Consumers Power Company

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April 14, 1982

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Harold R Denton, Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

MIDLAND PROJECT
MIDLAND DOCKET NO 50-329, 50-330
SAFETY EVALUATION REPORT INFORMATION
FILE: 0505.16 SERIAL: 16627

ENCLOSURES: (1) Steam Generator Inventory for Safety Analyses
(2) Fuel in DNB for Partial Pump Locked Rotor Analysis
(3) Locked Rotor with LOOP at Turbine Trip
(4) Performance Characteristics for Midland Units 1 and 2
(5) B&W Memorandum, Hot Shutdown-Reactivity Effects with All Rods Inserted

Enclosures (1) through (5) provide information requested by the NRC Staff to facilitate their review of the Midland FSAR with respect to generating the Safety Evaluation Report. This information will be incorporated in a forthcoming FSAR revision and will close out the outstanding items involved.

James W. Cook

JWC/JRW/fms

CC RJCook, Midland Resident Inspector, w/a
RHernan, USNRC, w/a
RWHuston, Washington, w/a
DBMiller, Midland, w/a

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CONSUMERS POWER COMPANY
Midland Units 1 and 2
Docket No 50-329, 50-330

Letter Serial 16627 Dated April 14, 1982

At the request of the Commission and pursuant to the Atomic Energy Act of 1954, and the Energy Reorganization Act of 1974, as amended and the Commission's Rules and Regulations thereunder, Consumers Power Company submits the following responses to outstanding NRC Staff items:

(1) Steam Generator Inventory for Safety Analyses

(2) Fuel in DNB for Partial Pump Locked Rotor Analysis

(3) Locked Rotor with LOOP at Turbine Trip

(4) Performance Characteristics for Midland Units 1 and 2

(5) B&W Memorandum, Hot Shutdown - Reactivity Effects with All Rods Inserted

CONSUMERS POWER COMPANY

By J W Cook
J W Cook, Vice President
Projects, Engineering and Construction

Sworn and subscribed before me this 19 day of April 1982.

Barbara Townsend
Notary Public
Jackson County, Michigan

My Commission Expires September 8, 1984

mi0482-0866d131

STEAM GENERATOR INVENTORY FOR SAFETY ANALYSES

The steam generator inventory assumed as an initial condition for a secondary system upset exerts a large influence on the consequences of the transient. Early steam generator inventory predictions for low and high power 177 operating plants of 55,000 and 62,500 lb, respectively, were used for double-ended steam line break analyses. These inventory predictions are a conservatively large estimation of the maximum amount of mass that could be present in the steam generator tube and downcomer region that would still result in a guaranteed minimum superheat production of 35° F during rated power operation with fouled steam generators. These predictions are based on consideration of actual steam generator geometry and the matching of tube region temperature distributions with an Alliance Research Center test steam generator model in order to obtain a profile of fluid density along the steam generator tube region.

These inventory predictions have, over the past several years, been refined to more accurately reflect true generator inventory. Operating experience at ANO-1 has indicated a very slight degradation in steam generator exit steam temperature as a function of fouling which was less than originally predicted. This results in a lower calculated inventory for fouled conditions than was previously considered necessary.

Steam generator inventory prediction with clean conditions has become necessary with the advent of loss of feedwater calculations. In general, core overheating transients are made more severe in RC pressure response by using a low prediction of secondary inventory.

A conservative approach to steam generator inventory as an initial condition for transient analyses is still ensured by using a nominal value ±10% in the direction of known conservatism.

FUEL IN DNB FOR PARTIAL PUMP LOCKED ROTOR ANALYSIS

The analysis of the locked reactor coolant pump rotor from an initial 2 pump (1/1) operation at 1251 MWt conservatively predicts that 27% of the core will experience a minimum DNBR below 1.3 (BAW-2). A calculation of maximum cladding surface temperature indicates a maximum surface temperature of about 1100° F a few seconds after film boiling is predicted. This surface temperature is considered to be insufficient to initiate appreciable Zr-H_2O reaction or result in clad damage by melting. The maximum centerline fuel temperature is calculated to increase less than 25° F above the nominal of 2250° F for operation at 50% power. In view of the small change in transient power and the fuel temperature excursion predicted for this transient, the potential for fuel cladding failure due to a rupture is negligible. Internal fuel pin pressure does not significantly increase as a result of the fuel temperature change. Therefore, rupture at low cladding temperature is precluded.

