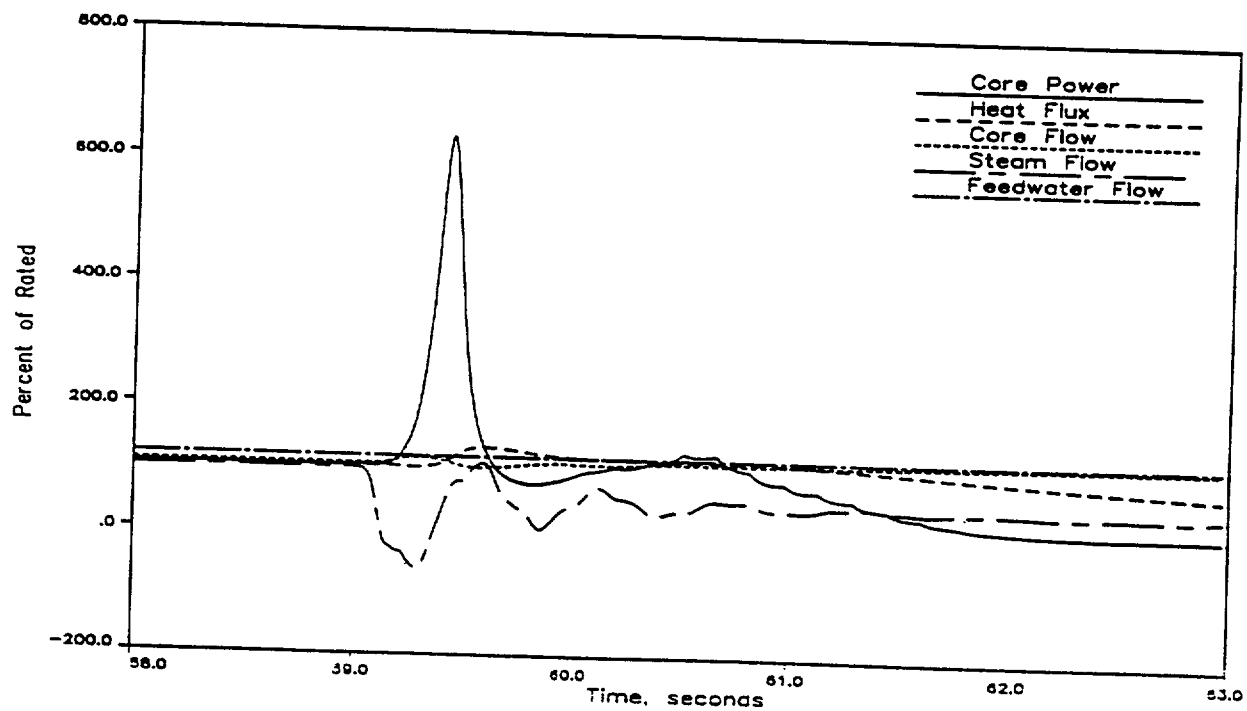


Figure 4.5 Feedwater Controller Failure at 100/108 - Key Parameters

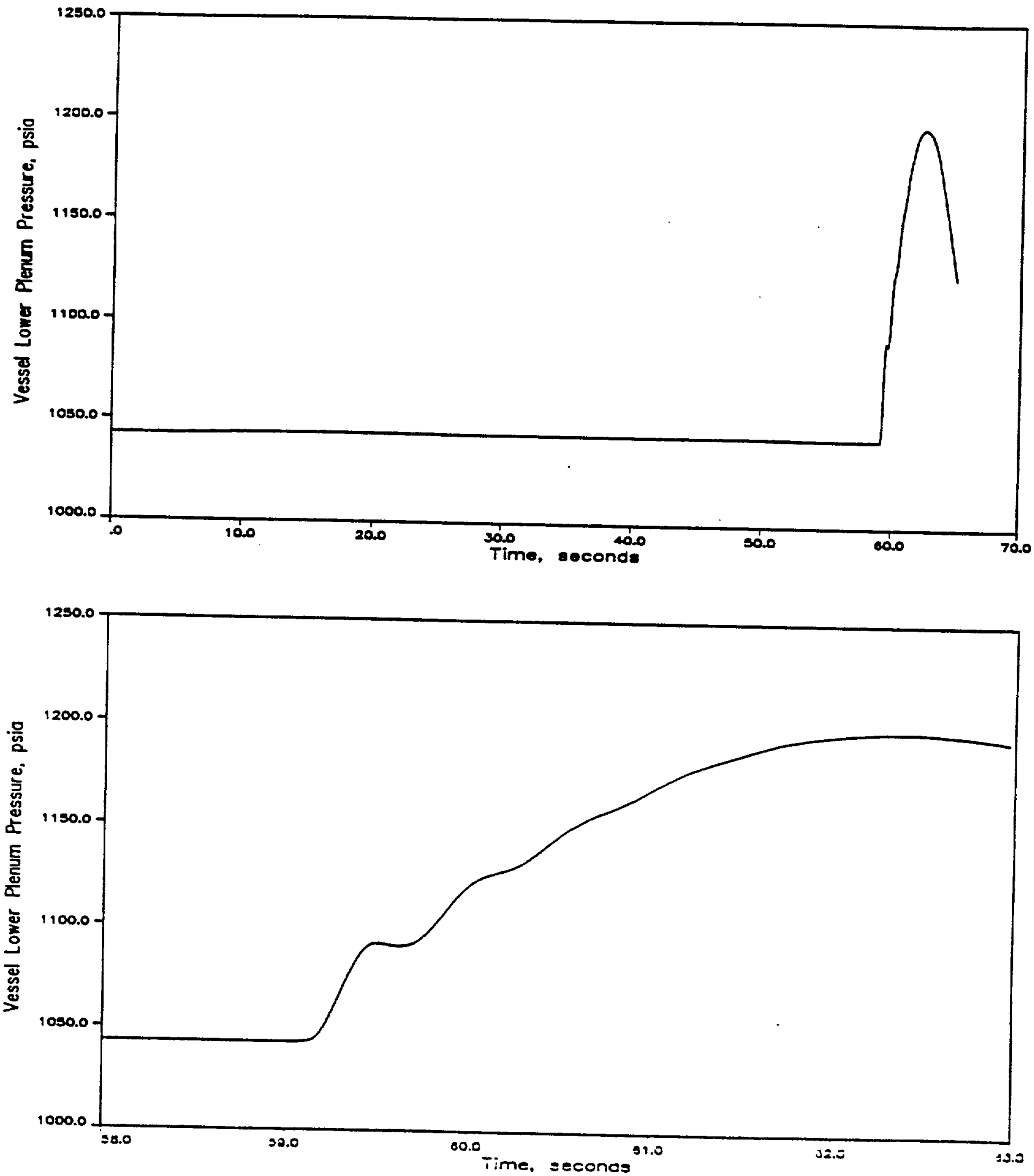
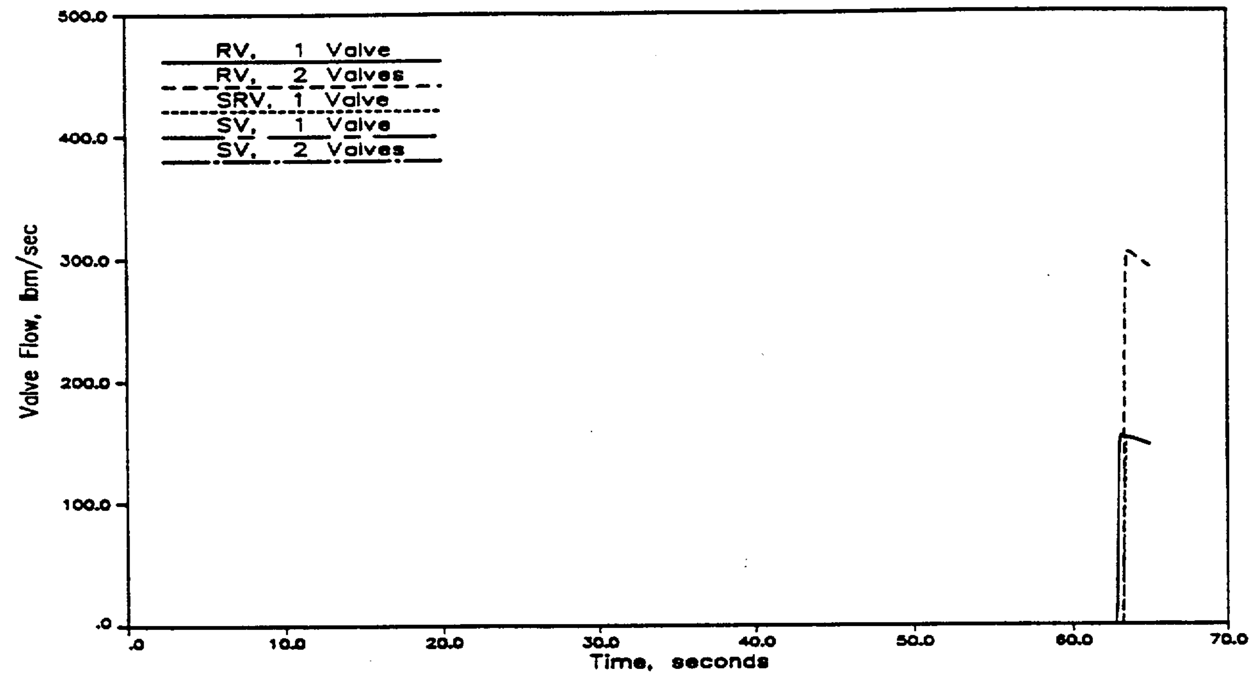


