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April 12, 2001

PSLTR: #01-0043

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

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Revision to Unit 2 and Unit 3 Cycle 17 Core Operating Limits Report

Reference:

(1) Letter from J. M. Heffley (ComEd) to U.S. NRC, "Unit 2 Cycle 17 Core Operating Limits Report," dated October 22, 1999

(2) Letter from Preston Swafford (ComEd) to U.S. NRC, "Unit 3 Cycle 17 Core Operating Limits Report," dated October 22, 1999

(3) Letter from U.S. NRC to O.D. Kingsley (Exelon Generation Company), "Issuance of Amendments," dated March 30, 2001

The purpose of this letter is to transmit the revision to the Core Operating Limits Report (COLR) in accordance with Technical Specification (TS) Section 6.9.A, "Routine Reports." The analytical methods used to determine the operating limits have been previously approved by the NRC. The COLR is enclosed as an attachment to this letter.

The purpose of this revision is to incorporate references to Improved Technical Specifications (ITS) (Reference 3), add the document, title, revision and the date for the documents listed in the approved methodlogy section, revise the Minimum Crtical Power Ratio (MCPR) section of the Unit 2 Cycle 17 COLR to clarify the MPCR coastdown penalty, and revised the wording and format of the MCPR section of the Unit 3 Cycle 17 COLR to be consistent with the Unit 2 Cycle 17 COLR. There were no changes to the limits or basis for developing the limits made as a result of this revision.

A001

Should you have any questions concerning this letter, please contact Mr. D.F. Ambler at (815) 942-2920, extension 3800.

Respectfully,

Preston Swafford
Site Vice President
Dresden Nuclear Power Station

Attachment: Core Operating Limits Report, Dresden Station Unit 2 Cycle 17, dated March 2001
Core Operating Limits Report, Dresden Station Unit 3 Cycle 17, dated March 2001

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

Core Operating Limits Report

Dresden Station

Unit 2

Cycle 17

March 2001

ISSUANCE OF CHANGES SUMMARY

Affected Section	Affected Pages	Summary of Changes	Date
All	All	Original Issue Cycle 17	10/99
Figure 2.2-1	2-2	Figure 2.2-1 and the table below the figure were revised to reflect how the limits are implemented into the core monitoring code[1].	10/12/99
5.2	5-1	Section 5.2 item d was revised to clarify that during coastdown, operation is limited to the lesser of 100% CTP or the CTP calculated from items i or ii	10/12/99
Table 5.2-1	5-2	Table 5.2-1 was revised to a.) relocate the coastdown limits to a separate section of the table and b.) Specify the exposures to which the coastdown limits are to be applied[2]	10/12/99
5.2, Table 5.2-1	5-1, 5-2	Section 5.2.d and table 5.2-1 were revised to clarify the use of the coastdown penalty. Added Section 5.2.e to describe the conditions which are supportable without penalty.	3/01
1.1, 2.1, Table 2.3-1, 3.1, 4.1, 5.1, 5.2.a, Figure 5.2-1, 6.1	1-1, 2-1, 2-3, 3-1, 4-1, 5-1 5-4, 6-1	Include ITS in Technical Specification references	3/01
Methodology	6-1	Added references CTS 6.9.A.6.b.1-6.9.A.6.b.13 (ITS 5.6.5.b.1-5.6.5.b.13), including the reference revision number and date	3/01
1.0	1-2	Added Allowable Value to the title heading of table 1.2-1 to be consistent with ITS.	3/01

[1] Powerplex can not implement a step change in an operating limit. Therefore, the COLR was revised to reflect a conservative implementation method for use in the Powerplex input deck.

[2] Seq 00 of the COLR identified EOFP = 28,908 MWd/MTU core average exposure. Calculation No. BNDD:99-060 provides the BOC core average exposure = 14,478.3 MWd/MTU. Therefore, EOFP cycle exposure is equal to 28,908 MWd/MTU – 14,478 MWd/MTU = 14,429.7 MWd/MTU.

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REFERENCES

1. Commonwealth Edison Company Docket No. 50-237, Dresden Nuclear Power Station, Unit 2, Facility Operating License, License No. DPR-19.

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

4. EMF-2273, Dresden Unit 2 Cycle 17 Plant Transient Analysis, September 1999, NDIT NFM9900186 Seq00.

5. EMF-2275, Dresden Unit 2 Cycle 17 Reload Analysis, September 1999, NDIT NFM9900187 Seq00.

6. Dresden Unit 2 Cycle 17 Neutronics Licensing Report (NLR), July 23, 1999, NDIT NFM9900126 Seq00.

7. EMF-92-149 (P) and Supplement 1 Revision 1, Dresden Units 2 and 3 Generic Coastdown Analysis with ATRIUM-9B, September 1996, NFM NDIT 960137 Revision 1.

8. GE DRF C51-00217-01, Instrument Setpoint Calculation Nuclear Instrumentation Rod Block Monitor, December 1999.

LIST OF FIGURES

LIST OF TABLES

1.0 ROD BLOCK MONITOR (RBM)

1.1 Technical Specification Reference

CTS 3.3.M - Rod Block Monitor (RBM)
ITS 3.3.2.1 – Control Rod Block Instrumentation
CTS Table 3.2.E-1 – Control Rod Block Instrumentation
CTS Table 4.2.E-1 – Control Rod Block Instrumentation Surveillance Requirements
ITS Table 3.3.2.1-1 – Control Rod Block Instrumentation

1.2 Description

The Rod Block Monitor Upscale Instrumentation Setpoints are determined from the relationships shown in Table 1.2-1.

TABLE 1.2-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION:	CTS TRIP LEVEL SETTING: ITS ALLOWABLE VALUE
Rod Block Monitor Upscale (Flow Bias)	
Dual Loop Operation	$\leq$ 0.65 W_d plus 55*
Single Loop Operation	$\leq$ 0.65 W_d plus 51*

*W_d - percent of drive flow required to produce a rated core flow of 98 Mlb/hr.

