



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-63

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated October 30, 1986, as supplemented January 15, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-63 is hereby amended to read as follows:

8703310559 870324
PDR ADOCK 05000220
P PDR



1950-1951

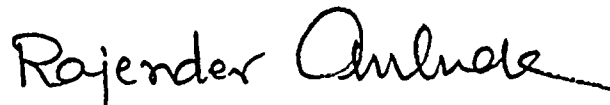
1952-1953

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 92, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

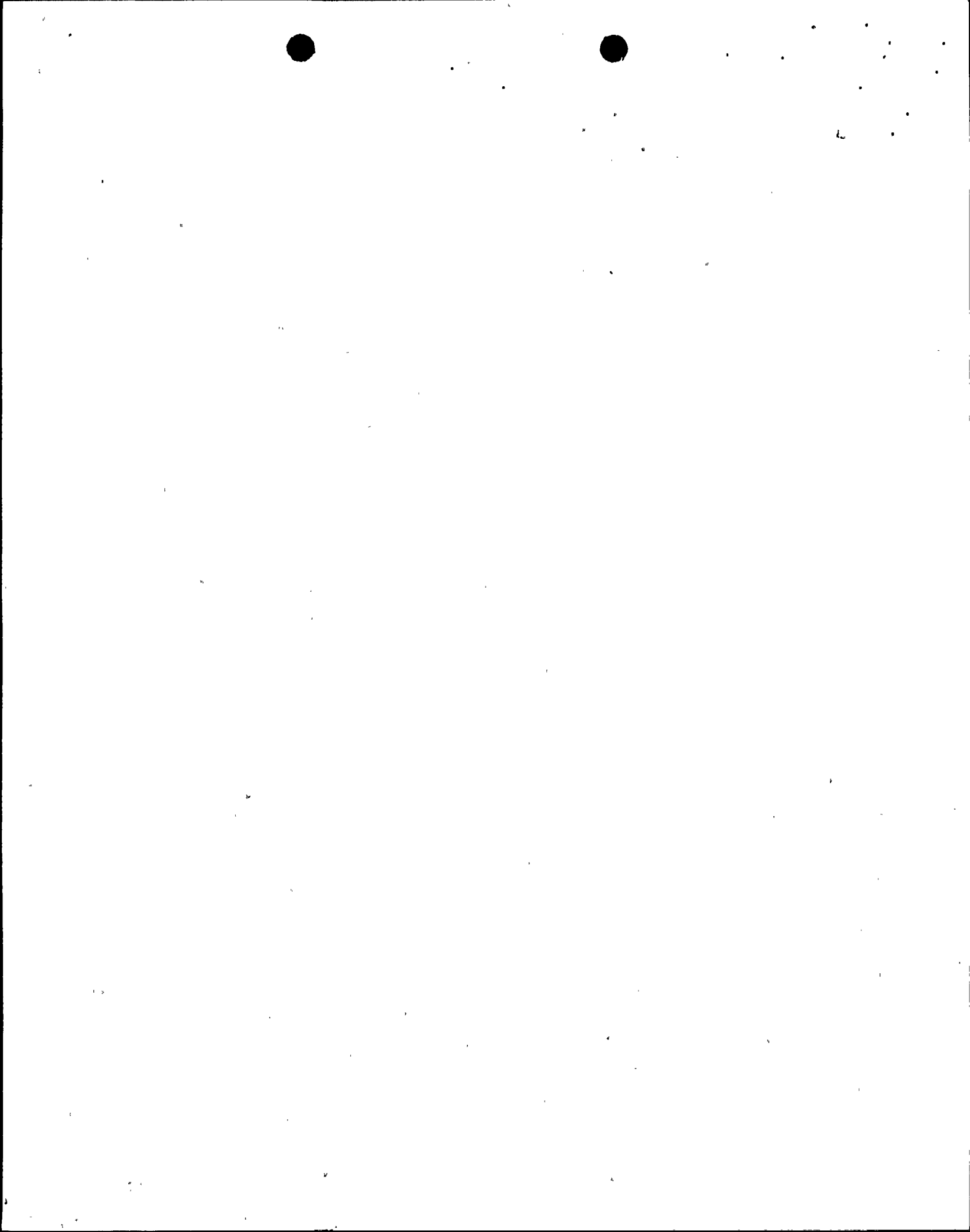
FOR THE NUCLEAR REGULATORY COMMISSION



Rajender Auluck, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 24, 1987



ATTACHMENT TO LICENSE AMENDMENT NO. 92

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

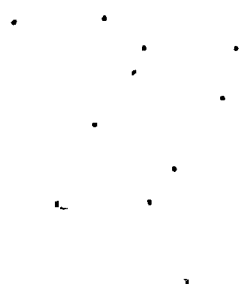
Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE

8
18
64e
70a
70d

INSERT

8
18
64e
70a
70d



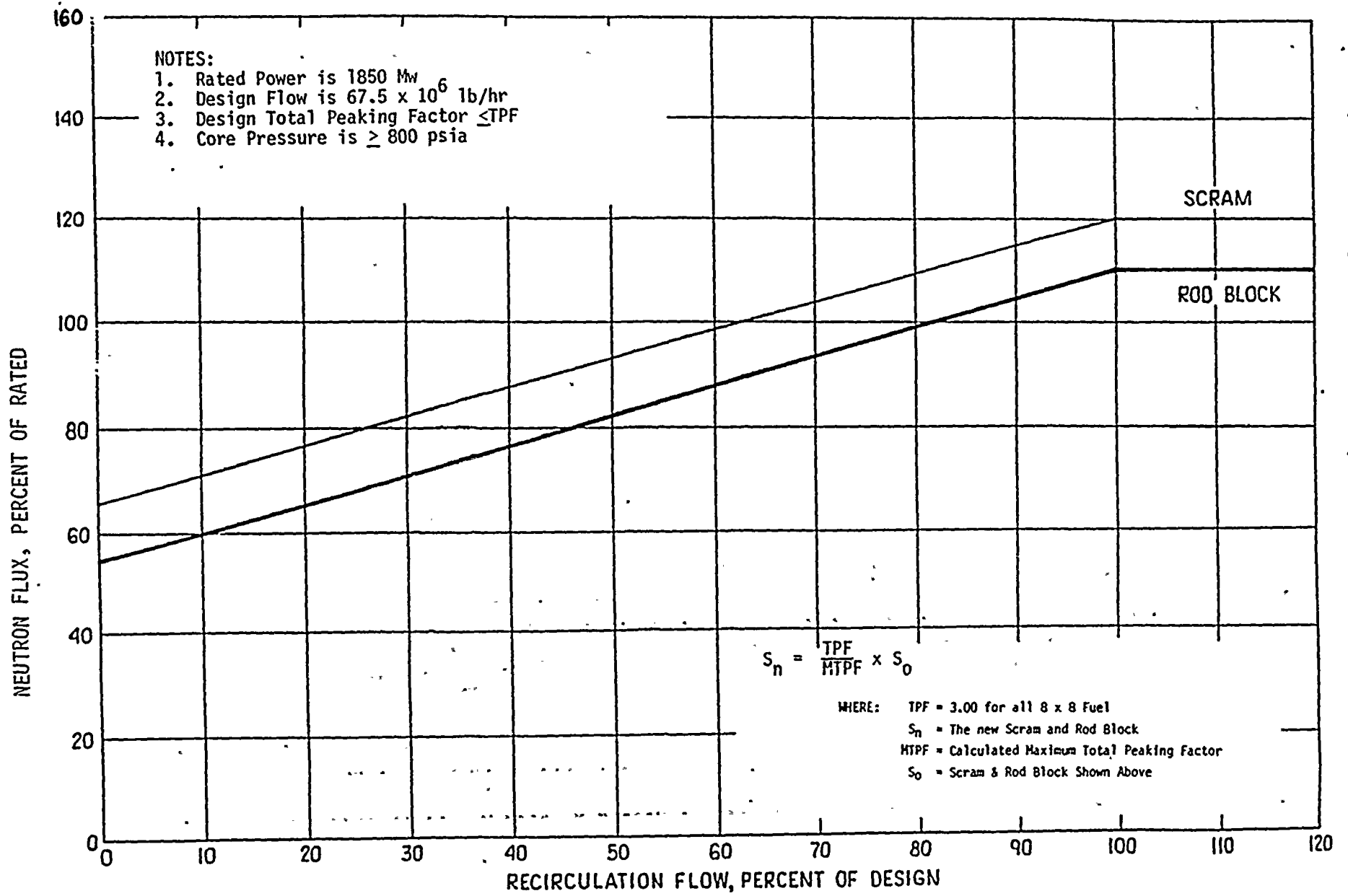
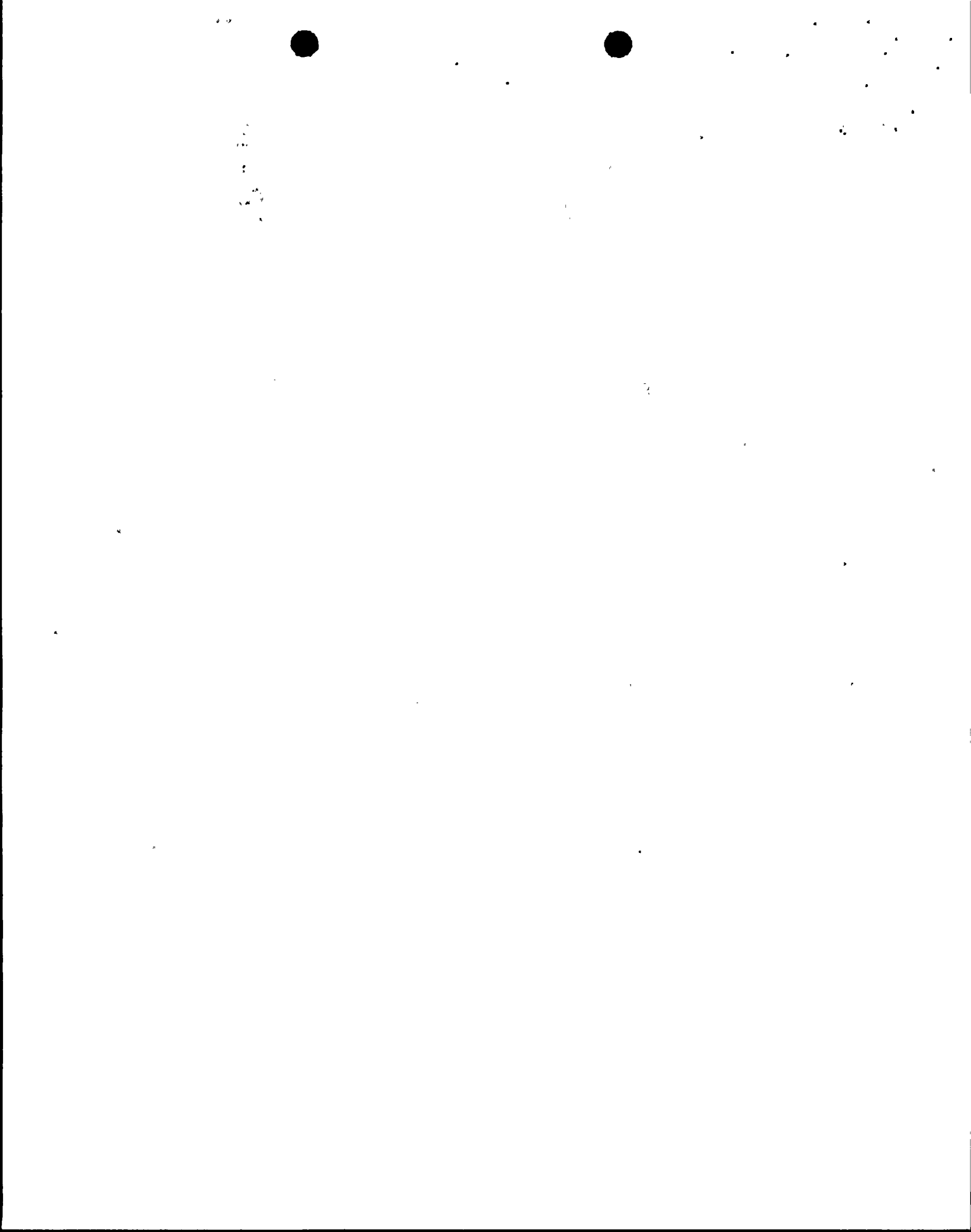


FIGURE 2.1.1 FLOW BIASED SCRAM AND APRM ROD BLOCK



BASES FOR 2.1.2 FUEL CLADDING - LS³

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.