Figure 4.7 Feedwater Controller Failure at 100/108 - Vessel Pressure Response



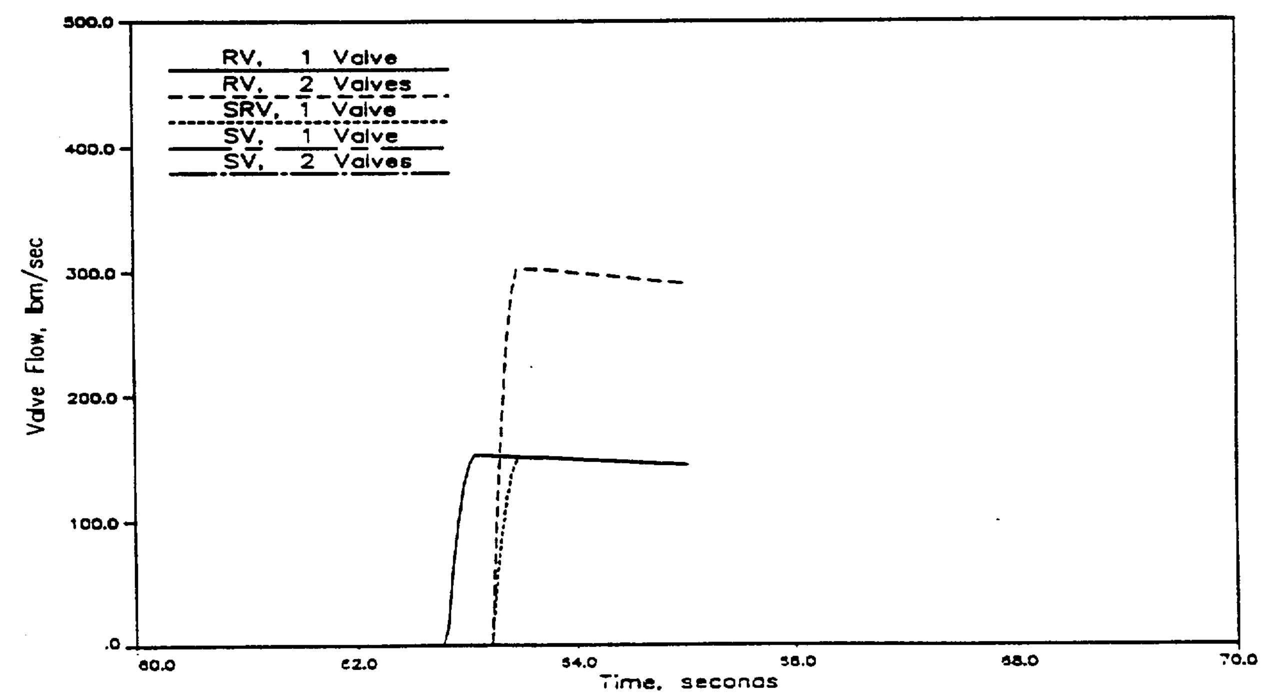


Figure 4.8 Feedwater Controller Failure at 100/108 - Safety/Relief Valves

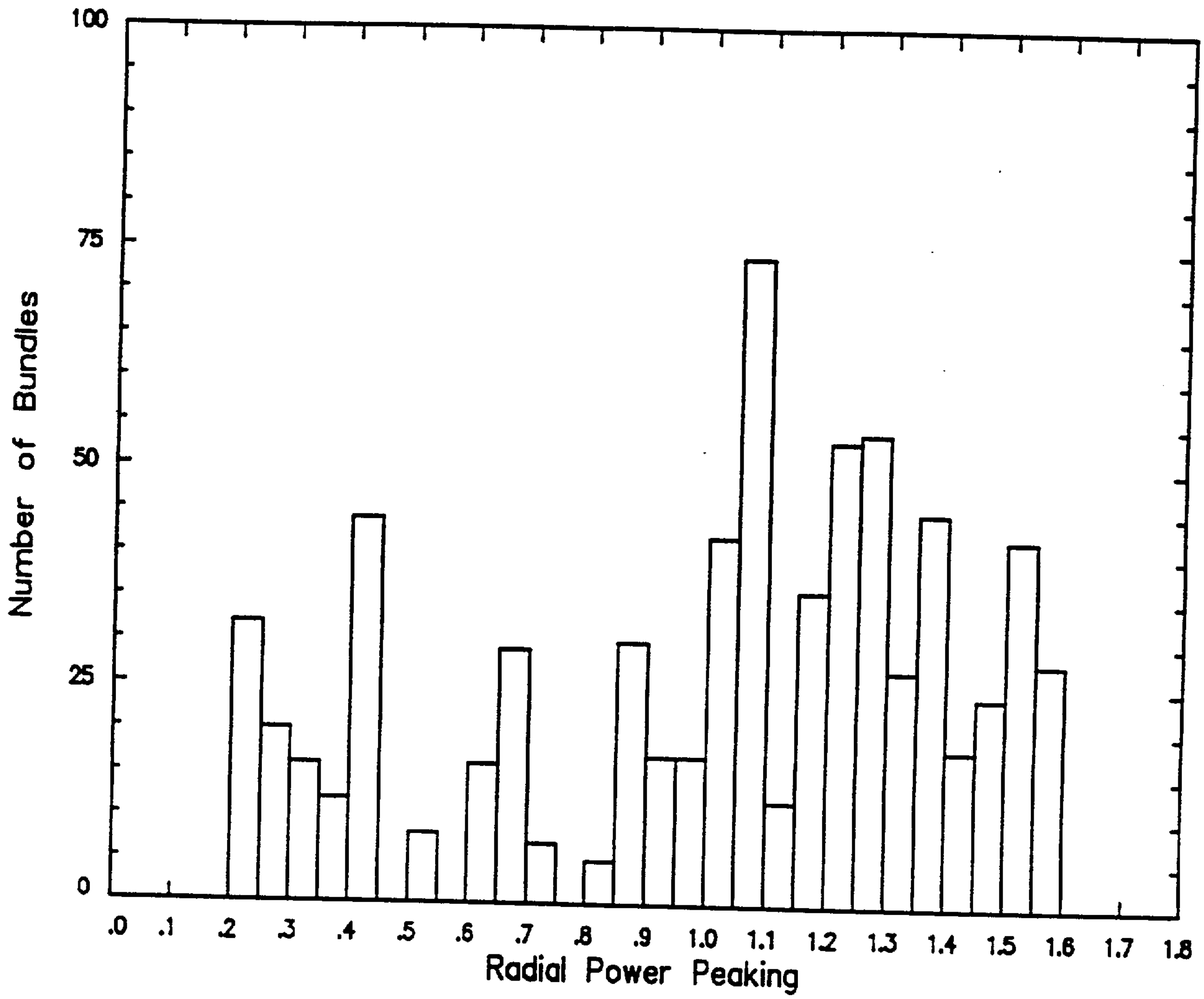


Figure 4.9 Radial Power Distribution for SLMCPR Determination

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				1.034	0.962	1.005
1.108	1.075	1.117		Internal Water Channel		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 2 Cycle 16
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

1.006	1.021	1.040	1.047	1.115	1.101	1.100	0.994	0.979
1.021	0.983	1.053	1.030	0.960	1.067	0.931	0.998	0.972
1.040	1.053	0.977	1.121	1.126	1.082	1.026	0.876	1.052
1.047	1.030	1.121				1.038	0.966	1.007
1.115	0.960	1.126		Internal Water Channel		1.073	0.827	0.984
1.101	1.067	1.082				1.029	0.921	0.964
1.100	0.931	1.026	1.038	1.073	1.029	0.821	0.912	0.961
0.994	0.998	0.876	0.966	0.827	0.921	0.912	0.810	0.917
0.979	0.972	1.052	1.007	0.984	0.964	0.961	0.917	0.823

Figure 4.11 Quad Cities Unit 2 Cycle 16
Safety Limit Local Peaking Factors With Channel Bow at Assembly
Exposure of 25,000 MWd/MTU (SPCA9-381B-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 four 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 three 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.

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.

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 ASME events analyzed for Cycle 16. Cycle 16 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/87% flow. The maximum vessel pressure (at the lower plenum) of 1353 psig was observed at 7.2 seconds. The maximum steam line pressure of 1329 psig and the maximum steam dome pressure of 1330 psig were observed at 7.4 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 87% flow.

Of the EOD/EOOS operating conditions described in Table 2.4, maximum pressurization evaluations are performed with only coastdown and combined ICF/coastdown conditions. As demonstrated in References 13, 20, and 21, all other EOD/EOOS conditions are nonlimiting for maximum pressurization events. Maximum pressurization analysis results provided in Section 7.0 confirm that the limiting EOD/EOOS MSIV closure transient has approximately 16 psi margin to the vessel pressure limit and 12 psi margin to the steam dome pressure limit.

**Table 5.1 Quad Cities Unit 2 Cycle 16
Results Summary of Base Case ASME Overpressurization
Analyses With TSSS Insertion Times**

Transient	Maximum Pressurization (psig)	
	Steam Dome	Lower Plenum
MSIV Closure		
100%P / 108%F	1323	1351
100%P / 100%F	1325	1351
100%P / 87%F	1330	1353
TCV/TSV Closure		
100%P / 108%F	1318	1346
100%P / 100%F	1320	1346
100%P / 87%F	1323	1347