2.0 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

2.1 Technical Specification References

CTS 3.11.A - AVERAGE PLANAR LINEAR HEAT GENERATION RATE
ITS 3.2.1 – AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

2.2 Description

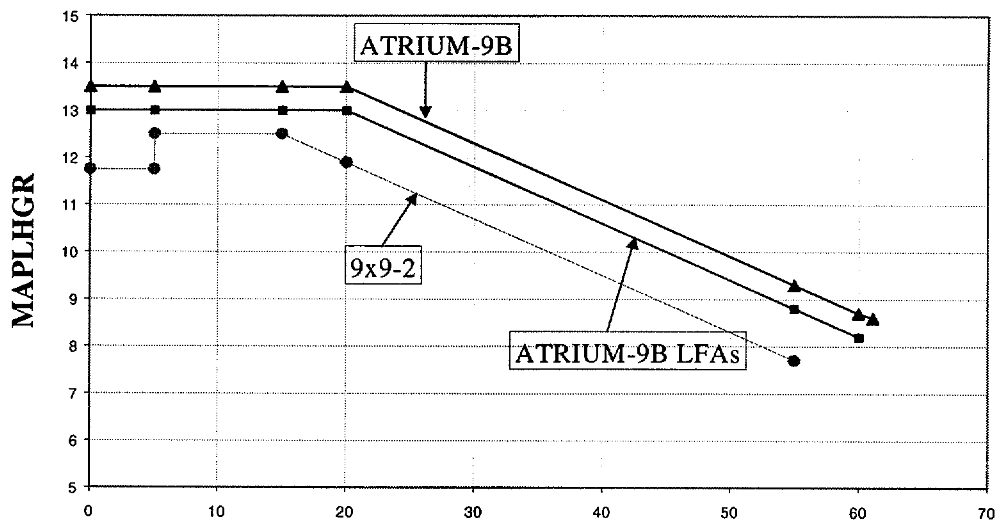
The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 2.2-1.

2.3 MAPLHGR Multipliers

The appropriate multiplicative factor, during single loop operation, to apply to the base MAPLHGR limits specified in Section 2.2 is shown in Table 2.3-1.

FIGURE 2.2-1

MAPLHGR LIMIT VS PLANAR AVERAGE EXPOSURE



Planar Average Exposure (GWD/MTU)

Planar Average Exposure (GWd/MTU)	MAPLHGR Limit 9x9-2 (kW/ft)
0	11.75
5	11.75
5.01	12.5
15	12.5
20	11.9
55	7.7

Planar Average Exposure (GWd/MTU)	MAPLHGR Limit ATRIUM-9B (offset and non-offset) (kW/ft)	MAPLHGR Limit ATRIUM-9B LFA (kW/ft)
0	13.5	13.0
5	13.5	13.0
5	13.5	13.0
15	13.5	13.0
20	13.5	13.0
55	9.3	8.8
60	8.7	8.2
61.1	8.6	-

TABLE 2.3-1

SINGLE LOOP OPERATION MAPLHGR LIMIT MULTIPLIERS

Technical Specification	Title of Technical Specification	Multiplicative Factor 9x9-2	Multiplicative Factor ATRIUM-9B (offset, non-offset, and LFA)
CTS 3.11.A	Average Planar LHGR	0.90	0.90
ITS 3.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	0.90	0.90
CTS 3.6.A action 1d	Recirculation Loops	0.90	0.90
ITS 3.4.1	Recirculation Loops Operating	0.90	0.90

3.0 STEADY STATE LHGR

3.1 Technical Specification Reference

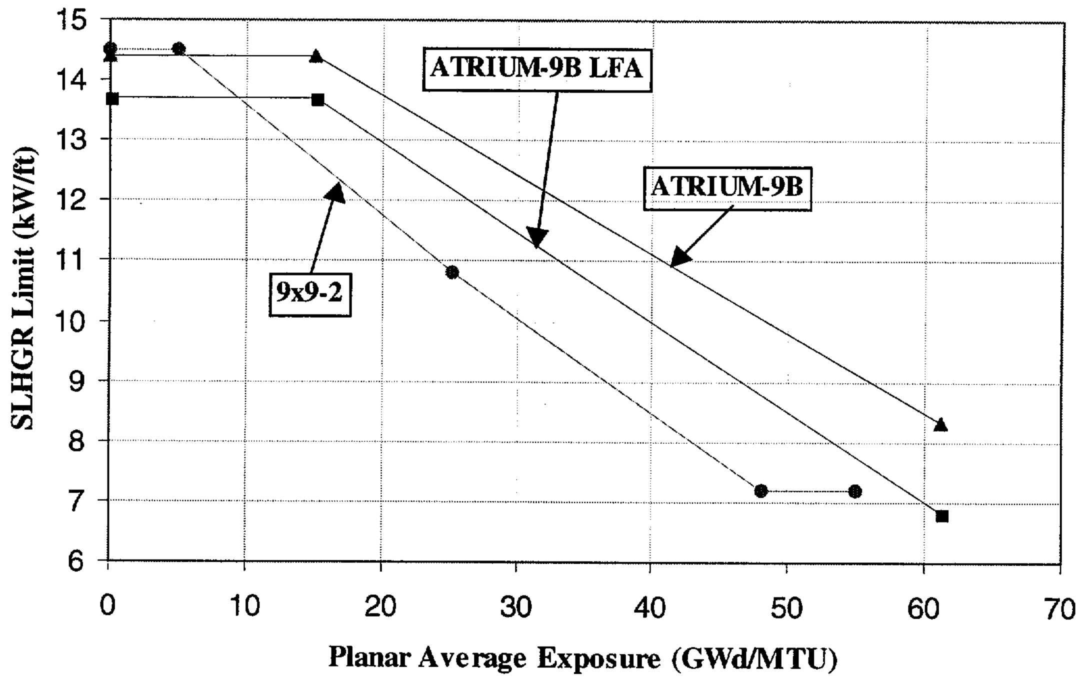
CTS 3.11.D - STEADY STATE LINEAR HEAT GENERATION RATE
ITS 3.2.3 – LINEAR HEAT GENERATION RATE (LHGR)

3.2 Description

The Steady State LHGR (SLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 3.2-1.

