ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

DOCKET NO. 50-220

Proposed Changes to Technical Specifications (Appendix A)

Existing pages 8, 18 (Bases), 64e, 70a(Bases) and 70d (Bases) will be replaced with the attached revised pages. Text has been retyped and graphs have been redrawn on these pages. Marginal markings have been added to these pages to indicate changes.

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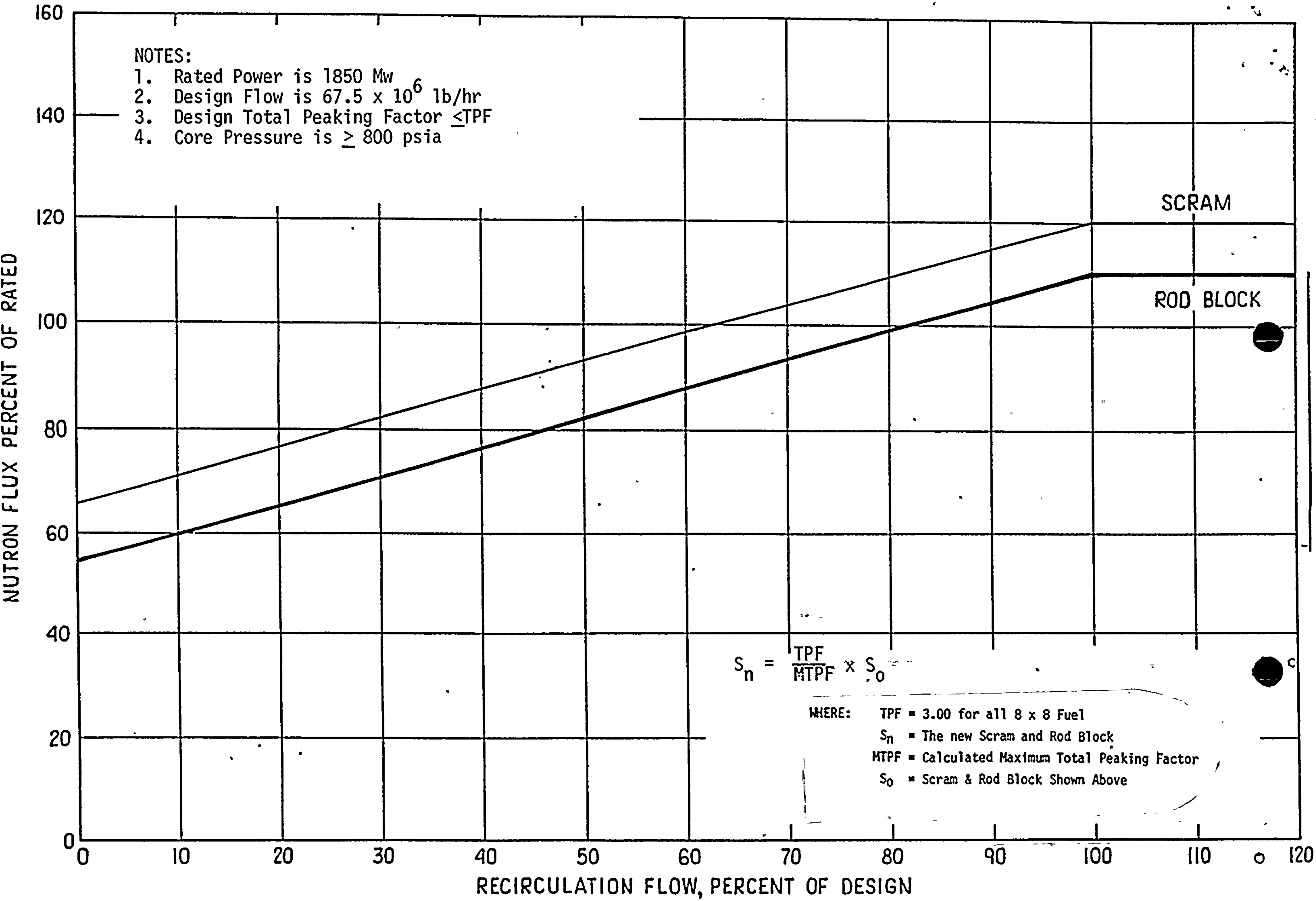

FIGURE 2.1.1 FLOW BIASED SCRAM AND APRM ROD BLOCK

steady-state operation is at 110% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the design peaking factor, thus, preserving the APRM rod block safety margin.

g-h. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line isolation on reactor low pressure and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at $\leq$10% valve closure, there is no increase in neutron flux and peak pressure in the vessel dome is limited to 1141 psig. (8, 9, 10).