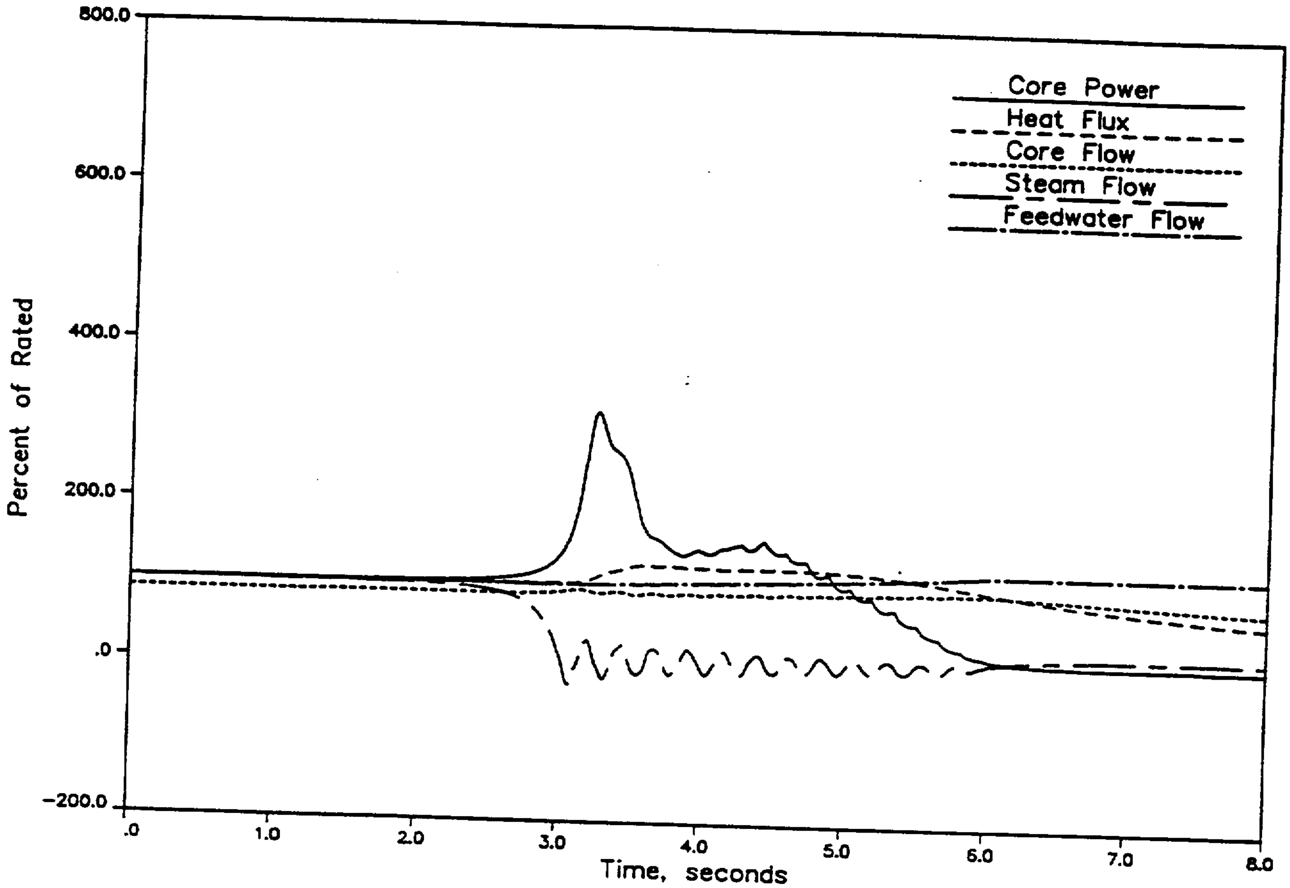


Figure 5.1 MSIV Closure at 100/87 - Key Parameters

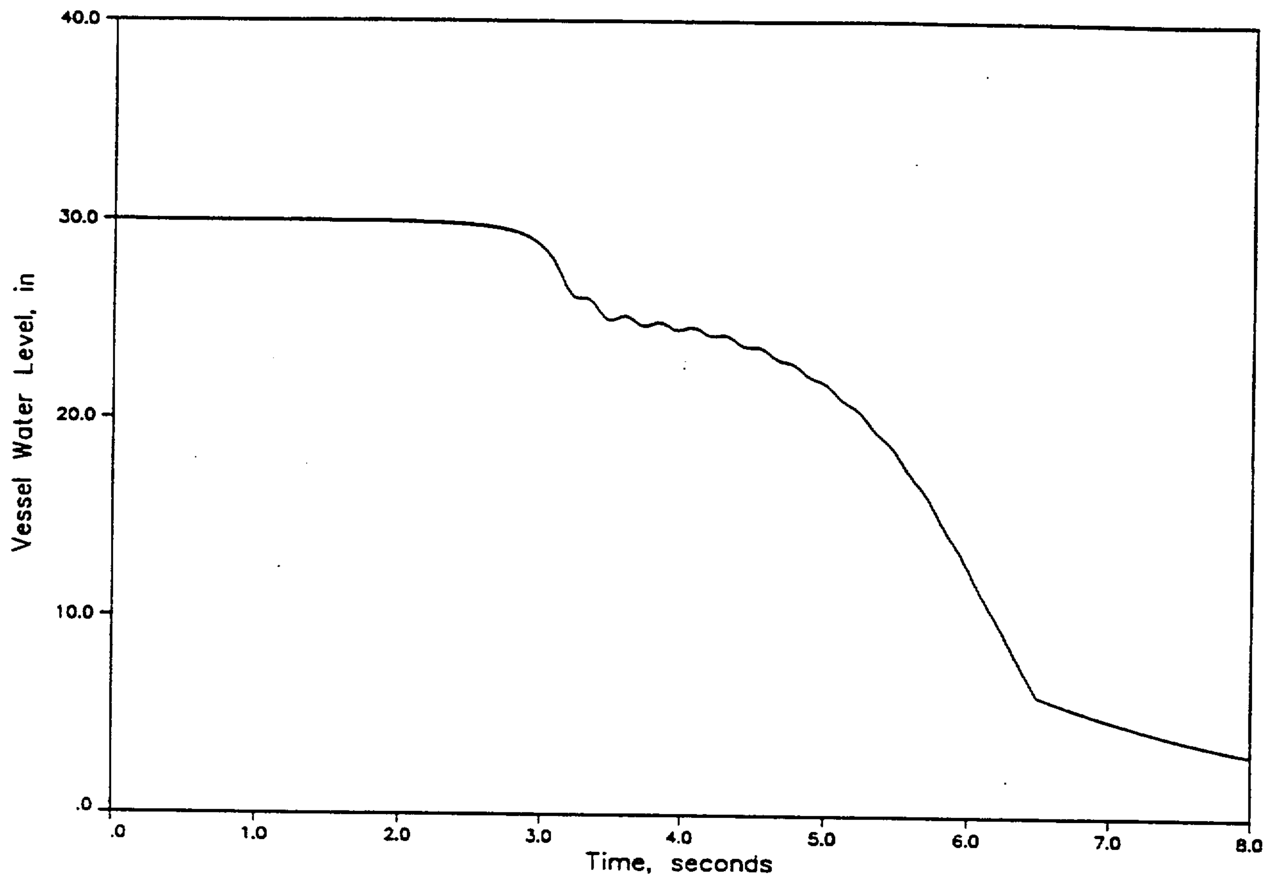


Figure 5.2 MSIV Closure at 100/87 - Vessel Water Level
(Referenced to Instrument Zero)