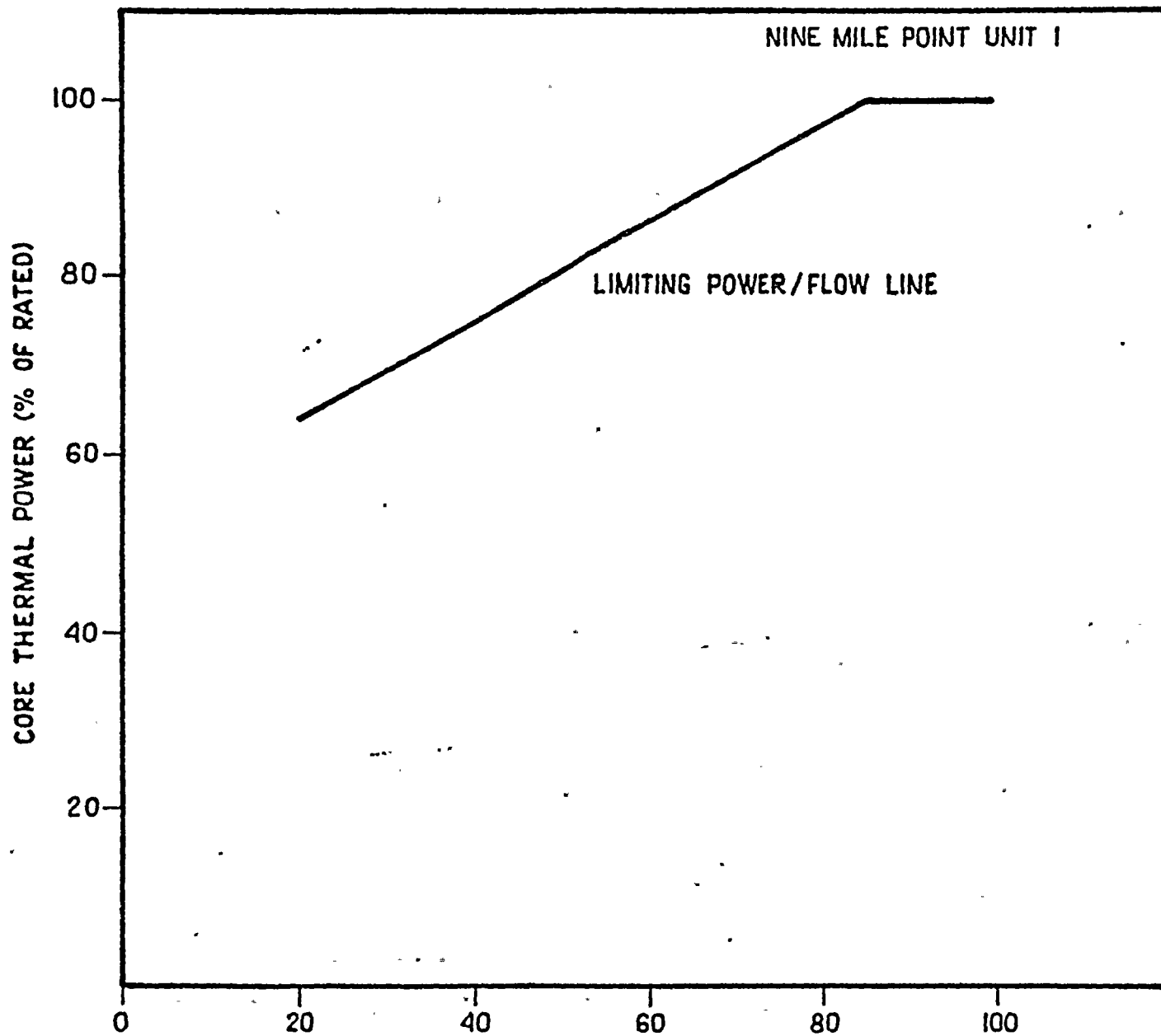
- g-h. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide 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 stem 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 LS³, other reactor protection system devices (LCO 3.6.2) serve as a secondary backup to the LS³ 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.







