NEDC-31126
Class II
February 1986

GENERAL ELECTRIC BOILING WATER REACTOR
EXTENDED LOAD LINE LIMIT ANALYSIS
FOR
NINE MILE POINT 1
CYCLE 9

Approved: ______________________
G. L. Sozzi, Manager
Application Analysis Services

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R. L. Gridley, Manager
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CONTENTS OF THIS REPORT
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CONTENTS

TABLES

ILLUSTRATIONS

1. SUMMARY

This report justifies the expansion of the operating region of the power/flow map for the Nine Mile Point Nuclear Power Station Unit 1 (NMP-1), Cycle 9. The operating envelope is modified to include the extended operating region bounded by the 108% average power range monitor (APRM) rod block line, the rated power line, and the rated load line, as shown in Figure 1-1. In this report, rated power is defined as 1850 MWt.

The technical analysis contained in this report is referred to as the extended load line limit analysis (ELLLA) and the entire shaded area in Figure 1-1 is referred to as the ELLLA region.

The discussion and analyses presented show that the consequences of most events initiated from within the ELLLA region are bounded by the consequences of the same events initiated from the licensing basis condition for NMP-1, Cycle 9. The Loss of Feedwater Heater (LFWH) transient initiated from within the ELLLA region is the only event that is not bounded by the same event initiated from the licensing basis condition. However, the consequences of this transient will not affect the minimum critical power ratio (MCPR) operating limit for NMP-1, Cycle 9. Therefore, it is shown that all safety bases normally applied to NMP-1 are satisfied throughout Cycle 9 for operation within the ELLLA region.