FIGURE 3.2-1

STEADY STATE LHGR (SLHGR) LIMIT VS. PLANAR AVERAGE EXPOSURE



Planar Average Exposure (GWd/MTU)	SLHGR Limit 9x9-2 (kW/ft)
0	14.5
5.0	14.5
25.2	10.8
48.0	7.2
55	7.2

Planar Average Exposure (GWd/MTU)	SLHGR Limit ATRIUM-9B LFA (kW/ft)	SLHGR Limit ATRIUM-9B (offset and non-offset) (kW/ft)
0	13.7	14.4
15.0	13.7	14.4
61.1	6.84	8.32

4.0 TRANSIENT LHGR

4.1 Technical Specification Reference

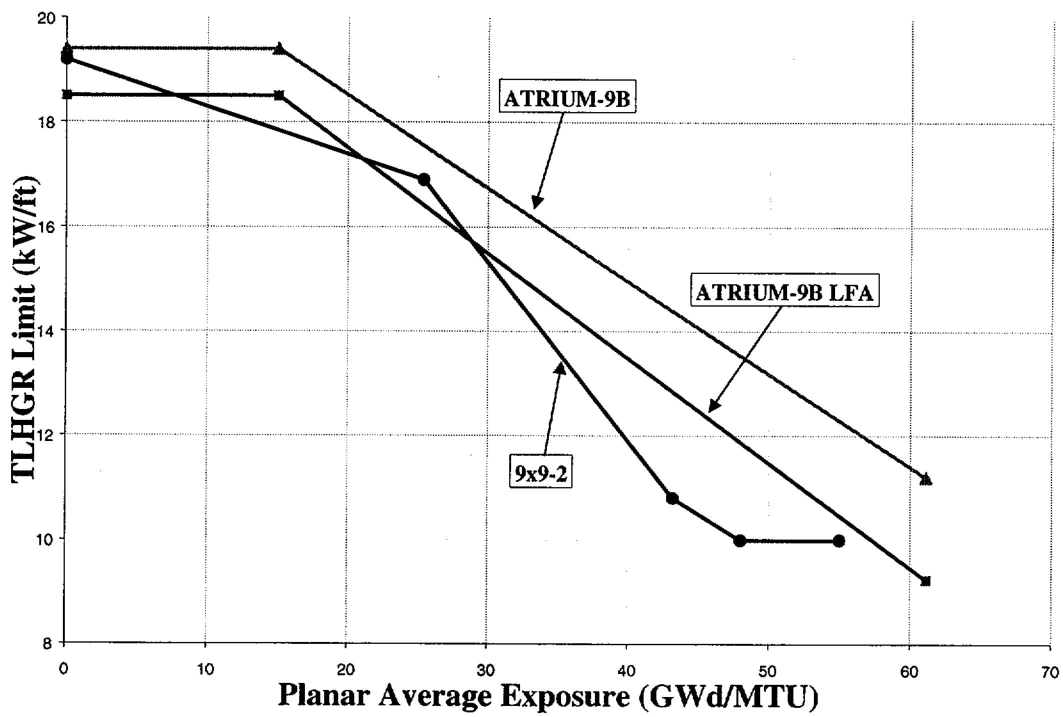
CTS 3.11.B - TRANSIENT LINEAR HEAT GENERATION RATE
ITS 3.2.4 – AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINT

4.2 Description

The Transient LHGR (TLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 4.2-1.

FIGURE 4.2-1

TRANSIENT LHGR (TLHGR) LIMIT VS. PLANAR AVERAGE EXPOSURE



Planar Average Exposure (GWd/MTU)	TLHGR Limit 9x9-2 (kW/ft)
0	19.2
25.4	16.9
43.2	10.8
48.0	10.0
55	10.0

Planar Average Exposure (GWd/MTU)	TLHGR Limit ATRIUM-9B LFA (kW/ft)	TLHGR Limit ATRIUM-9B (offset and non-offset) (kWft)
0	18.5	19.4
15.0	18.5	19.4
61.1	9.24	11.2

5.0 MINIMUM CRITICAL POWER RATIO

5.1 Technical Specification References

CTS 3.11.C - MINIMUM CRITICAL POWER RATIO

ITS 3.2.2 – MINIMUM CRITICAL POWER RATIO (MCPR)

ITS 3.7.7 – THE MAIN TURBINE BYPASS SYSTEM

5.2 Description

a. The Operating Limit MCPRs for D2C17 are listed in Table 5.2-1 for 9x9-2 and ATRIUM-9B (including LFAs). The OLMCPRs calculated for D2C17 are based on Technical Specification CRD Scram Insertion Speeds (CTS 3.3.E, ITS 3.1.4). When necessary the Operating Limit MCPR from Table 5.2-1 is supplemented by Table 5.2-2 as appropriate.

b. During Manual Flow Control, the Operating Limit MCPR for each fuel type at reduced core flow conditions can be determined from (i) or (ii), whichever is greater:

   i. Figure 5.2-1 using the appropriate flow rate, or

   ii. The Operating Limit MCPR determined from Table 5.2-1 as appropriate and supplemented by Table 5.2-2 as appropriate.

c. Automatic Flow Control is not supported for D2C17

d. During operation at cycle exposure > 14,429.7 MWd/MTU, power is limited to the lesser of 100% CTP or the following[1]

   i. Apply the appropriate limits for no CTP overshoot as described in Section 5.2.b and monitor and maintain core thermal power as follows:

$$\text{CTP}(\%\text{rated}) \le 100 - 10 * \left(\frac{\text{current\_exposure(MWd/MTU)} - \text{EOFP(MWd/MTU)}}{1000}\right)$$

   ii. Or apply the appropriate limits for 15% CTP overshoot as described in section 5.2.b and monitor and maintain core thermal power as follows:

$$\text{CTP }(\%\text{rated}) \le 100 - 10 * \left(\frac{\text{current\_exposure(MWd/MTU)} - (\text{EOFP} + 1500)\text{(MWd/MTU)}}{1000}\right)$$

e. The following conditions are supported without penalty:

- 40% TIP channels unavailable
- 50% LPRMs unavailable
- 2000 EFPH LPRM calibration interval

[1]EOFP is equal to a D2C17 cycle exposure of 14,429.7 MWd/MTU

TABLE 5.2-1

OPERATING LIMIT MCPR

OLMCPR for Operation ≤ **13,800** MWd/MTU Cycle Exposure		
Operating Scenario	**9x9-2 Fuel Operating Limit MCPR**	**ATRIUM-9B[1] Operating Limit MCPR**
Two Loop Operation[2]	1.48	1.45
Single Loop Operation[2]	1.49	1.46

13,800 MWd/MTU < OLMCPR for Cycle Exposure ≤ **14,429.7** MWd/MTU **OR** OLMCPR for Cycle Exposure > **14,429.7** MWd/MTU with no CTP Overshoot		
Operating Scenario	**9x9-2 Fuel Operating Limit MCPR**	**ATRIUM-9B[1] Operating Limit MCPR**
Two Loop Operation and CTP maintained per 5.2.d.i[3]	1.51	1.48
Single Loop Operation and CTP maintained per 5.2.d.i[3]	1.52	1.49

OLMCPR for Cycle Exposure > **14,429.7** MWd/MTU with 15% CTP Overshoot		
Operating Scenario	**9x9-2 Fuel Operating Limit MCPR**	**ATRIUM-9B[1] Operating Limit MCPR**
Two Loop Operation and CTP maintained per 5.2.d.ii[2]	1.59	1.52
Single Loop Operation and CTP maintained per 5.2.d.ii[2]	1.60	1.53

[1] Results presented are for both the offset, non-offset, and LFA designs.