The operator will set the pressure trip at greater than or equal to 850 psig and the isolation valve steam position scram setting at less than or equal to 10% of valve stem position from full open. However, the actual pressure set point can be as much as 15.8 psi lower than the indicated 850 psig and the valve position set point can be as much as 2.5% of stem position greater. These allowable deviations are due to instrument error, operator setting error and drift with time.

In addition to the above mentioned LS^3, other reactor protection system devicves (LCO 3.6.2) serve as a secondary backup to the LS^3 chosen. These are as follows:

High fission product activity released from the core is sensed in the main steam lines by the high radiation main steam line monitors. These monitors provide a backup scram signal and also close the main steam line isolation valves.

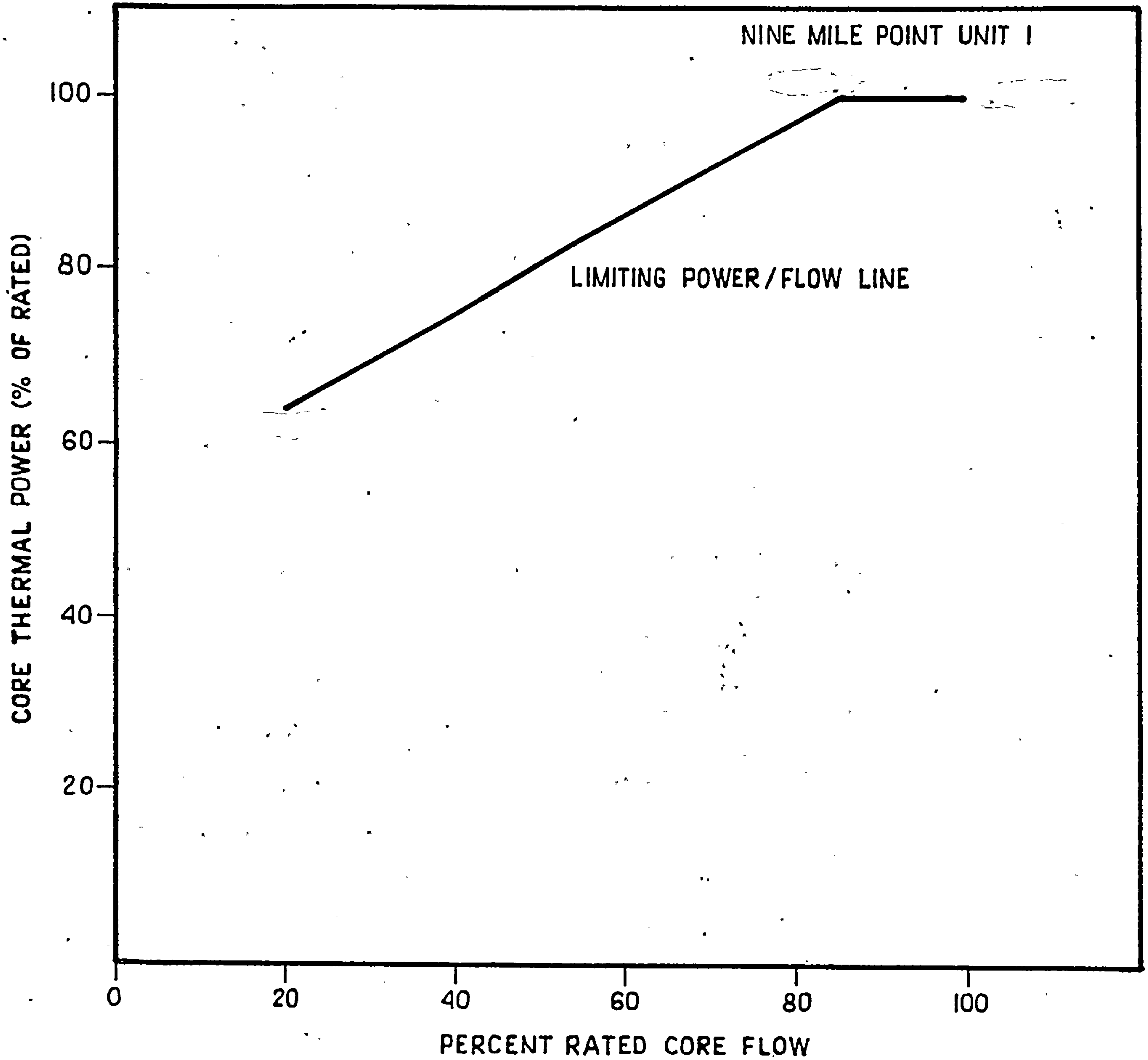

Figure 3.1.7.aa LIMITING POWER FLOW LINE

of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

Figure 3.1.7-1 is used for calculating MCPR during operation at other rated conditions. For the case of automatic flow control, the K_f factor is determined such that any automatic increase in power (due to flow control) will always result in arriving at the nominal required MCPR at 100% power. For manual flow control, the K_f is determined such that an inadvertent increase in core flow (i.e, operator error or recirculation pump speed controller failure) would result in arriving at the 99.9% limit MCPR when core flow reaches the maximum possible core flow corresponding to a particular setting of the recirculation pump MG set scoop tube maximum speed control limiting set screws. These screws are to be calibrated and set to a particular value and whenever the plant is operating in manual flow control the K_f defined by that setting of the screws is to be used in the determination of required MCPR. This will assure that the reduction in MCPR associated with an inadvertent flow increase always satisfies the 99.9% requirement. Irrespective of the scoop tube setting, the required MCPR is never allowed to be less than the nominal MCPR (i.e. K_f is never less than unity).