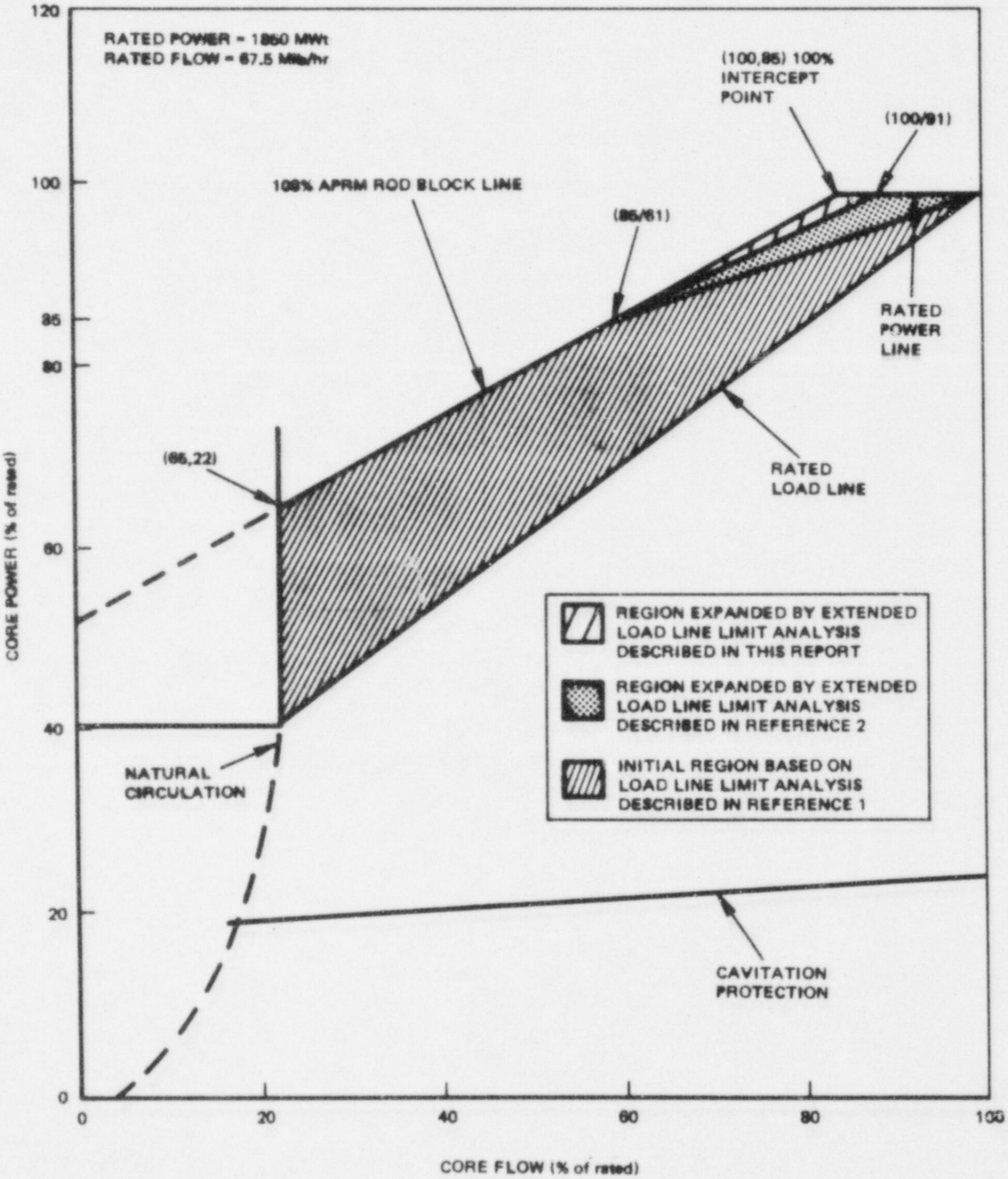

Figure 1-1. NMP-1 Operating Power/Flow Map

2. INTRODUCTION

The flexibility of a boiling water reactor (BWR) during power ascension in proceeding from the low-power/low-core-flow condition to the high-power/high-core-flow condition is limited by two factors. First, if the rated load line control rod pattern is maintained as core flow is increased, changing equilibrium xenon concentrations will result in less than rated power at rated core flow. Second, fuel pellet-cladding interaction considerations inhibit withdrawal of control rods at high power levels. The combination of these two factors can result in the inability to attain rated core power directly.

In this report the analytical bases are provided to overcome these limitations. This is accomplished by allowing operation with a rod pattern that requires few adjustments when ascending to full power. This requires an expansion of the power/flow map to allow 100% power operation at 85% flow (ELLLA). The operating envelope is modified as shown in Figure 1-1. Future reload submittals will incorporate the use of this extended load line in the analysis.

3. DISCUSSION

3.1 BACKGROUND

Previous analyses (References 1 and 2) provided the analytical bases for NMP-1 operation under a modified power/flow line designed to enable direct ascension to full power within the design bases previously applied. The load line limit analysis (LLLA), described in Reference 1 and illustrated in Figure 1-1, enabled the reactor to ascend to full power along a modified power/flow line. This line was bounded by the 108% rod block line up to the 85% power/61% flow point and from there proceeded along the rod block intercept line to the 100% power/100% flow point. An ELLLA, described in Reference 2 and also illustrated in Figure 1-1, provided analyses justifying operation up to 100% of rated power at 91% rated flow. The analysis described in this report extends the ELLLA region to now be entirely bounded by the 108% rod block line up to the rated power line. This allows operation of the plant at 100% rated power with flow as low as 85% rated.

3.2 ANALYSIS AND RESULTS

The modified power/flow curve shown in Figure 1-1 has been derived to provide relief from the operating restrictions inherently imposed during ascension to power utilizing the standard power/flow curve. Effects on the analyzed transients and accidents, which could possibly be impacted by operation in the ELLLA region, are discussed in the following sections.

3.2.1 Stability

A stability analysis was performed at the proposed extended APRM rod block line power and the natural circulation flow (65% power/22% flow). The channel hydrodynamic performance and the reactor core stability decay ratio are given in Table 3-1.

The results show that at this least stable condition, the channels and the reactor core decay ratio are within the ul ate performance criteria (1.0 decay ratio at all attainable conditions). No technical specifications for stability are required for the NMP-1 reactor.

3.2.2 Loss-of-Coolant Accident

The extended load line limit option will allow operation at rated power down to 85% core flow. The effects of this reduced core flow on the consequences of a postulated LOCA are as follows:

a. The lower initial core flow can affect the coastdown response and may yield an earlier boiling transition time.

b. The higher initial voiding in the bundle due to reduced core flow may result in a slightly earlier dryout time.

A discussion of low-flow effects on LOCA analyses for all operating plants (Reference 3) was presented to and approved by the Nuclear Regulatory Commission (Reference 4). The effects of reduced initial core flow as this applies to NMP-1 are addressed further in the following sections. These sections address the effects throughout the break spectrum and include considerations for 3, 4 or 5 recirculation loop operation. Note that full power cannot be achieved during 3-loop operation.

3.2.2.1 Small Breaks

There will be no significant effect on small break severity due to the lower initial core flow. The peak clad temperature (PCT) is sensitive to the vessel inventory, not the initial core flow. By the time the break uncovers, the core flow coastdown will have been completed. The length of time prior to core uncovery will also eliminate the effect of higher initial core voiding. Slight differences in dryout time (if any) will be insignificant compared to the time required to uncover the fuel. Current Maximum Average Planar Linear Heat Generation (MAPLHGR) limits set by the small break will remain unchanged under both normal operating condition and operation with one and two recirculation loops out-of-service.

