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U.S. Nuclear Regulatory Commission
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Dresden Nuclear Power Station, Unit 3
Facility Operating License No. DPR-25
NRC Docket No. 50-249

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

The purpose of this letter is to transmit the revision to the Core Operating Limits Report (COLR) in accordance with Technical Specification (TS) Section 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)." 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.

This revision includes allowance for a final feedwater temperature reduction of 120 degrees F, instead of the current 100 degrees F allowed reduction. On June 6, 2002, Framatome ANP Richland Inc., submitted analyses to Exelon documenting the acceptability of reducing final feedwater temperature by 120 degrees F prior to the implementation of Extended Power Uprate (EPU). There were no changes to the basis for developing the limits as a result of this revision.

Should you have any questions concerning this letter, please contact Mr. J. Hansen at (815) 416-2800.

Respectfully,

RJH

R.J. Hovey
Site Vice President
Dresden Nuclear Power Station

Attachment: Core Operating Limits Report, Dresden Station Unit 3 Cycle 17, dated August 2002

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

A001

Core Operating Limits Report

Dresden Station

Unit 3

Cycle 17

August 2002

ISSUANCE OF CHANGES SUMMARY

Affected Section	Affected Pages	Summary of Changes	Date
All	All	Original Issue Cycle 17	10/00
5.2 6.0 1.0	5-1, 5-2, 5-3 6-1, 6-2 1-2	Section 5.2 was revised to be consistent with the format used in the D2 C17 COLR. 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 Added Allowable Value to the title heading of table 1.2-1 to be consistent with ITS.	3/01
5 All	iii, 5-2, 5-4 ii, 1-1, 1-2, 2-1, 2-3, 3-1, 4-1, 5-1, 5-3, 6-1	Revised OLMCPR to reflect analysis results in response to Part 21 with notification event # 37874 Removal of historical Current Technical Specification References, identification of new Technical Specification references, and re-titling to reflect the new Technical Specifications	7/01
5	iii, 5-2, 5-4	Insert reference to allow 120°F Final Feedwater Temperature Reduction, updated Figure 5.2-2.	8/02

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.

10. Letter, David Garber (SPC) to Dr. R. J. Chin, "Dresden Unit 3 Cycle 17 Evaluation of Fuel Thermal Conductivity (Non-Proprietary Version for Exelon)," DEG:01:079, May 14, 2001.

11. Letter, DEG:01:082R, "Clarification of Results for Dresden Unit 3 Cycle 17 Fuel Thermal Conductivity Evaluation - Non-Proprietary Version for Exelon," June 14, 2001.

12. Letter, NFM-MW:02-0271, "Approval of FANP Analysis of D3C17 Feedwater Temperature with FHOOS for 120°F Reduction," July 24, 2002.

LIST OF FIGURES

LIST OF TABLES

1.0 ROD BLOCK MONITOR

1.1 Technical Specification Reference

3.3.2.1 - Control Rod Block Instrumentation

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:	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