Power/Flow Relationship

The power/flow curve is the locus of critical power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis (7, 8, 12, 14) justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

REFERENCES FOR BASES 3.1.7 AND 4.1.7 FUEL RODS

(1) "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7 and 8, NEDM-10735, August 1983

(2) Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

(3) Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

(4) "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Supplement 1 to Revision 1, December 1974.

(5) "General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K," NEDO-20566.

(6) General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G.L. Gyorey to Victor Stello Jr., dated December 20, 1974.

(7) "Nine Mile Point Nuclear Power Station Unit 1, Load Line Limit Analysis," NEDO-24012.

(8) Licensing Topical Report General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P-A, August 1978.

(9) Final Safety Analysis Report, Nine Mile Point Nuclear Station, Niagara Mohawk Power Corporation, June 1967.

(10) NRC Safety Evaluation, Amendment No. 24 to DPR-63 contained in a lettter from George Lear, NRC, to D. P. Dise dated May 15, 1978.

(11) "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1," NEDO-10722A.

(12) Nine Mile Point Nuclear Power Station Unit 1, Extended Load line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(13) Loss of Coolant Accident Analysis Report for Nine Mile Point Unit One Nuclear Power Station, NEDO-24348, August 1981.

(14) General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Nine Mile Point 1 Cycle 9, NEDC-31126, February 1986.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-63

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Supporting Information and No Significant Hazards Considerations Analysis

The proposed amendment to Sections Figures 2.1.1 and 3.1.7.aa of the Technical Specifications reflect changes required to prevent unnecessary restriction of operating ranges and provide consistency with the current reload analyses for the present Nine Mile Point Unit 1 core configuration and parameters (Reference 1 and 2). Reference 2 has been enclosed for your convenience.