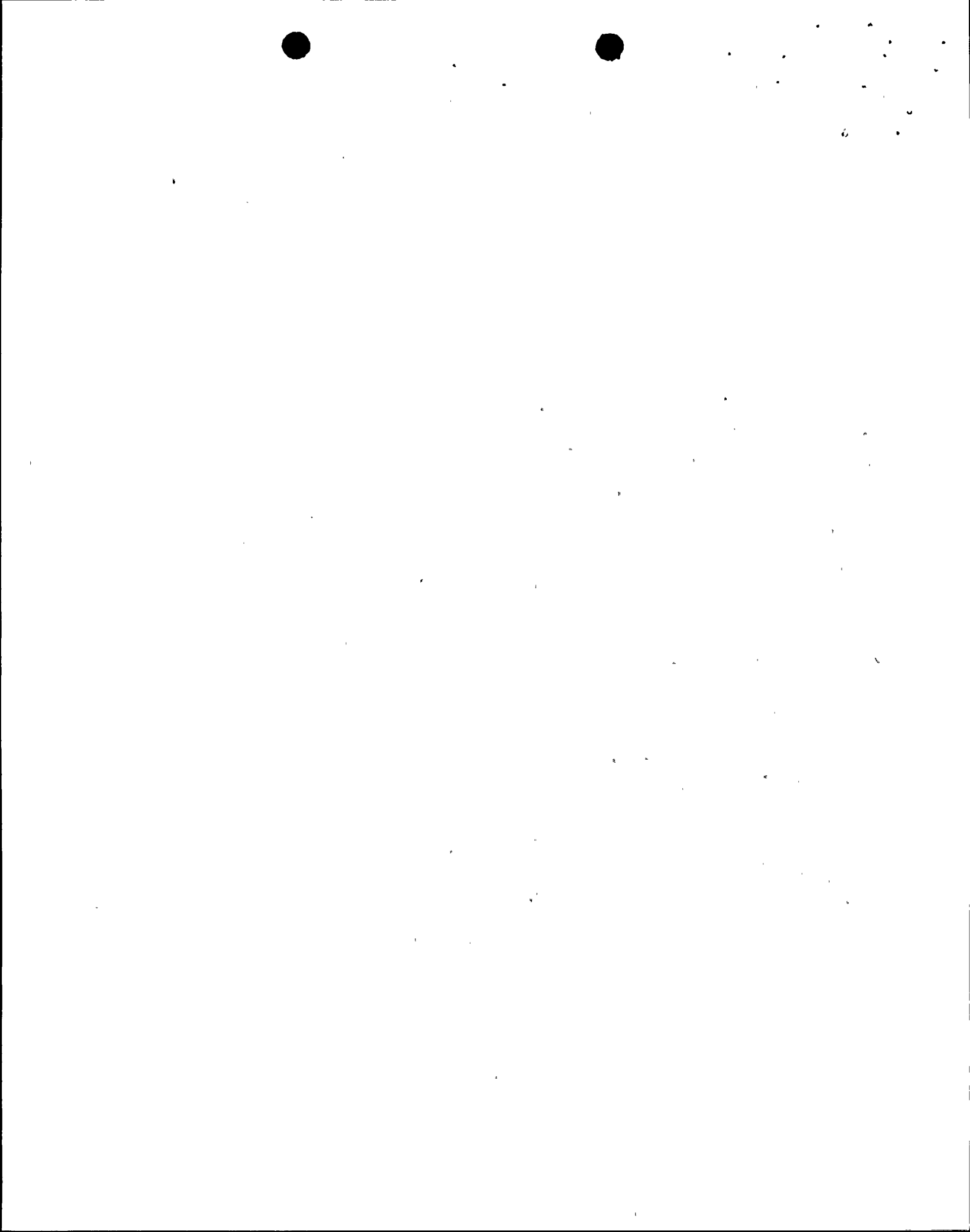
BASES FOR 3.1.7 AND 4.1.7 FUEL RODS

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 1973.
- (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 letter from George Lear, NRC, to D. P. Dize 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.