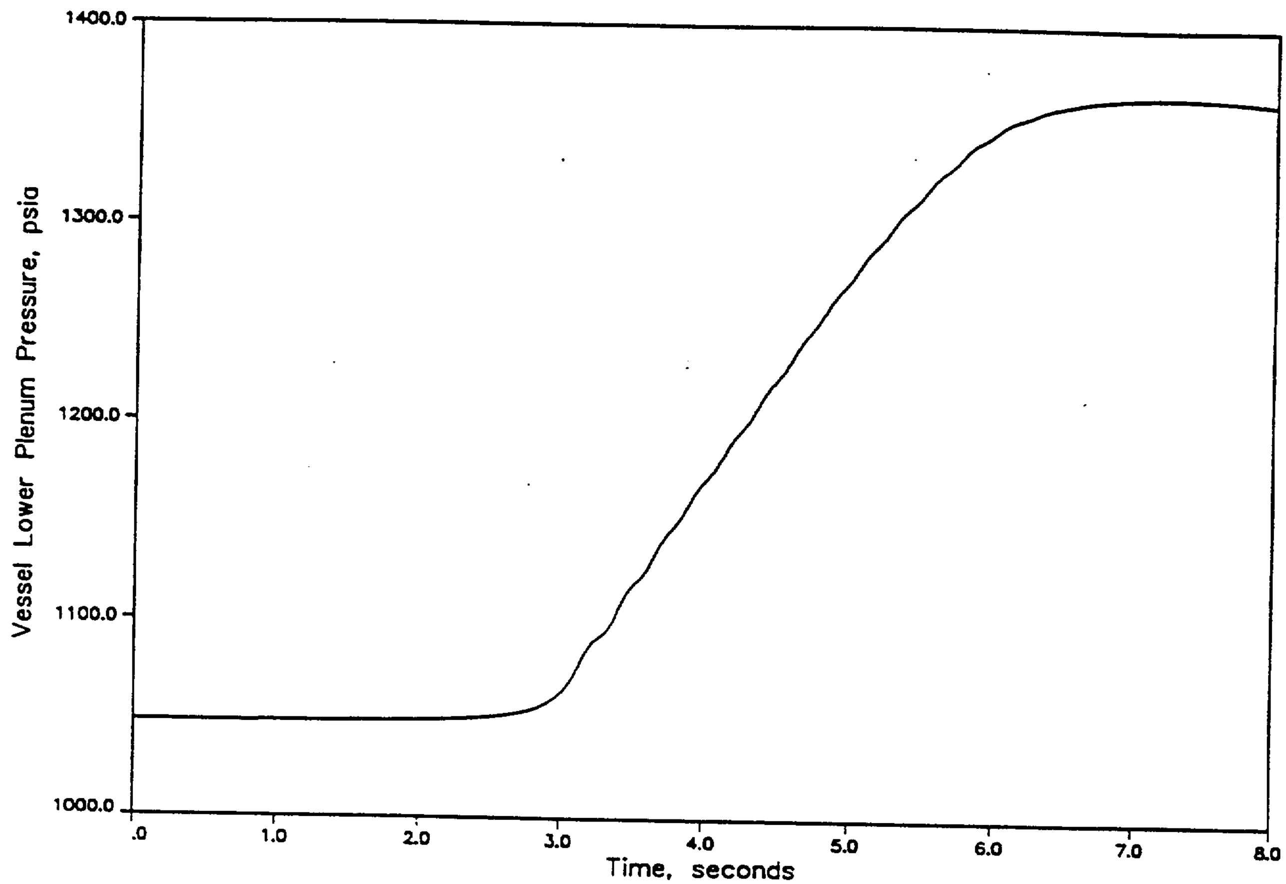


Figure 5.3 MSIV Closure at 100/87 - Vessel Pressure Response

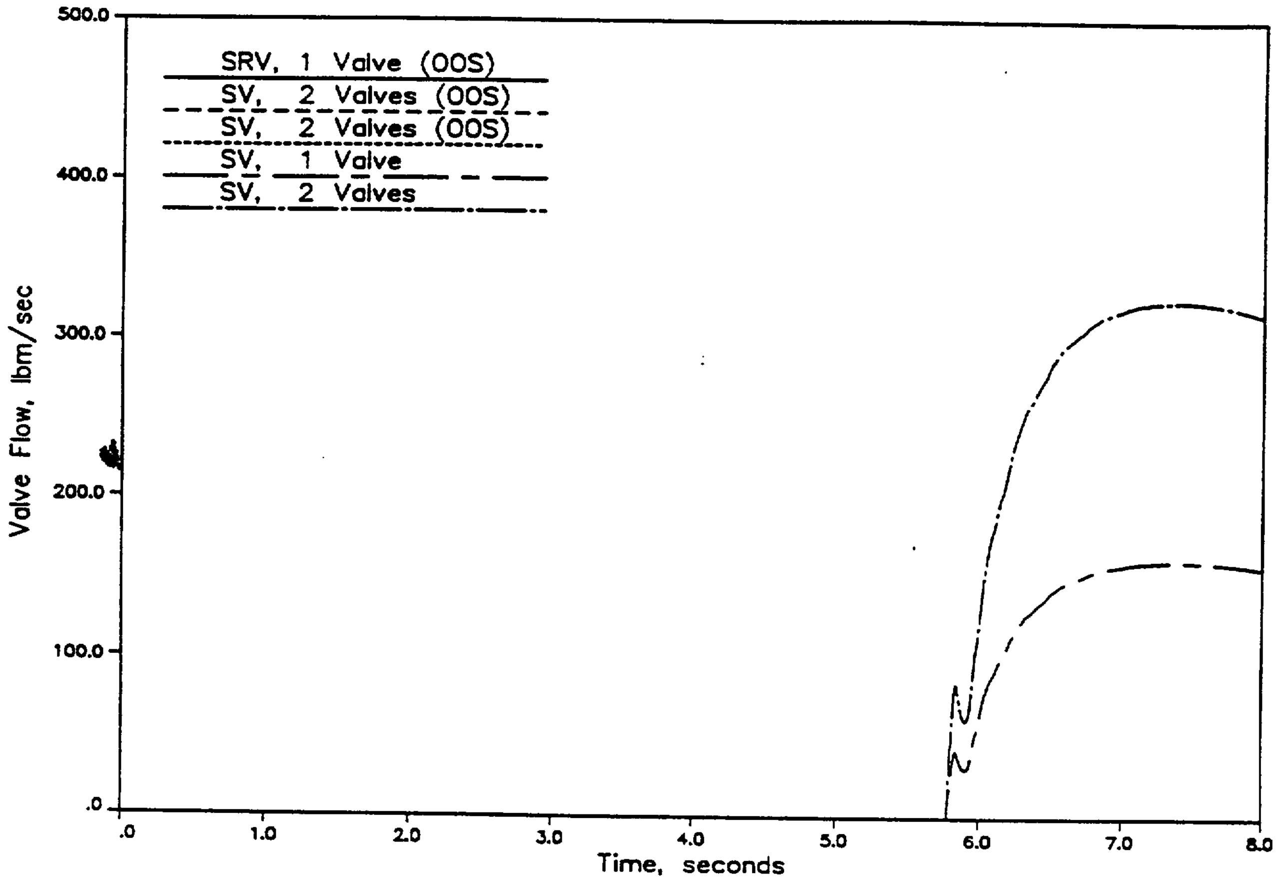


Figure 5.4 MSIV Closure at 100/87 - Safety/Relief Valve Flows

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 16 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 QC2C16.

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%/87%F 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%P/87%F 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 QC2C16.

Maximum pressurization transients were evaluated at all potentially limiting flow conditions that may be obtained at rated core power. For 100% power cases, analyses with 100% and 108% rated flow are bound by analyses with 87% rated flow. Due to modified analysis assumptions specified in Reference 12 (specifically the availability of only three safety valves), maximum pressurization transients are limiting for QC2C16 at reduced core 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 GE9/GE10 and ATRIUM-9B offset equilibrium cores as well as transition cores from GE9/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 48% power/30% flow to 125% power/110% flow for the MFC analysis. The flow run-up path for the AFC analysis begins at 37%power/30% flow and ends at 100% power/108% flow. Linear extrapolation of the 40% and 30% core flow XCOBRA analysis results is used to obtain MCPR limits below 30% of rated core flow.

The $MCPR_f$ limits are shown in Figures 6.1 and 6.2 for the limiting fuel in Quad Cities Unit 2 Cycle 16 for the automatic flow control event. Figure 6.3 details $MCPR_f$ limits pertaining to the manual flow control event for the limiting fuel in Quad Cities Unit 2 Cycle 16. The analysis results provide for operation up to EOFP and operation with EOD/EOOS. The cycle-specific MCPR limit for Quad Cities Unit 2 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 is to protect 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.

The automatic flow control analyses were performed to support the base case OLMCPRs as well as the EOD/EOOS OLMCPRs (refer to Table 2.2).

The $MCPR_f$ penalty described in Reference 18 has been applied to the GE9 and GE10 $MCPR_f$ limits shown in Figures 6.1–6.3. The penalty is a function of core flow with a value of 0.0 at 100% rated and increasing linearly to 0.05 at 40% rated. The penalty is linearly extrapolated for flows less than 40% of rated. Analysis results in Tables 6.3–6.5, with the addition of the penalty, are bound by the $MCPR_f$ limits of Figures 6.1–6.3.

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.

The reduced flow MCPR limits for AFC are presented in Figures 6.1 and 6.2 for the Cycle 16 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 16. 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 6.3 for the Cycle 16 fuel types. When in manual flow control, the cycle-specific MCPR limit for Quad Cities Unit 2 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.

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
GE9 Fuel
(Penalty Not Included)

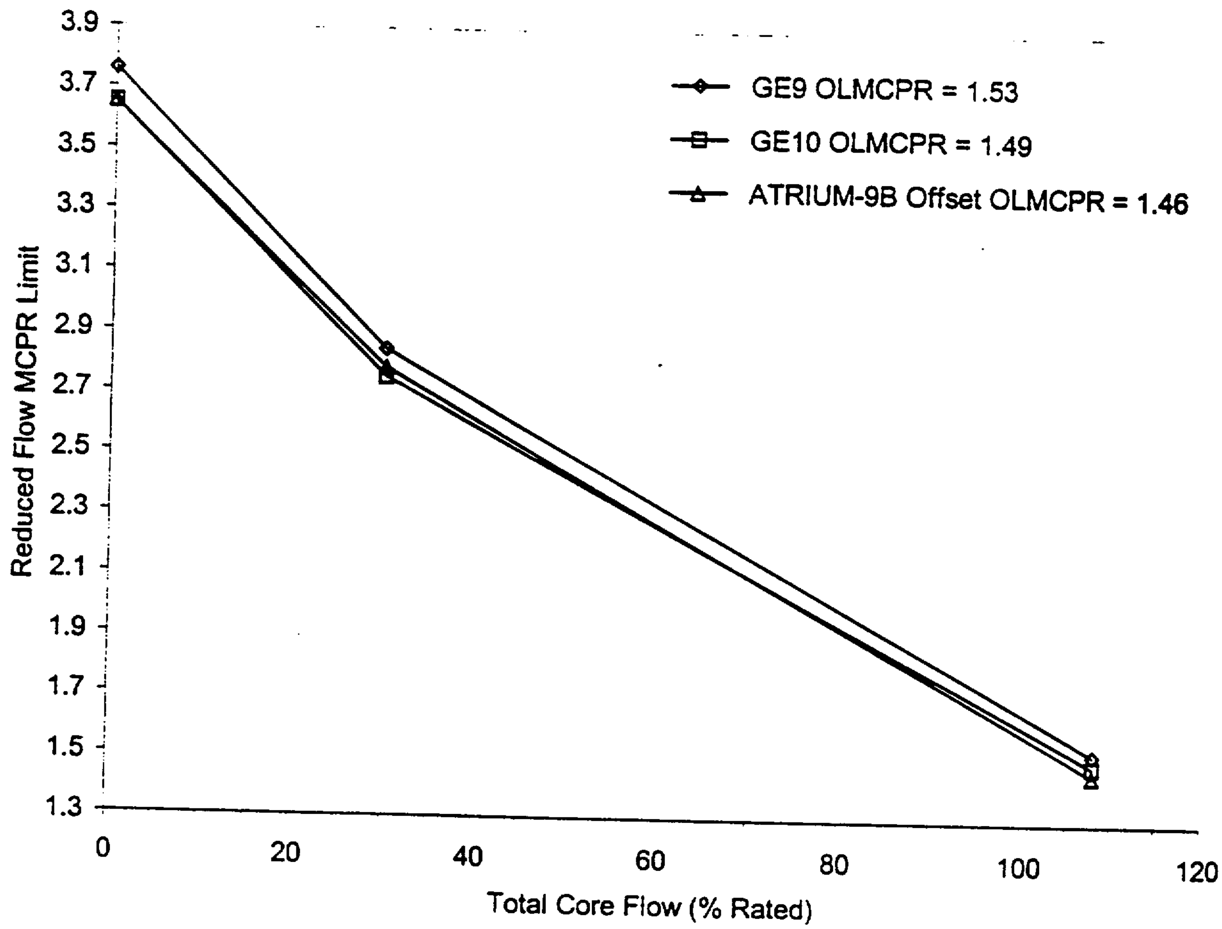
Total Core Flow (% of Rated)	Manual Flow Control	Automatic Flow Control MCPR	
		1.53 OLMCPR	1.55 OLMCPR
110	1.110	—	—
108	—	1.530	1.550
100	1.185	1.617	1.638
90	1.265	1.733	1.756
80	1.351	1.859	1.884
70	1.435	1.993	2.021
60	1.523	2.135	2.165
50	1.620	2.291	2.323
40	1.735	2.476	2.510
30	1.913	2.773	2.809

Table 6.4 Flow-Dependent MCPR Results
GE10 Fuel
(Penalty Not Included)

Total Core Flow (% of Rated)	Manual Flow Control	Automatic Flow Control MCPR	
		1.49 OLMCPR	1.59 OLMCPR
110	1.110	—	—
108	—	1.490	1.590
100	1.183	1.574	1.680
90	1.262	1.685	1.799
80	1.346	1.804	1.929
70	1.428	1.931	2.068
60	1.515	2.066	2.214
50	1.610	2.216	2.374
40	1.723	2.395	2.563
30	1.899	2.684	2.865