3.2.1 - AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

3.4.1 - Recirculation Loops Operating

2.2 Description

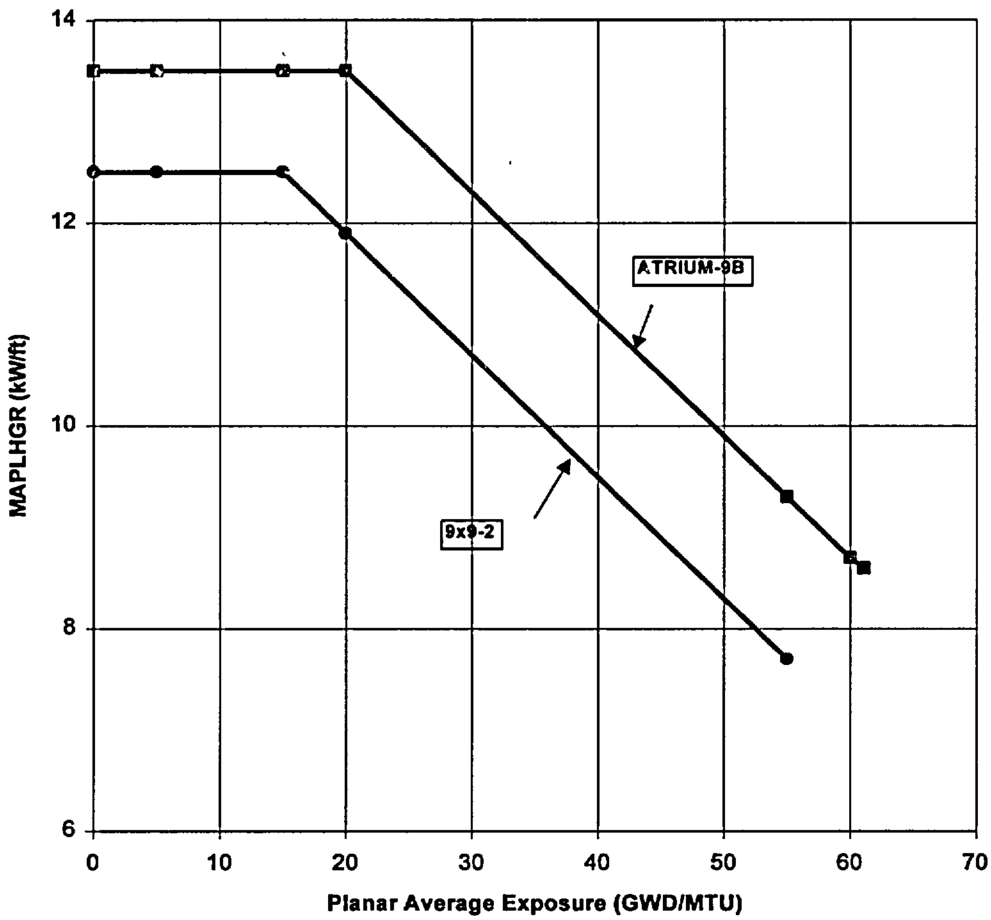
The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit versus Planar Average Exposure for each fuel type is determined from Figure 2.2-1.

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)
3.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	Single Loop Operation (SLO)	0.90	0.90
3.4.1	Recirculation Loops Operating			

3.0 LINEAR HEAT GENERATION RATE

3.1 Technical Specification Reference

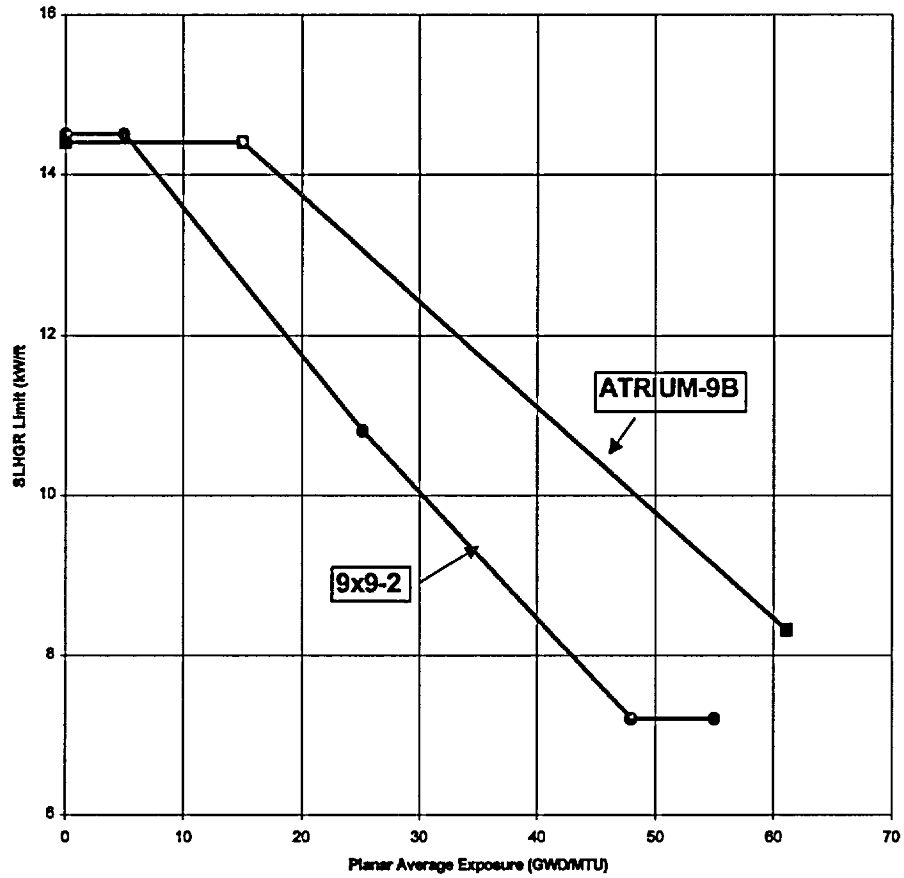
3.2.3 - LINEAR HEAT GENERATION RATE (LHGR)