[2] Includes operation with Feedwater Heaters Out of Service (FHOOS) for up to 100 °F reduction in feedwater temperature.

[3] Includes operation with FHOOS for up to 100 °F reduction in feedwater temperature. For cycle exposure > 14,429.7 MWd/MTU the 15% CTP Overshoot limits must be applied if feedwater temperature reduction causes CTP to exceed the limits of 5.2.d.i.

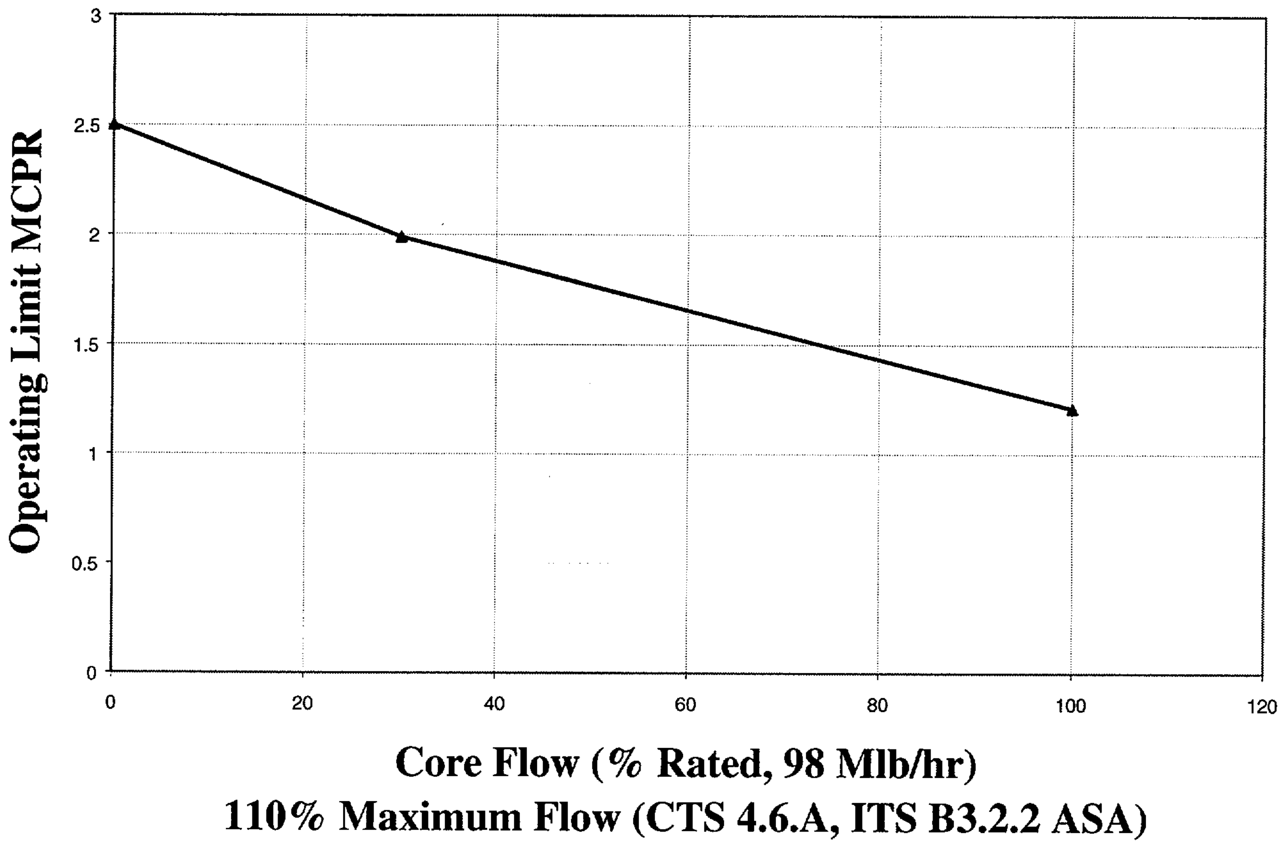
TABLE 5.2-2

BYPASS VALVE DEGRADATION OLMCPR ADDERS

Bypass Valve Delay Time (msec)	9x9-2 OLMCPR Adder (ΔCPR)	ATRIUM-9B (offset, non-offset, and LFA) OLMCPR Adder (ΔCPR)
$0 \leq t \leq 50$	0	0
$50 < t < 75$	0	0
$75 \leq t < 135$	0.01	0.01
$135 \leq t < 1078$	0.02	0.02
$1078 \leq t < 1150$	0.03	0.03
$1150 \leq t < 1288$	0.03	0.04
Bypass valves inoperable or $(t \geq 1288)$	0.03	0.05

FIGURE 5.2-1

OPERATING LIMIT MCPR FOR MANUAL FLOW CONTROL



Total Core Flow (% Rated)	Operating Limit MCPR for ATRIUM-9B (offset and non-offset) and 9x9-2 Fuel
100	1.21
30	1.99
0	2.50

6.0 METHODOLOGY

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. These methodologies are listed in CTS 6.9.A.6.b, and per ITS 5.6.5.b, the complete identification for each of the TS referenced topical reports used to prepare the COLR are listed below.

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

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

3) XN-NF-79-71(P)(A) Revision 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors." Exxon Nuclear Company Inc, November 1981.

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.

5) XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel." Exxon Nuclear Company, September 1986.

6) ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 1, 2, 3, and 4, "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis." Advanced Nuclear Fuels Corporation, August 1990.

7) XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualifications of ENC 9x9 BWR Fuel, Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

8) ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

9) ANF-89-98(P)(A) Generic Mechanical Design Criteria for BWR Fuel Designs, Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May, 1995.

10) ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, Advanced Nuclear Fuels Corporation, January 1993.

11) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods." Revision 0 and Supplements on Neutronics Licensing Analyses (Supplement 1) and La Salle County Unit 2 Benchmarking (Supplement 2), December 1991, March 1992, and May 1992, respectively.