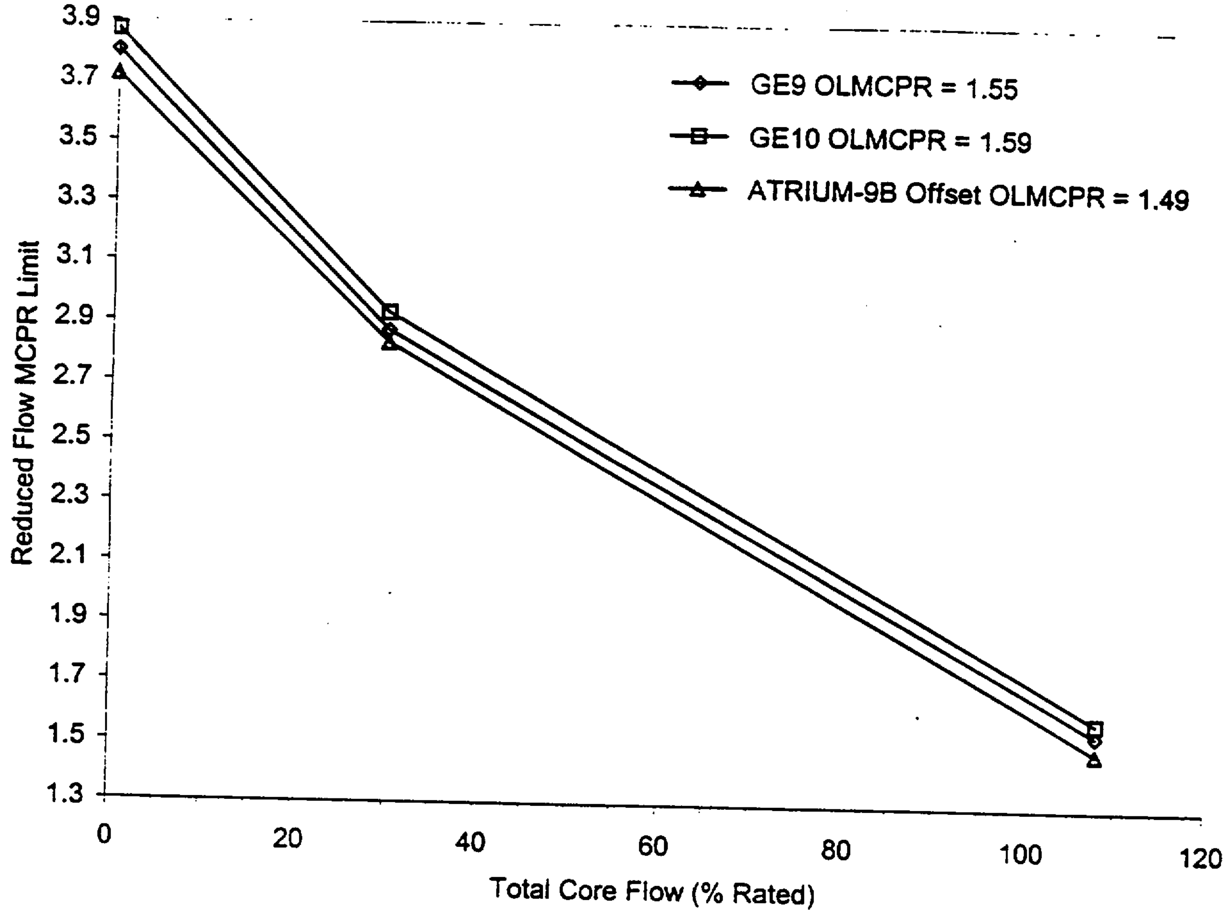
**Table 6.5 Flow-Dependent MCPR Results
ATRIUM-9B Offset Fuel**

Total Core Flow (% of Rated)	Manual Flow Control	Automatic Flow Control MCPR	
		1.46 OLMCPR	1.49 OLMCPR
110	1.110	—	—
108	—	1.460	1.490
100	1.194	1.551	1.582
90	1.290	1.678	1.712
80	1.392	1.815	1.853
70	1.496	1.964	2.006
60	1.602	2.122	2.168
50	1.711	2.290	2.340
40	1.835	2.481	2.535
30	2.017	2.772	2.830



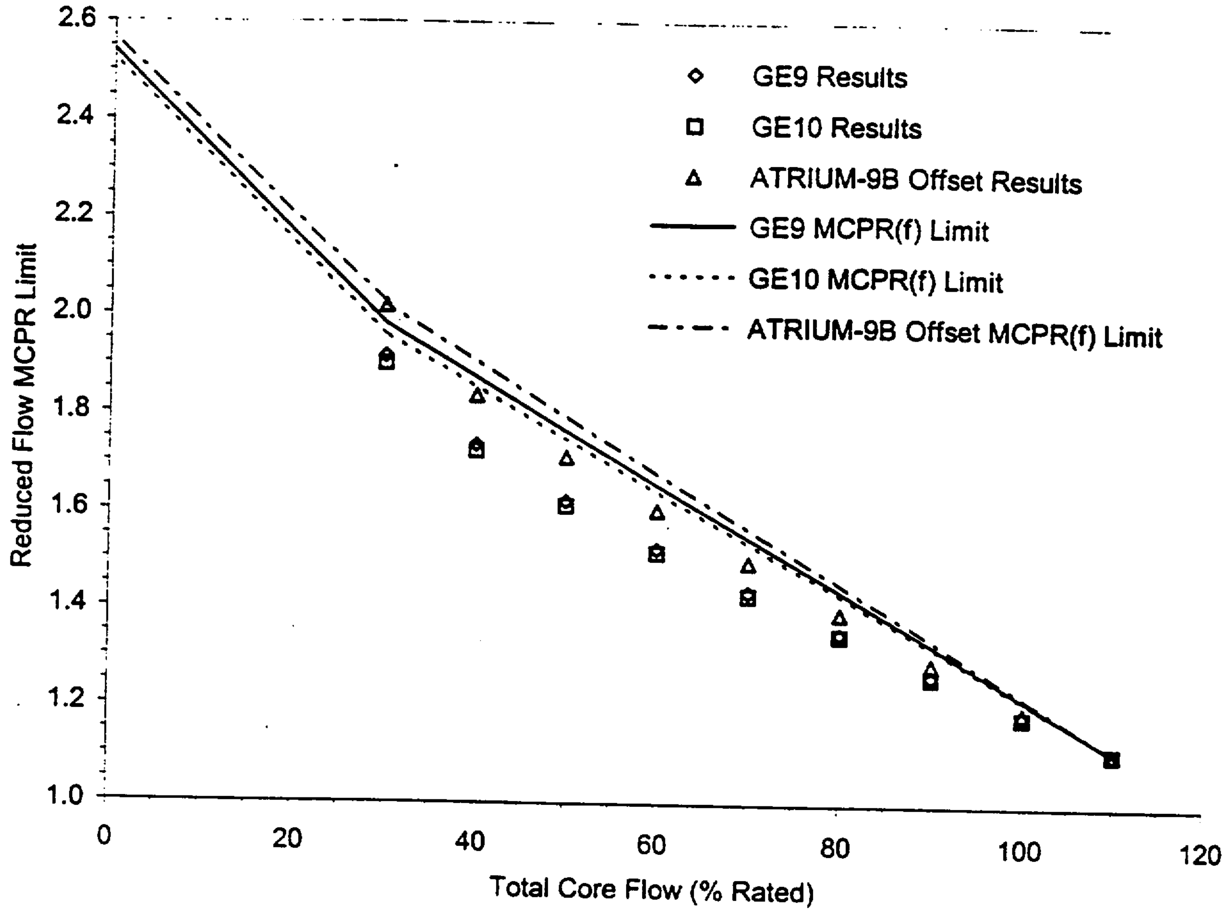
Total Core Flow (% of Rated)	GE9 $MCPR_f$ Limit for OLMCPR=1.53	GE10 $MCPR_f$ Limit for OLMCPR=1.49	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR=1.46
108	1.53	1.49	1.46
30	2.84	2.75	2.78
0	3.76	3.65	3.65

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



Total Core Flow (% of Rated)	GE9 $MCPR_f$ Limit for OLMCPR=1.55	GE10 $MCPR_f$ Limit for OLMCPR=1.59	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR=1.49
108	1.55	1.59	1.49
30	2.87	2.93	2.83
0	3.80	3.87	3.72

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



Total Core Flow (% of Rated)	GE9 MCPR Limit	GE10 MCPR Limit	ATRIUM-9B Offset MCPR Limit
110	1.11	1.11	1.11
30	1.98	1.96	2.02
0	2.54	2.52	2.57

Figure 6.3 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 operation in EOD/EOOS. The specific EOD/EOOS conditions supported for Quad Cities are identified in Table 2.4.

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 fuel type dependent cycle-specific OLMCPR penalties for QC2C16. 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. The limiting maximum pressurization conditions are explicitly evaluated and therefore, no EOD/EOOS pressure penalty is required for Cycle 16.

The Cycle 16 OLMCPR penalties for operation with FFTR, FHOOS, coastdown, or any combination thereof are 0.02, 0.10 and 0.03 for GE9, GE10 and ATRIUM-9B offset fuel, respectively. Other EOD/EOOS conditions require no OLMCPR penalty. OLMCPR penalties are determined by comparing all EOD/EOOS state points to the limiting base case state point at EOFP (100/108). Maximum pressurization analysis results confirm that the limiting EOD/EOOS MSIV closure transient (100/100 Coastdown) has approximately 16 psi margin to the vessel pressure limit and 12 psi margin to the steam dome pressure limit.

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

* The generic penalties (0.04 for OLMCPR and 5 psi for overpressurization) provided in Reference 13 are not applicable for QC2C16.

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

step after EOFP to provide for operation of 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 Tables 7.2 and 7.3. For coastdown conditions, the 100P/87%F state point is unattainable; therefore, the limiting state point for ASME overpressurization at EOFP+1500 MWd/MTU is 100%P/100%F. The limiting MSIV closure event was repeated with the SRV in service and one additional safety valve out of service (SVOOS). As seen in Table 7.3, SRVOOS bounds SVOOS.

7.3 *Combined Final Feedwater Temperature Reduction/Coastdown*

Results for combined FFTR/Coastdown are presented in Table 7.4.

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 7.5. 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 7.6.