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 FUEL DESIGN LIMIT RATIO - CENTERLINE MELT (FDLRC)

4.1 Technical Specification Reference

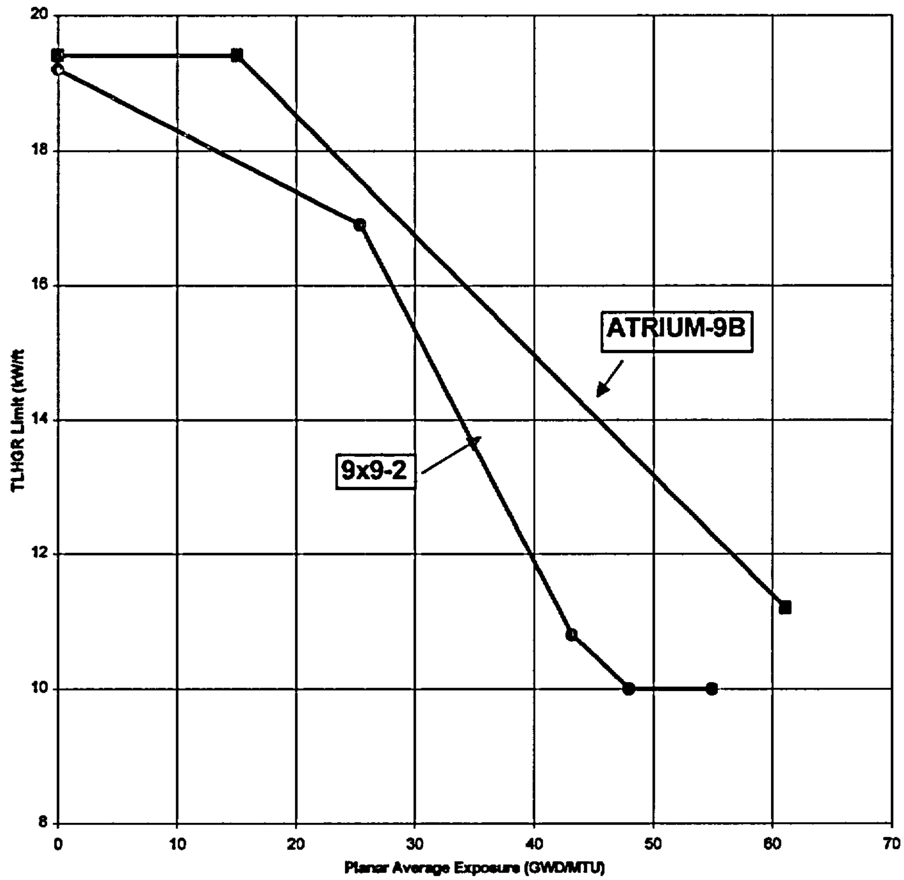
3.2.4 - Average Power Range Monitor (APRM) Gain and Setpoint