3.2.2.2 Design Basis Accident (Large Breaks)

For the Large Break Design Basis Accidents (DBA) breaks, credit is not taken for coastdown flow due to the rapid decrease in core inlet flow. The lower coastdown flow for ELLLA operation will not affect ECCS calculations for this break size. Dryout time will be slightly faster because of the higher initial voiding, but the effect on the MAPLHGR will be negligible as shown in Reference 3. The initial voiding in the core is not dependent on the number of recirculation loops operating, but on the initial flow only; therefore, these results are applicable to 3- and 4-loop operation.

3.2.2.3 Intermediate Breaks

The lower initial core flow can affect the PCT for the intermediate breaks because the onset of transition boiling and the uncovery of the high power axial plane occurs somewhat earlier. Calculation of the LOCA results using the same conservative LOCA models which were used for the original NMP-1 analysis (Reference 5) would predict higher PCTs for the intermediate breaks during ELLLA operation at 100% power/85% core flow. This could cause the intermediate break to be limiting for certain exposures. However, incorporation of the NRC approved modified Bromley film boiling correlation (Reference 6) into the intermediate break analysis, with the estimated consequences of reduced fuel gap conductance due to increased fission gas release, results in a more realistic intermediate break PCT which is well below the 2200°F limit and approximately 70 degrees below the large break (DBA) PCT. Therefore the small break will still remain limiting at lower exposures and the large (DBA) break limiting at higher exposures. Note that PCT constraints set MAPLHGR limits for the small break and maximum oxidation fraction constraints set MAPLHGR limits for the large break.

3.2.2.4 LOCA Summary

Based on the preceding discussion, the standard Cycle 9 MAPLHGR values listed in Reference 5 are applicable for ELLLA operation in 3-, 4- and 5-loop operation.

3.2.3 Containment Response

The impact of plant operation in the proposed domain for NMP-1 has been evaluated for the containment LOCA response. The operating condition was 102% of rated power and 85% flow. The results show no impact on the containment LOCA response. The maximum drywell pressurization rate observed is less than the value used in plant-unique testing for defining LOCA-related pool swell loads.

3.2.4 Transients

As shown in Reference 7, the most limiting transient event for NMP-1 Cycle 9 is the Turbine Trip Without Bypass event.

For the ELLLA, the following transient events were analyzed at the 100% power intercept point (100% power/85% flow): Turbine Trip Without Bypass, Feedwater Controller Failure and Loss of Feedwater Heater. These analyses were performed using the nuclear parameters resulting from the end of cycle (EOC) and EOC-1000 MWd/ST target exposure shapes, consistent with rated operation.

The results for both the licensing basis case and the reduced core flow case are shown in Table 3-2. Comparison of the initial conditions for these cases is presented in Tables 3-3 and 3-4. As shown in Table 3-2, the (100% power/85% flow) transient results are bounded by the licensing basis case (100% power/ 100% flow) for the Turbine Trip Without Bypass and the Feedwater Controller Failure transients. The Loss of Feedwater Heating (LFWH) transient, which is an exposure independent event, was not bounded by the licensing basis case. However, the increased ΔCPR result from this event is still bounded by the Cycle 9 licensing basis limiting MCPR result.

3.2.5 ASME Pressure Vessel Code Compliance

The Main Steam Isolation Valve (MSIV) Closure With No Scram event is used to determine compliance with the ASME Pressure Vessel Code. This event was analyzed at the 100% power intercept point (100% power/85% flow) using the nuclear parameters resulting from the EOC target exposure shape. The resulting peak vessel pressure is shown in Table 3-5 and is still below the design pressure of 1375 psig.

3.2.6 Rod Withdrawal Error

A Rod Withdrawal Error analysis is not required for the 100% Power and 85% Flow Extended Load Line because the APRM rod block line slope (0.55) does not change. The initial 100% power and 85% flow state's proximity to the APRM rod block line, as compared to previously analyzed states, results in a reduction of rod motion prior to a rod block. This reduced rod motion prior to a rod block results in a reduction of the ΔCPR at each rod block setpoint. Therefore, the ΔCPRs of the 100% power/100% flow state will bound the 100% power and 85% flow state.

3.3 CONCLUSION

The results of most of the transients for the 100% power intercept point (100% power/85% flow) are bounded by the same transients for the licensing basis point (100% power/100% flow). The only exception is the LFWH transient; however, this event does not affect the current Cycle 9 licensing basis MCPR operating limit. The containment LOCA response is unaffected at the operating condition of 102% power and 85% flow. The overpressurization protection analysis results are within ASME Pressure Vessel Code allowable for the 100% power/85% flow point. The stability results are within the bounds of the ultimate performance criteria, ≤ 1.0 decay ratio, and the MAPLHGR results are unchanged by the extended operating region.

Therefore, it is concluded that all safety bases normally applied to NMP-1 are satisfied throughout Cycle 9 for operation within the ELLLA region.