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

13) EMF-85-74 (P), Revision 0, and Supplement 1 (P)(A) and Supplement 2 (P)(A) RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

Core Operating Limits Report

Dresden Station

Unit 3

Cycle 17

March 2001

ISSUANCE OF CHANGES SUMMARY

Affected Section	Affected Pages	Summary of Changes	Date
All	All	Original Issue Cycle 17	10/00
5.2	5-1, 5-2, 5-3	Section 5.2 was revised to be consistent with the format used in the D2 C17 COLR.	3/01
6.0	6-1, 6-2	Added references CTS 6.9.A.6.b.1-6.9.A.6.b.13 (ITS 5.6.5.b.1-5.6.5.b.13), including the reference revision number and date	3/01
1.0	1-2	Added Allowable Value to the title heading of table 1.2-1 to be consistent with ITS.	3/01

TABLE OF CONTENTS

REFERENCES

1. Commonwealth Edison Company Docket No. 50-249, Dresden Nuclear Power Station, Unit 3, Facility Operating License DPR-25.

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

3. Dresden LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-9B and 9x9-2 Fuel, EMF-98-007(P), January 1998, NFS NDIT No. 9800072 Seq. 00.

4. DG00-001047, Dresden Unit 3 Cycle 17 Plant Transient Analysis, Siemens Document EMF-2406, Revision 0, July 2000.

5. DG00-001046, Dresden Unit 3 Cycle 17 Reload Analysis, Siemens Document EMF-2421, Revision 0, July 2000.

6. DG00-000907, Dresden Unit 3 Cycle 17 Neutronics Licensing Report (NLR), July 28, 2000, TODI No. NFM0000086 Seq. 00.

7. Dresden Units 2 and 3 Generic Coastdown Analysis with ATRIUM-9B, EMF-92-149 (P) and EMF-92-149(P) Supplement 1, Revision 1, September 1996, NFS NDIT No. 960137 Seq. 00.

8. Letter, David Garber (SPC) to Dr. R. J. Chin, "Dresden Operation with Final Feedwater Temperature Reduction," DEG:00:176, July 24, 2000.

9. GE DRF C51-00217-01, Instrument Setpoint Calculation Nuclear Instrumentation Rod Block Monitor, December 1999.

LIST OF FIGURES

LIST OF TABLES

1.0 ROD BLOCK MONITOR (RBM)

1.1 Technical Specification Reference

CTS 3.3.M - Rod Block Monitor (RBM)

ITS 3.3.2.1 - Control Rod Block Instrumentation

CTS Table 3.2.E-1 – Control Rod Block Instrumentation

CTS Table 4.2.E-1 – Control Rod Block Instrumentation Surveillance Requirements

ITS Table 3.3.2.1-1 – Control Rod Block Instrumentation

1.2 Description

The Rod Block Monitor Upscale Instrumentation Setpoints are determined from the relationships shown in Table 1.2-1.

TABLE 1.2-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION:	CTS TRIP LEVEL SETTING: ITS ALLOWABLE VALUE
Rod Block Monitor Upscale (Flow Bias)	
Dual Loop Operation	Less than or equal to (0.65 W_d plus 55)*
Single Loop Operation	Less than or equal to (0.65 W_d plus 51)*

*W_d - percent of drive flow required to produce a rated core flow of 98 Mlb/hr.

2.0 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

2.1 Technical Specification References

CTS 3.11.A - AVERAGE PLANAR LINEAR HEAT GENERATION RATE

ITS 3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

2.2 Description

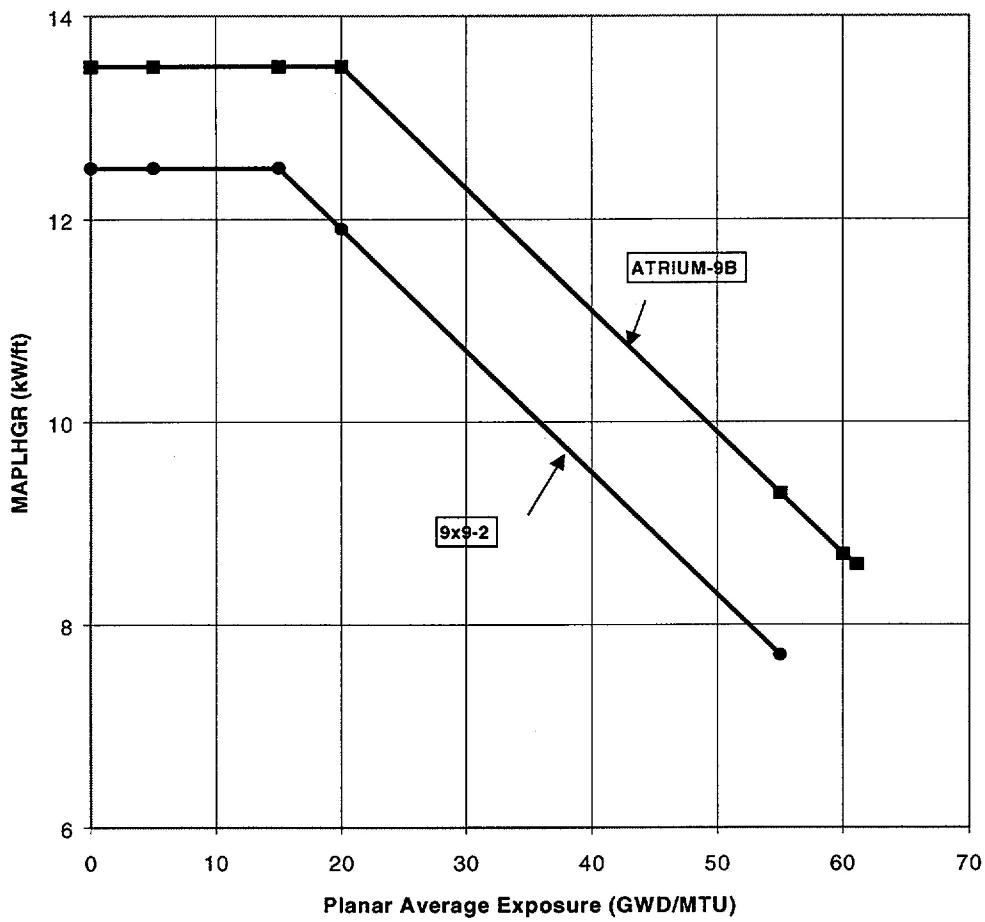
The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 2.2-1.