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

3.2.2 - MINIMUM CRITICAL POWER RATIO (MCPR)

3.4.1 - Recirculation Loops Operating

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 (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 ≤ 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:

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

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

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

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)

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

TABLE 5.2-1

OPERATING LIMIT MCPR

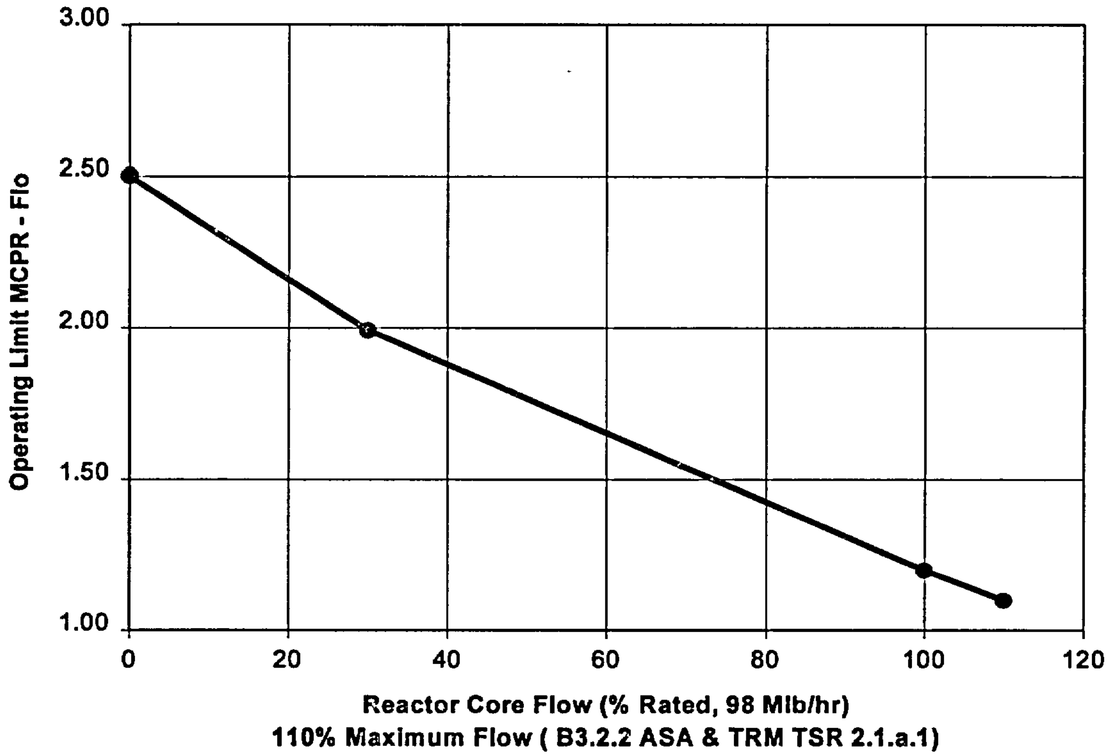
OLMCPR for Core Average Exposure ≤ **30,837** MWd/MTU **OR** OLMCPR for Core Average Exposure > **30,837** MWd/MTU with no CTP Overshoot		
Operating Scenario	**9x9-2 Fuel Operating Limit MCPR**	**ATRIUM-9B[1] Operating Limit MCPR**
Two Loop Operation and CTP maintained per 5.2.e.i[2]	1.44	1.44
Single Loop Operation and CTP maintained per 5.2.e.i[2]	1.45	1.45

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]	1.47	1.47
Single Loop Operation and CTP maintained per 5.2.e.ii[3]	1.48	1.48

[1] Includes both offset and non-offset ATRIUM-9B fuel designs.
[2] Includes up to 120 degrees 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 120 degrees F reduced feedwater temperature from normal.