At present, the technical specifications limit full power operation to the flow range from 91 percent to 100 percent of rated core flow (67.5 MLBs/hr). Below 91 percent flow thermal power is limited to the line defined by P = .55w + 50 (P = percent thermal power and w = percent core flow). In practice, this region is further restricted by the APRM rod block line. The Rod Block Line (RBL) is limited by the technical specifications to RBL = .55w + 50. This is the same as the power flow limit. In order to ensure compliance with this limit, Instrument and Control sets the RBL slightly below the technical specification limit. This effectively limits full power operation to a flow range of about 94 to 100 percent core flow. The proposed modification will change the range of full power operation to be between 85 percent and 100 percent. Below 85 percent flow the power level will be limited by P = .55w + 53. The RBL will be defined by RBL = .55w + 55. Thus, even with the conservative setting of this line, it will not constrict the operating map. The present Power/Flow (P/F) line and the new P/F line are shown in Figure 1.

10CFR50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis, using the standards in Section 50.92 about the issue of no significant hazards consideration. Therefore, in accordance with 10CFR50.91 and 10CFR50.92, the following analysis has been performed:

The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

At present, the rod block line is given by the line RBL = .55w + 50. This line coincides with the P/F line. With the expanded P/F map, this line will be moved to RBL = .55w + 55. This line is above the P/F line. Therefore, conservative setting of the RBL (done to ensure that the drift of the RBL will not result in a technical specification violation) will not restrict the low flow operation. The only analysis affected by the RBL change is the rod withdrawal error. The change in the RBL increases the delta CPR for this

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event from .30 to .33. This transient is the limiting transient from BOC to EOC minus 2,000 MWD/ST. The MCPR limit for this exposure range increases from 1.37 to 1.40. The current technical specification limit is 1.40 for the exposure range during which the rod withdrawal error is limiting. Accordingly, no change in the technical specification MCPR limit is required.

Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

For events of moderate frequency, the NRC requires that the spring safety valves be calculated not to open. The limiting transient for this is the MSIV closure with scram. This transient is limiting at the 100/100 point.

Operation at the 100/85 power/flow point (in accordance with Appendix K requirements, the analysis was done at 102/85) will not require any change in MAPLHGR limits. For small breaks, the change in initial core flow has no effect since the recirculation pump coastdown is complete long before the core uncovers. Similarly, in a large break (DBA), core flow has no effect since the inventory decreases so rapidly. During the intermediate size break, the pump coastdown affects the core uncovery time and thus, the Peak Cladding Temperature (PCT) can increase. Incorporating the change in pump coastdown, the intermediate break is shown to have a PCT 70°F lower than the DBA. Intermediate break sizes, therefore, are not limiting.

A LOCA initiated during operation at the 100/85 power/flow point (the analysis was done at 102/85) does not exceed the maximum drywell pressurization rate allowed for NMP1. Accordingly, the containment requirements are met by the new P/F map.

Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The transients analyzed in the Cycle 9 reload licensing submittal (Reference 1) were reanalyzed to determine if operation at the 100/85 point will change any operating safety limits. As shown in Reference 2, no technical specification changes (other than changing the P/F map and the rod line) are affected. In all but three transients, results for initiation from the 100/85 point are bound by the 100/100 point. The loss of feedwater heating transient results in an increase in the delta Critical Power Ratio (CPR) of this transient when initiated from the 100/85 point. This transient in either case is not the limiting transient. Therefore, the increase does not affect a safety margin in the technical specifications. The peak pressure during the MSIV closure without scram increases by 1 psi. This is still 46 psi below the 1375 psi ASME Pressure Vessel Code limit. The rod withdrawal error analysis was discussed above.

For cores with conventional General Electric fuel designs and loading patterns typical of previous patterns, cycle-by-cycle stability analysis is not required as discussed in NRC Generic Letter 86-02. This is true of Reload 10, however, the P/F region was increased. Therefore, a stability analysis was performed. This analysis shows that with the expanded P/F map, the requirements of GL 86-02 are met.

Therefore, this change will not involve a significant reduction in a margin of safety.