Table 3-1

STABILITY RESULTS

Rod Line Analyzed:	Extrapolated Rod Block Line Natural Circulation Power
Reactor Core Stability Decay Ratio X_2/X_0:	0.74
Channel Hydrodynamic Performance Decay Ratio X_2/X_0: (P8x8R)	0.74

Table 3-2

TRANSIENT RESULTS

	Exposure (MWd/ST)	Initial Power/ Flow (% NBR)	Peak Neutron Flux (% NBR)	Peak Heat Flux (% NBR)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)	Δ CPR P8x8R
Turbine Trip	EOC-1000	100/100	686.0	121.3	1270	1285	0.270
without Bypass	EOC-1000	100/85	646.0	120.5	1269	1284	0.233
Turbine Trip	EOC	100/100	695.8	125.9	1292	1306	0.340
without Bypass	EOC	100/85	665.3	122.0	1282	1297	0.272
Feedwater Con-	EOC-1000	100/100	157.4	109.4	1139	1176	0.078
troller Failure	EOC-1000	100/85	148.2	108.8	1140	1173	0.073
Feedwater Con-	EOC	100/100	205.9	114.9	1145	1184	0.156
troller Failure	EOC	100/85	181.3	112.1	1143	1177	0.098
Loss of	-	100/100	115.9	115.4	1017	1073	0.142
Feedwater Heater[a]	-	100/85	118.5	117.7	1019	1075	0.158

[a]Exposure independent event

Table 3-3

TRANSIENT INPUT DATA AND OPERATING CONDITIONS

	Licensing Basis Point (100% power/100% flow)	Intercept Point (100% power/85% flow)
Thermal Power (MWt/%)	1850/100	1850/100
Steam Flow (Mlb/hr/%)	7.32/100	7.33/100
Core Flow (Mlb/hr/%)	67.5/100	57.4/85
Dome Pressure (psig)	1030	1029
Turbine Pressure (psig)	950	949
Relief Valves (No./% NBR)	6/45.7	6/45.7
Low Setpoint (psig)[a]	1102	1102
Spring Safety Valves (No./% NBR)	16/140.8	16/140.8
Low Setpoint (psig)[a]	1230	1230

[a]Nominal setpoint + 1%

Table 3-4

GETAB ANALYSIS INITIAL CONDITIONS

	Licensing Basis Point (100% power/100% flow)	Intercept Point (100% power/85% flow)
Core Power (MWt)	1850	1850
Core Flow (Mlb/hr)	67.5	57.4
Core Pressure (psig)	1045	1042
Inlet Enthalpy (Btu/lb)	526.6	522.3
Nonfuel Power Fraction	0.04	0.04
Axial Peaking Factor	1.40	1.40
P8x8R Fuel		
Local Peaking Factor	1.20	1.20
Radial Peaking Factor	1.71[a]/1.63[b]	1.69[a]/1.64[b]
R-Factor	1.051	1.051
Bundle Power (MWt)	5.806[a]/5.542[b]	5.725[a]/5.572[b]
Bundle Flow (10^3 lb/hr)	102.14[a]/104.09[b]	86.05[a]/87.03[b]

[a]EOC - 1000 MWd/ST
[b]EOC

Table 3-5

ASME PRESSURE VESSEL CODE COMPLIANCE: MSIV CLOSURE (NO SCRAM)

Exposure (MWd/ST)	Initial Power/Flow (% NBR)	Peak Neutron Flux (% NBR)	Peak Heat Flux (% NBR)	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
EOC	100/100	572	147	1289	1328
EOC	100/85	574	145	1290	1329

4. REFERENCES

1. "Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis License Amendment Submittal," General Electric Company, May 1977 (NEDO-24012).

2. J. V. Woodford, "Nine Mile Point Nuclear Power Station Unit 1 Extended Load Line Limit Analysis License Amendment Submittal (Cycle 6)," General Electric Company, April 1979 (NEDO-24185).

3. Letter from R. L. Gridley (GE) to D. G. Eisenhut (NRC), "Review of Low Core Flow Effects on LOCA Analysis for Operating BWRs," May 8, 1978.

4. Letter from D. G. Eisenhut (NRC) to R. L. Gridley (GE), enclosing "Safety Evaluation Report Revision of Previously Imposed MAPLHGR (ECCS-LOCA) Restrictions for BWRs at Less Than Rated Flow," May 19, 1978.

5. "Loss-of-Coolant Accident Analysis Report for Nine Mile Point Unit One Nuclear Power Station," General Electric Company, August 1981 (NEDO-24348, as amended).

6. J. E. Leonard, et al., "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K-Amendment No. 1 - Calculation of Low Flow Film Boiling Heat Transfer for BWR LOCA Analysis," General Electric Company, October 1982 (NEDO-20566-1-A, Rev. 1).

7. "Supplemental Reload Licensing Submittal for Nine Mile Point Nuclear Power Station Unit 1, Reload 10," General Electric Company, October 1985 (23A4717).