FIGURE 5.2-1

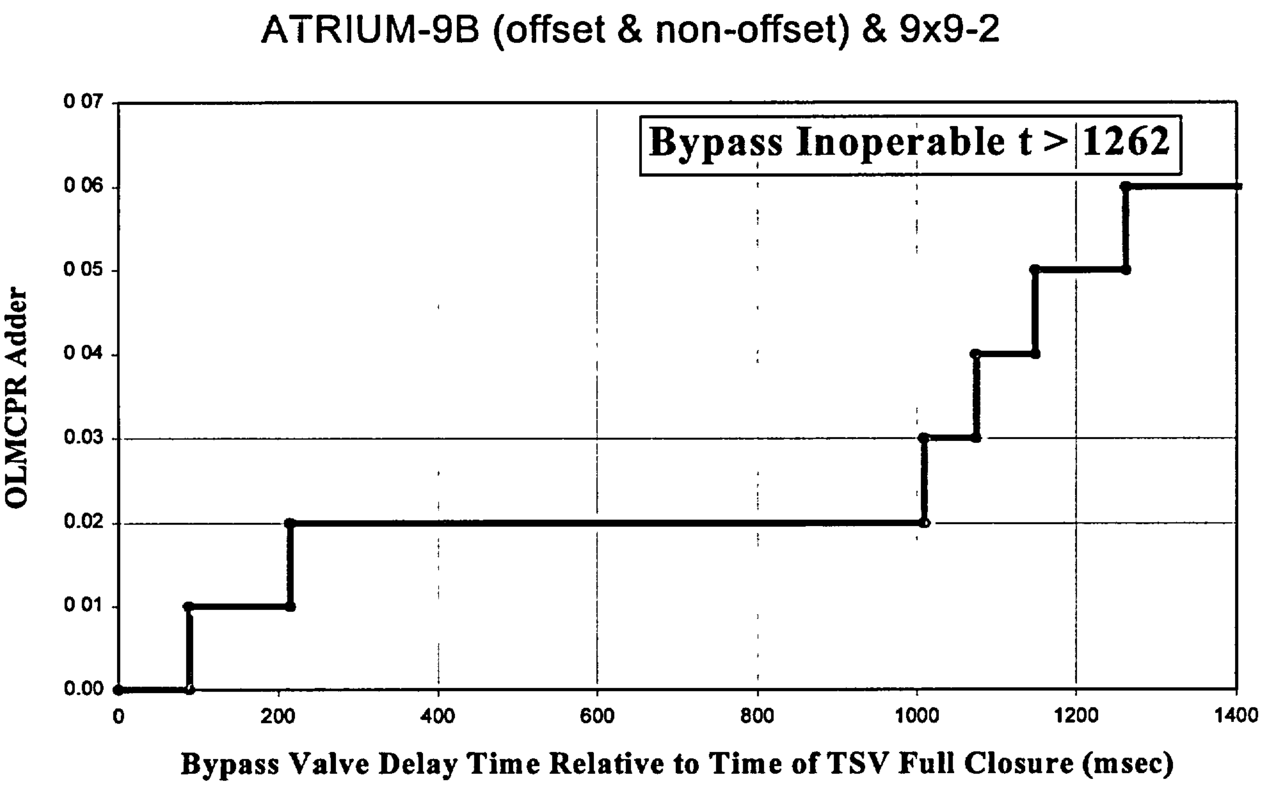
OPERATING LIMIT MCPR FOR MANUAL FLOW CONTROL



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
$0 \leq t \leq 50$	0.00
$50 < t \leq 89$	0.00
$89 < t \leq 215$	0.01
$215 < t \leq 1010$	0.02
$1010 < t \leq 1075$	0.03
$1075 < t \leq 1149$	0.04
$1149 < t \leq 1262$	0.05
$t > 1262$	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 TS section 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 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 1998.

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 LaSalle 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.