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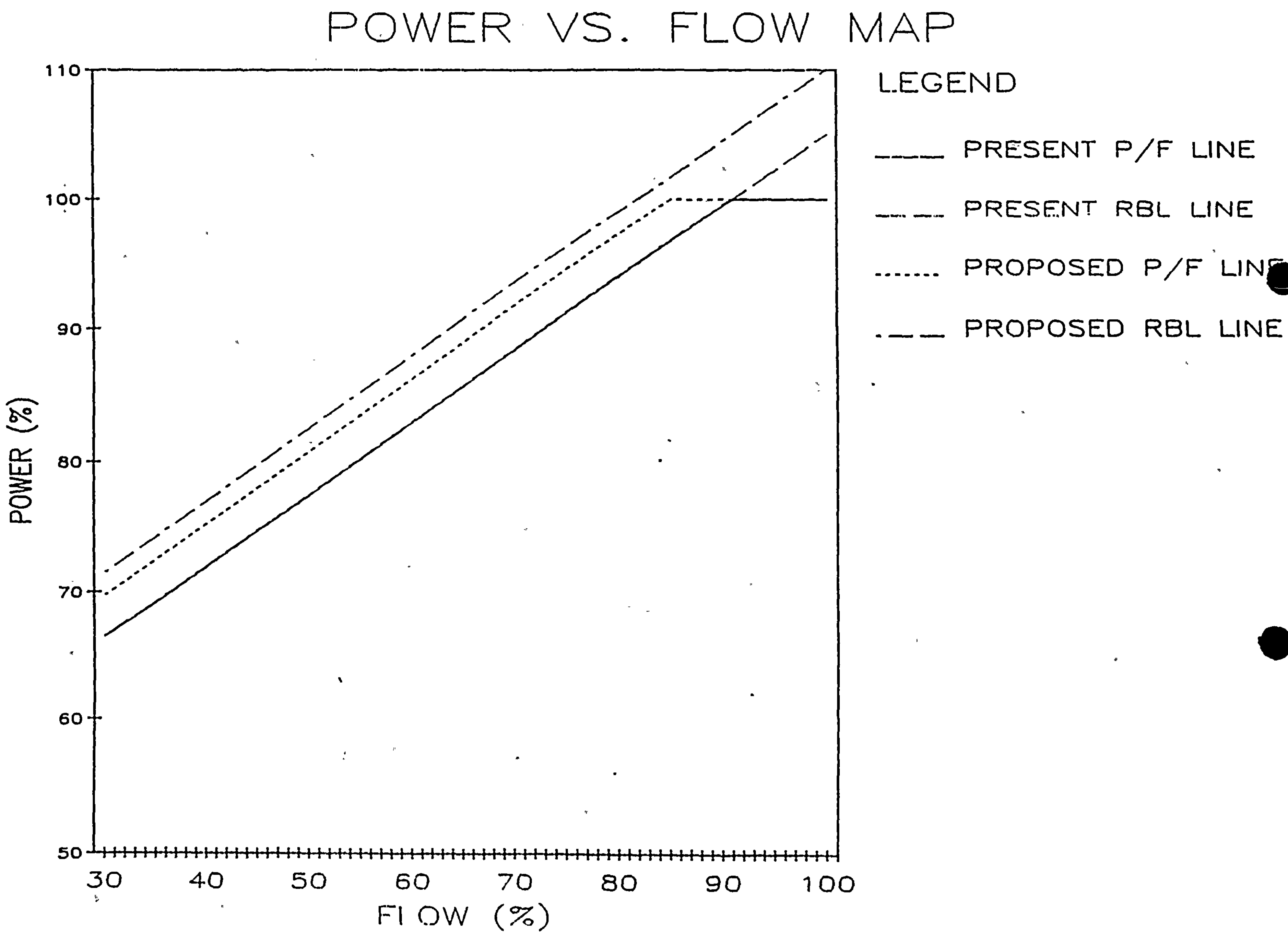
POWER VS. FLOW MAP
LEGEND
PRESENT P/F LINE
PRESENT RBL LINE
PROPOSED P/F LINE
PROPOSED RBL LINE
POWER (%)
FLOW (%)
110
100
90
80
70
60
50
30
40
50
60
70
80
90
100

Figure 1

References

1. "Supplemental Reload Licensing Submittal for NMP1 Reload 10," GE, October 1985 (23A4717).

2. "GE BWR Extended Load Line Analysis License Amendment Submittal (Cycle 6)," GE, February 1986 (NEDO-31126).

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