2.3 MAPLHGR Multipliers

The appropriate multiplicative factor, during power operation with equipment out of service, to apply to the base MAPLHGR limits specified in Section 2.2 is shown in Table 2.3-1.

FIGURE 2.2-1

MAPLHGR LIMIT VS PLANAR AVERAGE EXPOSURE



Planar Average Exposure (GWD/MTU)	MAPLHGR Limit (kW/ft) 9x9-2	MAPLHGR Limit (kW/ft) ATRIUM-9B (offset & non-offset)
0	12.5	13.5
5	12.5	13.5
15	12.5	13.5
20	11.9	13.5
55	7.7	9.3
60		8.7
61.1		8.6

TABLE 2.3-1

EQUIPMENT OUT OF SERVICE MAPLHGR LIMIT MULTIPLIERS

Technical Specifications		Scenario	Multiplicative Factor, 9x9-2	Multiplicative Factor, ATRIUM-9B (offset & non-offset)
CTS 3.11.A,	Average Planar LHGR	Single Loop Operation (SLO)	0.90	0.90
ITS 3.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)			
CTS 3.6.A Action 1.d	Recirculation Loops			
ITS 3.4.1	Recirculation Loops Operating			

3.0 STEADY STATE LHGR

3.1 Technical Specification Reference

CTS 3.11.D - STEADY STATE LINEAR HEAT GENERATION RATE

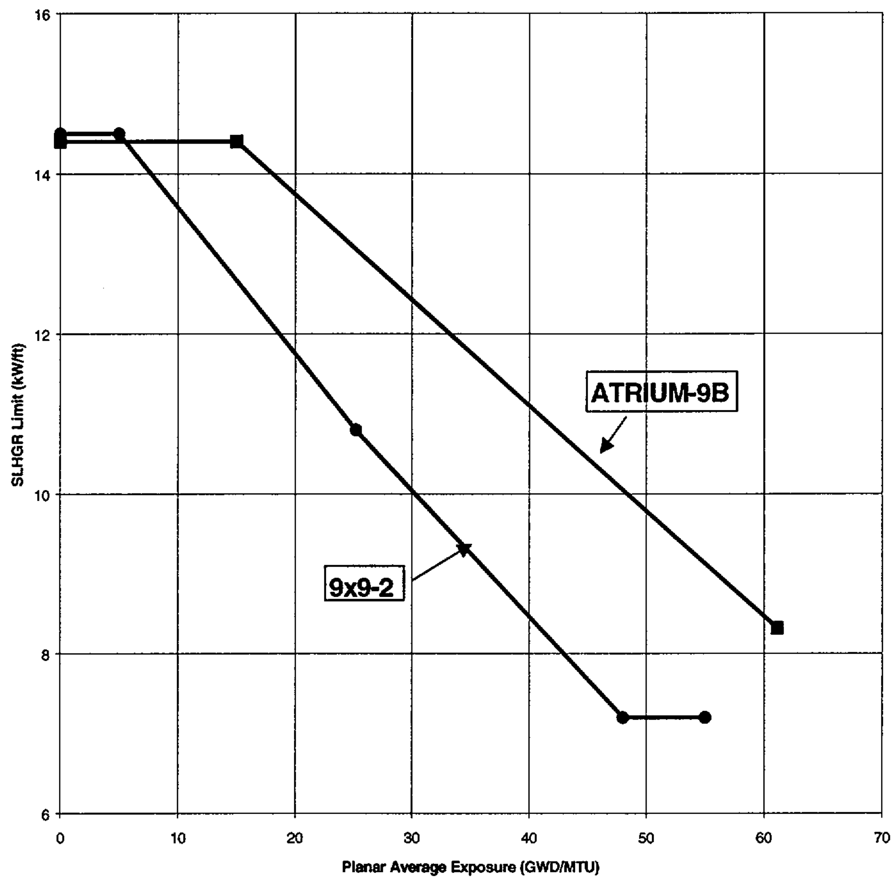
ITS 3.2.3 - LINEAR HEAT GENERATION RATE (LHGR)

3.2 Description

The Steady State LHGR (SLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 3.2-1.

FIGURE 3.2-1

STEADY STATE LHGR (SLHGR) LIMIT VS. PLANAR AVERAGE EXPOSURE



Planar Average Exposure (GWD/MTU)	SLHGR Limit (kW/ft) ATRIUM-9B (offset & non-offset)
0.0	14.4
15.0	14.4
61.1	8.32

Planar Average Exposure (GWD/MTU)	SLHGR Limit (kW/ft) 9x9-2
0.0	14.5
5.0	14.5
25.2	10.8
48.0	7.2
55.0	7.2

4.0 TRANSIENT LHGR

4.1 Technical Specification Reference

CTS 3.11.B - TRANSIENT LINEAR HEAT GENERATION RATE

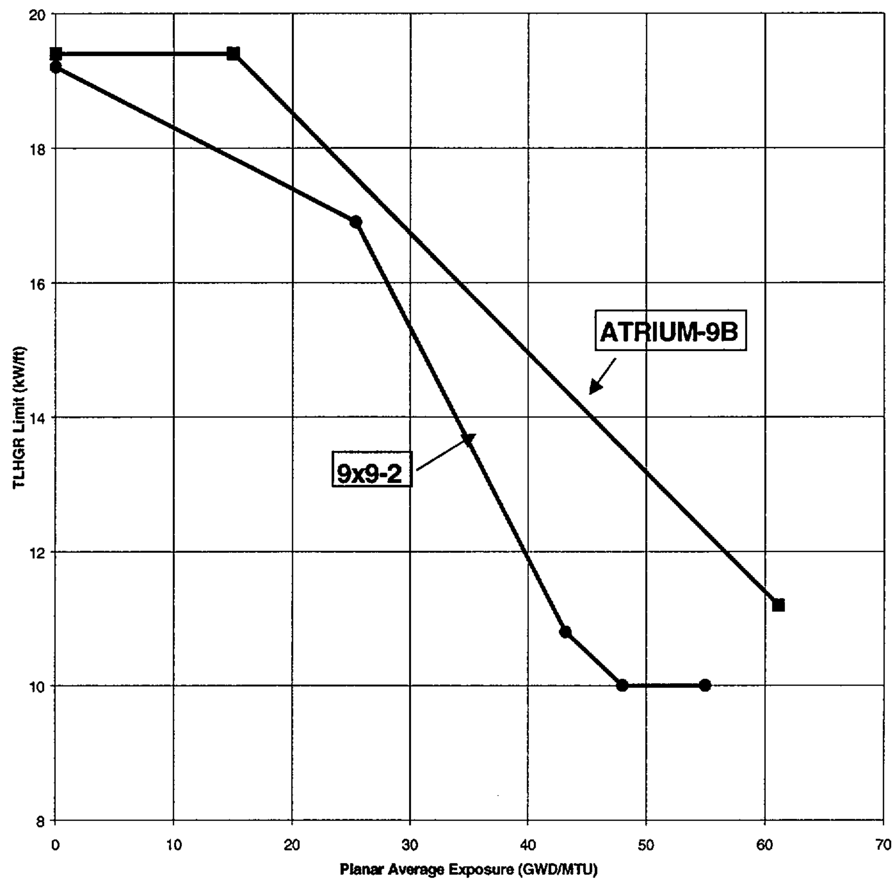
ITS 3.2.4 - Average Power Range Monitor (APRM) Gain and Setpoint

4.2 Description

The Transient LHGR (TLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 4.2-1.