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

**Table 7.1 Quad Cities Unit 2 Cycle 16
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	533	126	1247 / 1214	0.37 / 0.39 / 0.29	-0.05 / 0.01 /-0.06
LRNB	100 / 100	507	126	1247 / 1216	0.36 / 0.36 / 0.27	-0.06 /-0.02 /-0.08
FWCF	100 / 108	538	138	1136 / 1102	0.43 / 0.46 / 0.37	0.01 / 0.08 / 0.02
FWCF	100 / 100	512	138	1134 / 1102	0.41 / 0.44 / 0.36	-0.01 / 0.06 / 0.01

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

**Table 7.2 Quad Cities Unit 2 Cycle 16
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	689	134	1307 / 1273	0.43 / 0.47 / 0.37	0.01 / 0.09 / 0.02
LRNB	100 / 100	632	133	1306 / 1275	0.41 / 0.44 / 0.34	-0.01 / 0.06 /-0.01
FWCF	100 / 108	667	137	1193 / 1158	0.43 / 0.47 / 0.37	0.01 / 0.09 / 0.02
FWCF	100 / 100	608	136	1190 / 1158	0.41 / 0.45 / 0.35	-0.01 / 0.07 / 0.00

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

**Table 7.3 Quad Cities Unit 2 Cycle 16
Coastdown Operation ASME Overpressurization Analysis Results
and Comparison to Base Case**

Transient	Power/Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel*/ Dome Pressure (psig)	Change in Vessel*/ Dome Pressure From Base Case (psi)
MSIV	100 / 108	342	131	1358 / 1330	7 / 8
MSIV	100 / 100	337	129	1359 / 1333	8 / 8
MSIV	100 / 100†	337	129	1352 / 1326	NA

* Lower plenum.

† SVOOS. Results demonstrate that SRVOOS is more limiting than SVOOS.

**Table 7.4 Quad Cities Unit 2 Cycle 16
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	574	129	1256 / 1222	0.40 / 0.41 / 0.33	-0.02 / 0.03 /-0.02
LRNB	100 / 100	534	128	1256 / 1225	0.38 / 0.39 / 0.29	-0.04 / 0.01 /-0.06
FWCF	100 / 108	571	140	1141 / 1107	0.44 / 0.48 / 0.38	0.02 / 0.10 / 0.03
FWCF	100 / 100	528	139	1139 / 1108	0.43 / 0.46 / 0.37	0.01 / 0.08 / 0.02

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

**Table 7.5 Quad Cities Unit 2 Cycle 16
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	523	138	1133 / 1098	0.43 / 0.39 / 0.36	0.01 / 0.01 / 0.01
FWCF	100 / 100	500	137	1130 / 1099	0.41 / 0.38 / 0.35	-0.01 / 0.00 / 0.00

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

Table 7.6 Quad Cities Unit 2 Cycle 16
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	559	140	1139 / 1105	0.44 / 0.47 / 0.38	0.02 / 0.09 / 0.03
FWCF	100 / 100	524	138	1136 / 1105	0.42 / 0.45 / 0.37	0.00 / 0.07 / 0.02

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

8.0 References

1. EMF-2299 Revision 0, *Quad Cities Unit 2 Cycle 16 Reload Analysis*, Siemens Power Corporation, November 1999.

2. Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Final Quad Cities Unit 2 Cycle 16 Calculation Plan," DEG:99:127, May 7, 1999.

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-2197 Revision 0, *Quad Cities Unit 2 Cycle 16 Principal Transient Analysis Parameters*, Siemens Power Corporation, July 1999.

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 2 Cycle 16 Transient Power History Data for Confirming Mechanical Limits for GE Fuel," DEG:99:267, October 5, 1999.

8.0 References *(Continued)*

15. EMF-96-015 Revision 3, *Quad Cities Unit 2 Cycle 15 Principal Transient Analysis Parameters*, Siemens Power Corporation, August 1999.

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 - Nuclear Division, 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 2, *Impact of Failed/Bypassed LPRMs and TIPs and Extended LPRM Calibration Interval on Radial Bundle Power Uncertainty*, Siemens Power Corporation, October 1996.

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 2 Cycle 16 Safety Limit MCPR Analysis Results," DEG:99:225, August 18, 1999.

25. EMF-2176 Revision 0, *Principal Reload Fuel Design Parameters Quad Cities 2 QCB-2, Cycle 16 (ATRIUM™-9B)*, Siemens Power Corporation, February 1999.

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 Tecnnical Specifications*, as amended.

28. EMF-2222(P) Revision 0, *Dresden and Quad Cities Evaluation of Changed Analytical Neutron Flux Scram and Safety Valve Set Points*, Siemens Power Corporation, August 1999.

Appendix A

Margin to Unpiped Safety Valves

SPC performed analyses for Quad Cities Unit 2 Cycle 16 to determine the margin between peak steam line pressure and the lowest set point of the unpiped safety valves. ComEd adopts a limit of 60 psi margin for the main steam isolation valve closure - unpiped safety valve margin (MSIVC-USM) analysis. The load rejection no bypass - unpiped safety valve margin (LRNB-USM) analysis was also performed. At EOFP the limiting initial conditions for steam line pressurization occur at 100% core power and 87% core flow (100%P/87%F). For coastdown conditions of EOFP+1500 MWd/MTU, the state point 100%P/87%F is unattainable; therefore, the limiting state point for EOFP+1500 MWd/MTU is 100%P/100%F. The lowest nominal set point for a Quad Cities unpiped safety valve is 1254.7 psia.

Because the unpiped safety valve margin analyses are not licensing analyses, some of the conservatism normally assumed in COTRANSA2 analyses is relaxed. The MSIVC-USM analysis with direct scram results in a fairly mild reactor pressurization. The relief valves have sufficient capacity to depressurize the reactor once the valves actuate. The MSIVC-USM analyses with direct scram were performed with plant-specific scram insertion from Section 8.6 of Reference A.1. Technical specification relief valve (RV) opening times and delays were used with nominal RV set points for the MSIVC-USM analyses. Analyses were performed with the safety/relief valve (SRV) not credited (SRVOOS). Analyses were performed at 100%P/108%F, 100%P/100%F and 100%P/87%F for EOFP and at 100%P/108%F and 100%P/100%F for EOFP+1500 MWd/MTU to cover coastdown operation (Reference A.2). For the MSIVC-USM transient, the calculated peak steam line pressure is 1129.7 psia. This results in a calculated margin of 125.0 psi to the lowest unpiped safety valve set point as shown in Table A.1. The required 60 psi margin is met.

For the LRNB-USM analysis, nominal RV set points, stroke times and delays are used. All relief valves are assumed to be operable. A best-estimate RV opening delay time of 1.25 seconds and an opening time of 0.20 second were used in the analyses based on values from Reference A.1. Analyses are performed with and without credit for the SRV. Scram insertion is based on plant-specific data provided in Section 8.6 of Reference A.1.

The results of the LRNB-USM analyses are presented in Table A.1. Analyses were performed at 100%P/108%F, 100%P/100%F and 100%P/87%F for EOFP and at 100%P/108%F and 100%P/100%F for EOFP+1500 MWd/MTU to cover coastdown operation (Reference A.2).

Quad Cities analyses indicate that a 1% decrease in rated core power increases pressure margin approximately 4 psi (Reference A.3).

Table A.1 Margin to Opening Unpiped Safety Valve Results

Transient	Exposure	Power/Flow	Maximum SRV Pressure (psia)	Margin (psi)
LRNB-USM	EOFP	100 / 108	1224.3	30.4
LRNB-USM	EOFP+1500 MWd/MTU	100 / 108	1230.3	24.4
LRNB-USM SRVOOS	EOFP	100 / 108	1232.7	22.0
LRNB-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 108	1238.5	16.2
LRNB-USM	EOFP	100 / 100	1225.8	28.9
LRNB-USM	EOFP+1500 MWd/MTU	100 / 100	1231.4	23.3
LRNB-USM SRVOOS	EOFP	100 / 100	1234.1	20.6
LRNB-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 100	1239.6	15.1
LRNB-USM	EOFP	100 / 87	1228.4	26.3
LRNB-USM SRVOOS	EOFP	100 / 87	1236.9	17.8
MSIVC-USM SRVOOS	EOFP	100 / 108	1129.7	125.0
MSIVC-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 108	1129.7	125.0
MSIVC-USM SRVOOS	EOFP	100 / 100	1129.7	125.0
MSIVC-USM SRVOOS	EOFP+1500 MWd/MTU	100 / 100	1129.7	125.0
MSIVC-USM SRVOOS	EOFP	100 / 87	1129.7	125.0

A.1 References

A.1 EMF-2197 Revision 0, *Quad Cities Unit 2 Cycle 16 Principal Transient Analysis Parameters*, Siemens Power Corporation, July 1999.

A.2 Letter, D. E. Garber (SPC) to R. J. Chin (ComEd), "Final Quad Cities Unit 2 Cycle 16 Calculation Plan," DEG:99:127, May 7, 1999.

A.3 Letter, M. L. Hymas (SPC) to R. J. Chin (ComEd), "Dresden and Quad Cities Unpiped Safety Valve Margin Analyses," MLH:96:025, May 23, 1996.

Appendix B

Turbine Bypass Valve Out of Service

SPC performed analyses for Quad Cities Unit 2 Cycle 16 to determine specific EOOS OLMCPR penalties for operation with: (1) one bypass valve out of service (BPVOOS) and (2) all bypass valves out of service. The limiting EOFP FWCF event at 100%P/108%F was analyzed with all BPVOOS and with parameters specified in Reference B.1 for one BPVOOS.

Analysis results and a summary of corresponding OLMCPRs are provided in Table B.1. Fuel-dependent OLMCPR penalties of 0.01 (GE9), 0.02 (GE10), and 0.01 (ATRIUM-9B offset) are required for operation with one BPVOOS. For operation with all BPVOOS, the OLMCPR penalties are 0.02, 0.04, and 0.03 for GE9, GE10, and ATRIUM-9B offset fuel, respectively. These penalties must be applied to base case OLMCPRs provided in Table 2.2 to support BPVOOS operation. The OLMCPRs provided in Table B.1 are based on the BPVOOS analysis results, the plant Technical Specification two-loop SLMCPR of 1.11 and analysis of the limiting system transient analyzed in this report. The actual cycle operating limits may be higher if analyses within ComEd's scope of responsibility result in a ΔCPR higher that those in Table 2.1. For single-loop operation, the Technical Specification SLO SLMCPR of 1.12 increases the OLMCPR by 0.01.

Reduced flow MCPR limits provided in Section 6.2 may be used to determine appropriate limits for base case operation with BPVOOS.