FIGURE 4.2-1

TRANSIENT LHGR (TLHGR) LIMIT VS. PLANAR AVERAGE EXPOSURE



Planar Average Exposure (GWD/MTU)	TLHGR Limit (kW/ft) ATRIUM-9B (offset & non-offset)
0.0	19.4
15.0	19.4
61.1	11.2

Planar Average Exposure (GWD/MTU)	TLHGR Limit (kW/ft) 9x9-2
0.0	19.2
25.4	16.9
43.2	10.8
48.0	10
55.0	10

5.0 MINIMUM CRITICAL POWER RATIO

5.1 Technical Specification References

CTS 3.11.C - MINIMUM CRITICAL POWER RATIO

ITS 3.2.2 - MINIMUM CRITICAL POWER RATIO (MCPR)

ITS 3.7.7 - The Main Turbine Bypass System

5.2 Description

a. The Operating Limit MCPRs for D3C17 are listed in Table 5.2-1 for 9x9-2 and ATRIUM-9B. The OLMCPRs calculated for D3C17 are based on Technical Specification CRD Scram Insertion Speeds (CTS 3.3.E, ITS 3.1.4). When necessary for slower than normal bypass valve opening times or for operation with inoperable turbine bypass valves[1], apply the appropriate Operating Limit MCPR adder provided in Figure 5.2-2.

b. During Manual Flow Control, the Operating Limit MCPR for each fuel type at reduced core flow conditions can be determined from (i) or (ii), whichever is greater:

   i. Figure 5.2-1 using the appropriate flow rate, or

   ii. The Operating Limit MCPR determined from Table 5.2-1 as appropriate and supplemented by Figure 5.2-2 as appropriate.

c. Automatic Flow Control is not supported for D3C17.

d. Core Flow must be maintained $\leq$ 108% of rated.[1]

e. During operation at core average exposure > 30,837 MWd/MTU, power is limited to the lesser of 100% CTP or the following:

   i. Apply the appropriate limits for no CTP overshoot as described in section 5.2.b and monitor and maintain CTP as follows:

$$\text{CTP (\%rated)} \leq 100 - 10 * \left(\frac{\text{current core average exposure (MWD/MTU)} - 30{,}837\ \text{(MWD/MTU)}}{1000} \right)$$

   ii. Apply the appropriate limits for 15% CTP overshoot as described in section 5.2.b and monitor and maintain CTP as follows:

$$\text{CTP (\%rated)} \leq 100 - 10 * \left(\frac{\text{current core average exposure (MWD/MTU)} - 32{,}337\ \text{(MWD/MTU)}}{1000} \right)$$

f. The following conditions are supported without penalty:

- 40% TIP channels unavailable
- 50% LPRMs unavailable
- 4 Safety Valves OOS[1]
- 1 Relief Valve OOS[1]
- 2500 EFPH LPRM Calibration interval (2000 EFPH + 25% Grace)

[1] Ensure the unit's licensing basis permits operation in this condition prior to crediting this flexibility.

TABLE 5.2-1
OPERATING LIMIT MCPR

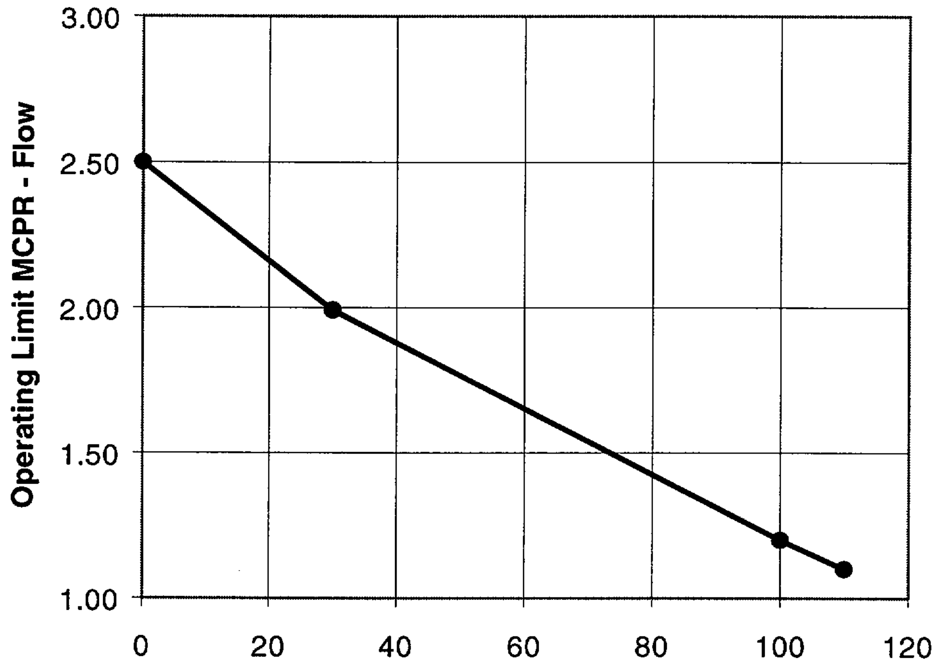
OLMCPR for Core Average Exposure ≤ **30,837** MWd/MTU **OR** OLMCPR for Core Average Exposure > **30,837** MWd/MTU with no CTP Overshoot		
Operating Scenario	**9x9-2 Fuel Operating Limit MCPR**	**ATRIUM-9B[1] Operating Limit MCPR**
Two Loop Operation and CTP maintained per 5.2.e.i[2]	1.43	1.43
Single Loop Operation and CTP maintained per 5.2.e.i[2]	1.44	1.44

OLMCPR for Core Average Exposure > **30,837** MWd/MTU with 15% CTP Overshoot		
Operating Scenario	**9x9-2 Fuel Operating Limit MCPR**	**ATRIUM-9B Operating Limit MCPR**
Two Loop Operation and CTP maintained per 5.2.e.ii[3]	1.46	1.46
Single Loop Operation and CTP maintained per 5.2.e.ii[3]	1.47	1.47

[1] Includes both offset and non-offset ATRIUM-9B fuel designs.
[2] Includes up to 100°F reduced feedwater temperature from normal. For core average exposure >30,837 MWd/MTU, the 15% CTP Overshoot limits must be applied if feedwater temperature reduction causes CTP to exceed the limits of 5.2.e.i.
[3] Includes up to 100°F reduced feedwater temperature from normal.