Table B.1 Quad Cities Unit 2 Cycle 16
Turbine Bypass Valve(s) Out of Service Results

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	661	136	1200 / 1165	0.43 / 0.40 / 0.36	0.01 / 0.02 / 0.01
All BPVOOS	100 / 108	689	138	1304 / 1270	0.44 / 0.42 / 0.38	0.02 / 0.04 / 0.03

MCPR Operating Limit

Transient	OLMCPR for Base Case Operation With BPVOOS		
	GE9	GE10	ATRIUM-9B Offset
Feedwater Controller Failure (100%P / 108%F - 1 BPVOOS)	1.54	1.51	1.47
Feedwater Controller Failure (100%P / 108%F - All BPVOOS)	1.55	1.53	1.49

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

B.1 References

B.1 EMF-2197 Revision 0, *Quad Cities Unit 2 Cycle 16 Principal Transient Analysis Parameters*, Siemens Power Corporation, July 1999.

Appendix C

Power Load Unbalance Out of Service

SPC performed analyses for Quad Cities Unit 2 Cycle 16 to determine MFLCPR multipliers that protect the MCPR safety limit (SLMCPR) when the power load unbalance (PLU) is out of service. Analyses were performed using parameters specified in Reference C.1.

If the PLU is out of service due to testing when a load rejection occurs, the following sequence of events will occur. The PLU will not sense the power load unbalance and a turbine control valve fast closure will not occur. The turbine will overspeed as a result of the imbalance. This turbine overspeed will result in a higher frequency power supply and an increased speed for the recirculation pump that is provided power from the main generator. This will result in increased core flow and an associated increase in thermal power until a turbine trip occurs. A turbine trip was assumed on 62.4 Hz main generator overfrequency at 0.454 second into the event. Per Reference C.1, the turbine overspeed produced a linear increase in power supply frequency from 60 to 62.4 Hz at 0.454 second.

The recirculation pump speed is conservatively assumed to increase proportionately to the frequency increase. After the turbine trip, the pump speed linearly decreases to the initial speed in 5 seconds. The end result is a turbine trip occurring from more limiting power and flow conditions. This event is more limiting than the base case load rejection without bypass (LRNB) event described in Section 4.3.

The analyses were performed at the limiting state point of 100%P/108%F for EOFP and EOFP+1500 MWd/MTU (coastdown). Pump overspeed was modeled as a 5% linear increase of one recirculation pump from event initiation (time zero) to a time of 0.454 second. A conservative 5% increase bounds the 60–62.4 Hz frequency excursion of the turbine. After 0.454 second, the pump speed was linearly decreased from an initial normalized pump speed of 1.05 to 1.00 during the following 5 seconds. Turbine stop valve and turbine control valve closure were initiated at 0.454 second. The analyses assumed the conservative scram delay of 0.08 second associated with TCV fast closure.

Analysis results for the GE9, GE10, and ATRIUM-9B offset fuel are summarized in Table C.1. MFLCPR multipliers were determined based on the increase in ΔCPR for the PLUOOS events and the corresponding LRNB results at EOFP and EOFP+1500 MWd/MTU. OLMCPR and MFLCPR results provided in Table C.1 are based on PLUOOS analysis results, LRNB analysis results at EOFP and EOFP+1500 MWd/MTU, the plant Technical Specification two-loop SLMCPR of 1.11, and analysis of the limiting system transient analyzed in this report. Actual MFLCPR results may be lower if analyses within ComEd's scope of responsibility result in a ΔCPR higher than those provided in Table 2.1. For single-loop operation, the Technical Specification SLO SLMCPR of 1.12 increases the OLMCPR by 0.01.

The MFLCPR multipliers provided in Table C.1 may also be applied to the reduced flow MCPR limits provided in Section 6.2 to support PLUOOS operation at reduced flow conditions.

Table C.1 Quad Cities Unit 2 Cycle 16 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	755	136	1301 / 1267	0.44 / 0.41 / 0.37	0.03 / 0.04 / 0.04
PLUOOS Coastdown	100 / 108	795	139	1311 / 1277	0.47 / 0.52 / 0.39	0.04 / 0.05 / 0.02

MFLCPR Multipliers

Transient	Power/Flow	OLMCPR†	MFLCPR Multiplier†, §
PLUOOS EOFP	100 / 108	1.53 / 1.49 / 1.46	0.980 / 0.973 / 0.973
PLUOOS Coastdown	100 / 108	1.55 / 1.59 / 1.49	0.974 / 0.969 / 0.986

* Lower plenum.

† Values for GE9/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):

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

C.1 References

C.1 NDIT NFM9900224, Sequence No. 00, "Power Load Unbalance Plant Transient Input Parameters to be used in Siemens Analysis," October 27, 1999.

Distribution

D. E. Garber, 38 (15)

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Attachment D- Draft Q2C16 COLR Letter

DRAFT

January 10, 2000

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Quad Cities Nuclear Power Station
Unit 2, Cycle 16 Reload and Core Operating Limits Report
NRC Docket No. 50-265

References:

1. B. Siegel (NRC) Letter to T. Kovach (ComEd), "Approving Technical Specification Amendment and Core Operating Limits Report per Generic Letter 88-16", Dated February 8, 1990

Quad Cities Unit 2, which has completed its fifteenth cycle of operation, is currently preparing for Cycle 16 startup (estimated startup date is February 15, 2000). The purpose of this letter is to advise you of the Commonwealth Edison Company (ComEd) review and approval of the Cycle 16 reload under the provisions of 10 CFR 50.59 and to transmit the Core Operating Limits Report (COLR) for the upcoming cycle consistent with Generic Letter 88-16.

The Quad Cities Unit 2 Cycle 16 core, which consists of NRC-approved fuel types developed by Siemens Power Corporation (SPC) and General Electric (GE), was designed to operate under approved fuel design parameters, Technical Specifications, and related bases. An analysis has been performed to demonstrate that the limiting postulated FSAR events which could be affected by the reload are within allowable limits.

The reload licensing analyses performed for Cycle 16 utilized NRC-approved methodologies. The cycle-specific power distribution limits for Cycle 16 are presented in the attached COLR.

ComEd has performed a detailed review of the relevant licensing documents, the associated bases, and references. Based on that review, a safety evaluation was prepared, as required by 10 CFR 50.59, which concludes that the reload presents no unreviewed safety questions.

Further verification of the reload core design will be performed during the startup testing of Quad Cities Unit 2 Cycle 16. The startup tests will be consistent with Technical Specifications and the Draft Regulatory Guide (task SC 521-4). A summary of the key startup tests will be transmitted within 90 days following the resumption of commercial operation.

Based on the previous discussion, ComEd concludes that NRC review and approval of the Quad Cities Unit 2 reload analyses for Cycle 16 are not required for operation of the Cycle 16 core.

Please contact this office should further information be required.

Sincerely,

Joel P. Dimmette

A. Bill Beach, Regional Administrator, Region III
Quad Cities Project Manager
Senior Resident Inspector Quad Cities
Office of Nuclear Facility Safety - IDNS

COLR ATTACHMENT 3

Quad Cities Unit 2 Cycle 16

Evaluation of Fuel Thermal Conductivity

(Non-Proprietary Version for Exelon)

FRAMATOME ANP

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Mary 14, 2001
DEG:01:078

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 2 Cycle 16 Evaluation of Fuel Thermal Conductivity (Non-Proprietary Version for Exelon)

Ref.: 1. Letter, R. J. Chin to D. E. Garber (DEG:01:057) dated April 13, 2001. Subject: "Quad Cities Unit 2 Cycle 16 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/ecma

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 2 Cycle 16
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 2 Cycle 16 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 7 and 8 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 0.03 (GE9), 0.09 (GE10), and 0.04 (ATRIUM™-9B*). These penalties are required to support operation with FFTR, FHOOS, Coastdown or any combination thereof when core exposure is greater than the licensing basis core exposure at EOC16 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.

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 9 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.

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* ATRIUM is a trademark of Framatome ANP.

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 10 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 11 presents transient results and MFLCPR multipliers for PLUOOS.

Since the 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 and EOD/EOOS conditions. Results for the BPVOOS OLMCPRs can be determined from the limits shown here. Tables 4–6 and Figures 1 and 2 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-2302 Revision 0, *Quad Cities Unit 2 Cycle 16 Plant Transient Analysis*, Siemens Power Corporation, November 1999.

A.3 EMF-2299 Revision 0, *Quad Cities Unit 2 Cycle 16 Reload Analysis*, Siemens Power Corporation, November 1999.

A.4 Letter, D. E. Garber (SPC) to R. J. Chin (Exelon), "Quad Cities Unit 2 Cycle 16 Transient Power History Data for Confirming Mechanical Limits for GE Fuel – Revision 1," DEG:99:334, December 2, 1999.

Table 1 Quad Cities Unit 2 Cycle 16 Base Case ΔCPRs at Rated Power With TSSS Insertion Times

Transient	ΔCPR		
	GE9	GE10	ATRIUM-9B Offset
Load Rejection No Bypass			
100% power / 108% flow	0.41	0.37	0.34
Feedwater Flow Controller Failure			
100% power / 108% flow	0.42	0.39	0.35

Table 2 Quad Cities Unit 2 Cycle 16 MCPR Operating Limit and Maximum Pressurization Summary

MCPR Operating Limit*, †			
Operating Conditions	GE9	GE10	ATRIUM-9B Offset
Base case	1.53	1.50	1.46
Base case with 1 BPVOOS	1.54	1.51	1.47
Base case with all BPVOOS	1.56	1.53	1.50
EOD/EOOS‡	1.56	1.59	1.50

Maximum Pressurization (psig)			
Transient	Steam Dome	Lower Plenum	Steam Lines
MSIV closure without position scram (100% power / 87% flow, Base Case)	1331	1354	1330
MSIV closure without position scram (100% power / 100% flow, EOD/EOOS)	1334	1360	1334

* 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 Tables 4–6 for reduced flow MCPR limits.

† The operating limit for all fuel types is based on FWCF 100% power/108% flow. This result is shown in Table 1.

‡ Fuel-dependent cycle-specific OLMCPR penalty of 0.03 (GE9), 0.09 (GE10), and 0.04 (ATRIUM-9B offset) 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 EOC16 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.

Table 3 Quad Cities Unit 2 Cycle 16 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	664	132	1297 / 1263
Feedwater Flow Controller Failure			
100% power / 108% flow	644	135	1186 / 1151
MSIV Closure Without Position Scram (EOD/EOOS)			
100% power / 100% flow	342	130	1360 / 1334

* Lower plenum pressure.

Table 4 Flow-Dependent MCPR Results
GE9 Fuel
(Penalty Not Included)

Total Core Flow (% of rated)	1.53 OLMCPR	1.56 OLMCPR
108	1.530	1.560
100	1.617	1.649
90	1.733	1.767
80	1.859	1.896
70	1.993	2.034
60	2.135	2.179
50	2.291	2.339
40	2.476	2.527
30	2.773	2.827

Table 6 Flow-Dependent MCPR Results
ATRIUM-9B Offset Fuel

Total Core Flow (% of rated)	1.46 OLMCPR	1.50 OLMCPR
108	1.460	1.500
100	1.551	1.593
90	1.678	1.723
80	1.815	1.866
70	1.964	2.020
60	2.122	2.183
50	2.290	2.357
40	2.481	2.552
30	2.772	2.850

Table 7 Quad Cities Unit 2 Cycle 16
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	702	134	1307 / 1273	0.43 / 0.47 / 0.38	0.01 / 0.08 / 0.03

Table 8 Quad Cities Unit 2 Cycle 16
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	580	140	1142 / 1108	0.45 / 0.48 / 0.39	0.03 / 0.09 / 0.04

* Lower plenum.

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

Table 9 Quad Cities Unit 2 Cycle 16
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	672	136	1200 / 1165	0.43 / 0.40 / 0.36	0.01 / 0.01 / 0.01
All BPVOOS	100 / 108	701	138	1304 / 1271	0.45 / 0.42 / 0.39	0.03 / 0.03 / 0.04

Table 10 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	1239.8	14.9

* Lower plenum.

† Values for GE9/GE10/ATRIUM-9B offset fuel.

Table 11 Quad Cities Unit 2 Cycle 16
Power Load Unbalance Out-of-Service Results

MCPR Results and
Comparison to Corresponding
Base Case LRNB ΔCPR Results

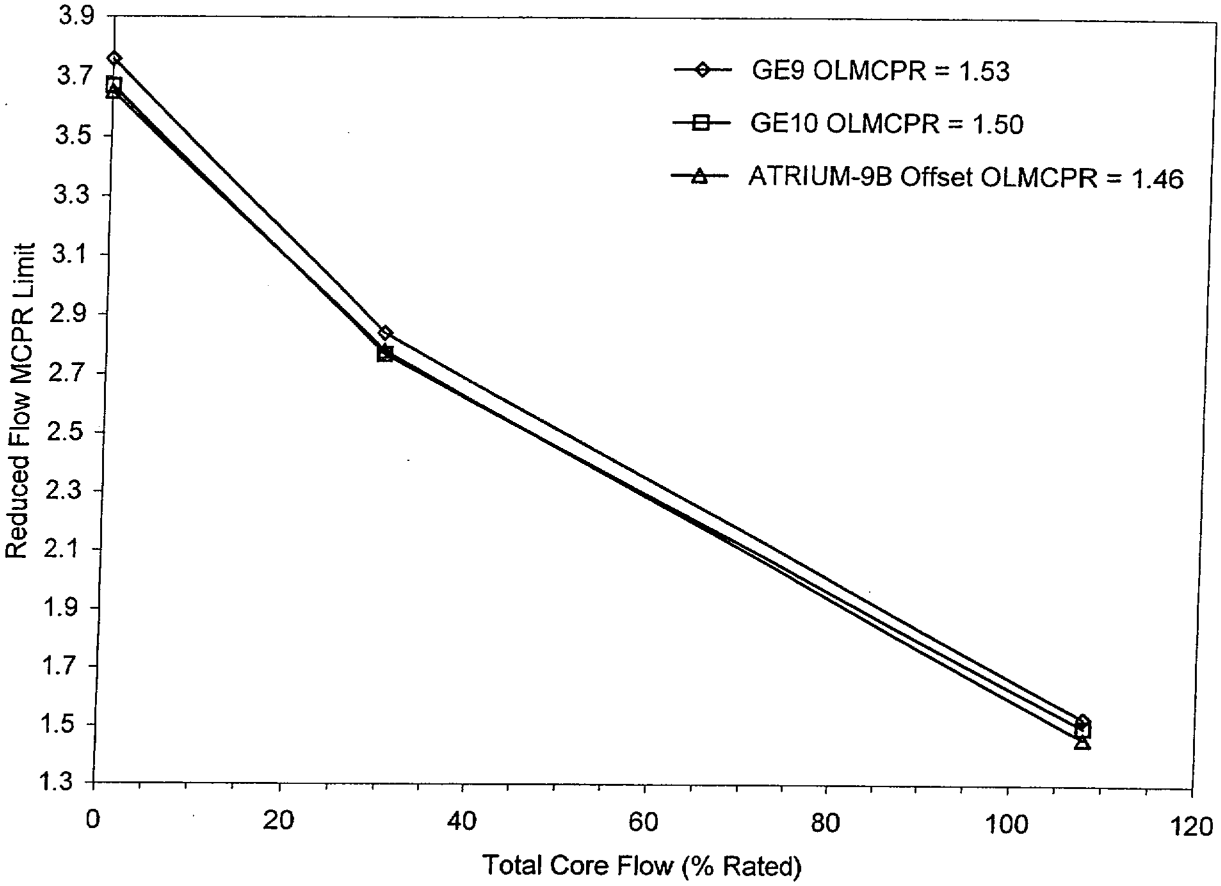
Transient	Power / Flow	Peak Neutron Flux (% rated)	Peak Heat Flux (% rated)	Maximum Vessel* / Dome Pressure (psig)	ΔCPR†	Δ(ΔCPR)†, ‡
PLUOOS EOFP	100 / 108	771	137	1302 / 1267	0.45 / 0.42 / 0.38	0.04 / 0.05 / 0.04
PLUOOS Coastdown	100 / 108	812	139	1311 / 1277	0.48 / 0.53 / 0.41	0.05 / 0.06 / 0.04

MFLCPR
Multipliers

Transient	Power / Flow	OLMCPR†	MFLCPR Multiplier†, §
PLUOOS EOFP	100 / 108	1.53 / 1.50 / 1.46	0.974 / 0.967 / 0.973
PLUOOS Coastdown	100 / 108	1.56 / 1.59 / 1.50	0.968 / 0.963 / 0.974

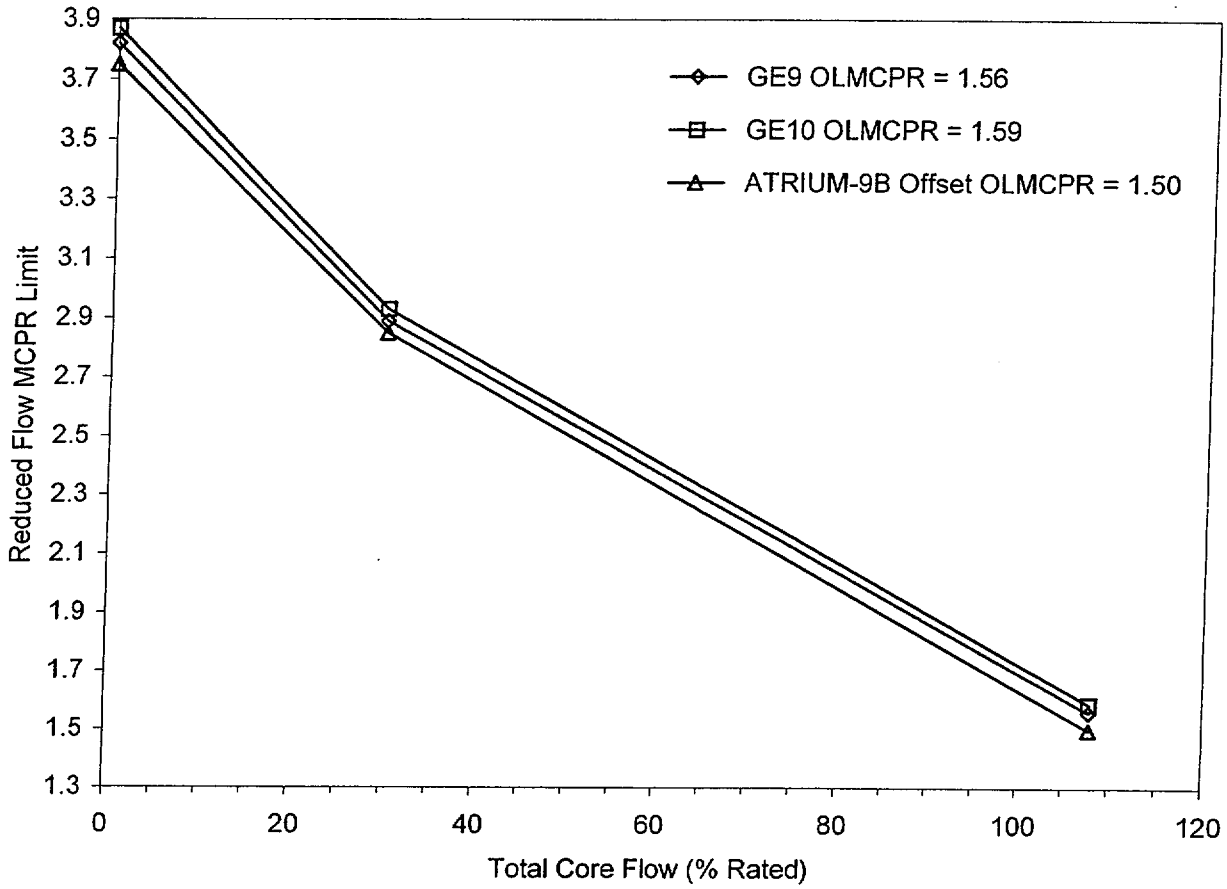
* Lower plenum.
† Values for GE10/ATRIUM-9B offset fuel.
‡ Based on PLUOOS results and corresponding base case and EOD/EOOS LRNB results.
§ The MFLCPR multipliers are calculated using the following equation (results were conservatively rounded down):

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



Total Core Flow (% of rated)	GE9 $MCPR_f$ Limit for OLMCPR = 1.53	GE10 $MCPR_f$ Limit for OLMCPR = 1.50	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR = 1.46
108	1.53	1.50	1.46
30	2.84	2.77	2.78
0	3.76	3.67	3.65

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



Total Core Flow (% of Rated)	GE9 $MCPR_f$ Limit for OLMCPR = 1.56	GE10 $MCPR_f$ Limit for OLMCPR = 1.59	ATRIUM-9B Offset $MCPR_f$ Limit for OLMCPR = 1.50
108	1.56	1.59	1.50
30	2.89	2.93	2.85
0	3.82	3.87	3.75

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