FIGURE 5.2-1

OPERATING LIMIT MCPR FOR MANUAL FLOW CONTROL

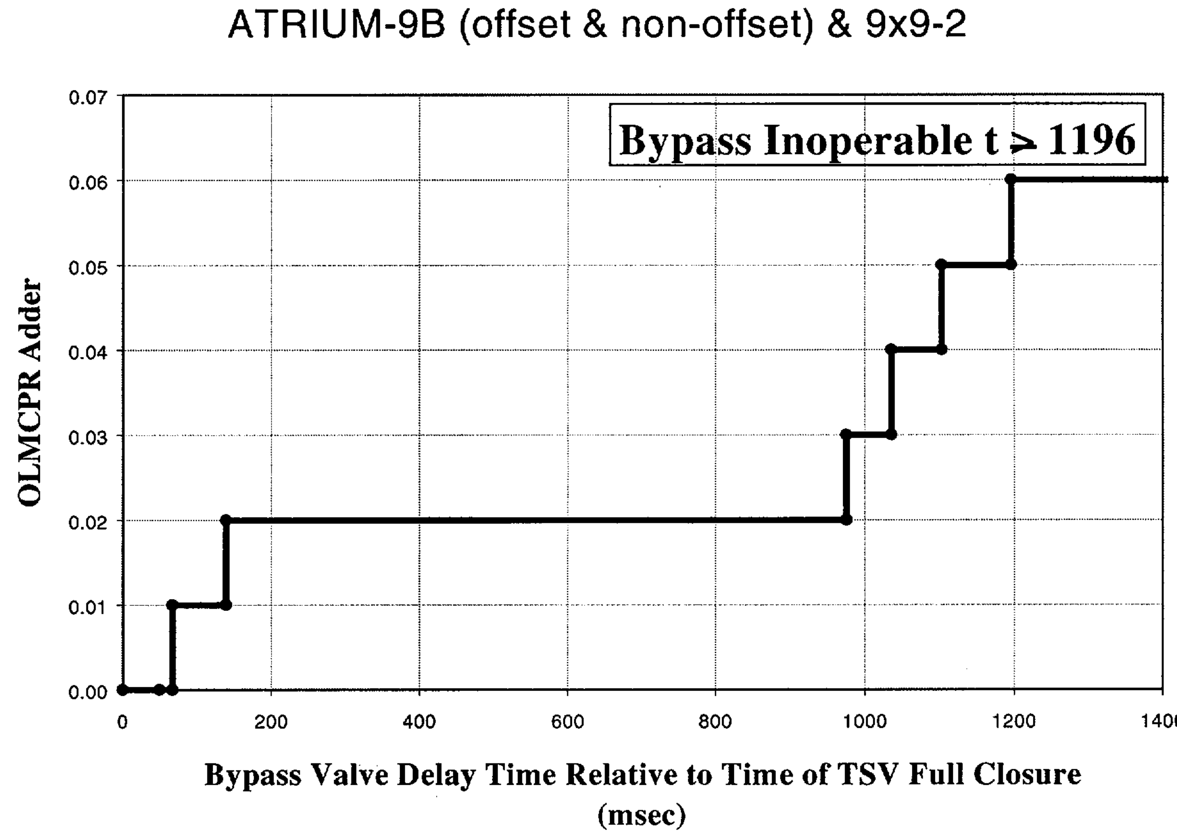


Reactor Core Flow (% Rated, 98 Mlb/hr)
110% Maximum Flow (CTS 4.6.A, ITS B3.2.2 ASA)

Reactor Core Flow (% Rated)	Operating Limit MCPR - Flow ATRIUM-9B (offset & non-offset) and 9x9-2
110	1.10
100	1.20
30	1.99
0	2.50

FIGURE 5.2-2

MAIN TURBINE BYPASS VALVE OLMCPR ADDERS



Bypass Valve (BPV) Delay Time (msec)	OLMCPR Adder[1,2] (ΔCPR) ATRIUM-9B (offset and non-offset) & 9x9-2
$0 \leq t \leq 67$	0
$67 < t \leq 139$	0.01
$139 < t \leq 975$	0.02
$975 < t \leq 1036$	0.03
$1036 < t \leq 1103$	0.04
$1103 < t \leq 1196$	0.05
$t > 1196$	0.06
Two or more BPV inoperable[2]	0.06

[1] Includes the effects of one BPV inoperable with no OLMCPR adjustment required.
[2] Ensure the unit's licensing basis permits operation with one or more BPV inoperable prior to utilizing the associated penalty.

6.0 METHODOLOGY

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. These methodologies are listed in CTS 6.9.A.6.b, and per ITS 5.6.5.b, the complete identification for each of the ITS referenced topical reports used to prepare the COLR are listed below.

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

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.

3) XN-NF-79-71(P)(A) Revision 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors." Exxon Nuclear Company, November 1981.

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.

5) XN-NF-85-67(P)(A) Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel." Exxon Nuclear Company, September 1986.

6) ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 1, 2, 3, and 4, "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis." Advanced Nuclear Fuels Corporation, August 1990.

7) XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualifications of ENC 9x9 BWR Fuel, Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

8) ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

9) ANF-89-98(P)(A) Generic Mechanical Design Criteria for BWR Fuel Designs, Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May, 1995.

10) ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, Advanced Nuclear Fuels Corporation, January 1993.

11) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods." Revision 0 and Supplements on Neutronics Licensing Analyses (Supplement 1) and La Salle County Unit 2 Benchmarking (Supplement 2), December 1991, March 1992, and May 1992, respectively.

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

13) EMF-85-74 (P) Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model, Siemens Power Corporation, February 1998.