LOCKED ROTOR WITH LOOP AT TURBINE TRIP

The reactor coolant pump shaft seizure event is a non-mechanistically postulated design basis event. Following the seizure of one RC pump rotor, forced RC flow is assumed to continue using the unaffected pump(s). The transient results in reactor trip followed shortly thereafter by turbine trip.

The consequences of a loss of offsite power during the transient are considered to have an inconsequential impact on the film boiling predictions. A loss of offsite power at the initiation of the event would result in an early reactor trip thus improving the DNB response. A loss of forced reactor coolant flow initiated at the time of turbine trip following reactor trip results in approximately an additional 2% flow degradation in the second following initiation. The RC flow degradation in this time after reactor trip has an inconsequential impact on the prediction of the DNB response.

PERFORMANCE CHARACTERISTICS FOR MIDLAND UNITS 1 AND 2

Parameters	4-Pump	3-Pump	2-Pump
Design Power, % of 2452 MWt	100	72	50
Design Overpower, % of 2452 MWt	112	90.2	62.5
System Pressure, psia	2200	2200	2200
DNBR			
Design Power	2.36	-	-
Design Overpower	1.95	1.85	2.49
Minimum DNBR	1.30	1.30	1.30
Critical Heat Flux Correlation	BAW-2	BAW-2	BAW-2
Coolant Flow			
Total Flow Rate (10^6 lb/hr)	131.2	98.2	65.0
Average Velocity Along Fuel Rods (fps)	15.5	11.6	7.7
Coolant Temperature, °F			
Vessel Inlet	555.2	555.2	555
Vessel Outlet	602.8	602.8	603
Heat Transfer			
Total Heat Transfer Surface (ft^2)	49,130	49,130	49,130
Average Heat Flux (BTU/hr-ft^2)	166,000	124,500	83,000
Maximum Heat Flux (BTU/hr-ft^2)	443,000	332,400	221,600
Average Linear Heat Rate (kw/ft)	5.47	4.10	2.73

Babcock & Wilcox **Nuclear Power Generation Division**

Enclosure (5)

To	J.D. Agar, Project Engineering F.J. Levandoski, NSS Product Line		BWNP-20553(7-81)
From	D.L. Smith, Operational Analysis *DL Smith*	Customer or File	620-0012,13 T3.38, T3.81
Subject	Hot Shutdown - Reactivity Effects with all Rods Inserted	Date	April 12, 1982

RCS boration via the Emergency Boration System (EBS) will not be necessary for the Midland plants for the achievement and maintenance of hot shutdown provided all the control rods are inserted. This is valid for both the first and second fuel cycles (where later cycles will be analyzed as the necessary nuclear input becomes available). The attached figures provide the necessary justification of the above statement.

1. Figure 1 shows that the Hot Full Power (HFP) critical boron concentration is in excess of that required for hot shutdown (with no Xenon, 1% subcriticality, all rods inserted) for the first fuel cycle. Thus, no RCS boration will be necessary for the achievement and maintenance of hot shutdown for this first cycle (provided no major injection of de-borated water occurs after trip).

2. Similarly, Figure 2 shows that no RCS boration will be necessary for the second fuel cycle.

Reference:

B&W Document 86-1128114-00 "RCS Boration with all Rods Inserted," N. Smith.

QA Statement

Information transmitted in this memo is applicable to the first and second fuel cycle only for Midland 1 and 2 plants. This information is approved.

Name *J.R. Burris*

Date *4/12/82*

DLS/vab

cc: J.D. Carlton
J.R. Burris
R.B. Lange
L.J. Rudy
R.G. Griese
D.B. Fairbrother
S.G. Toney
N. Smith

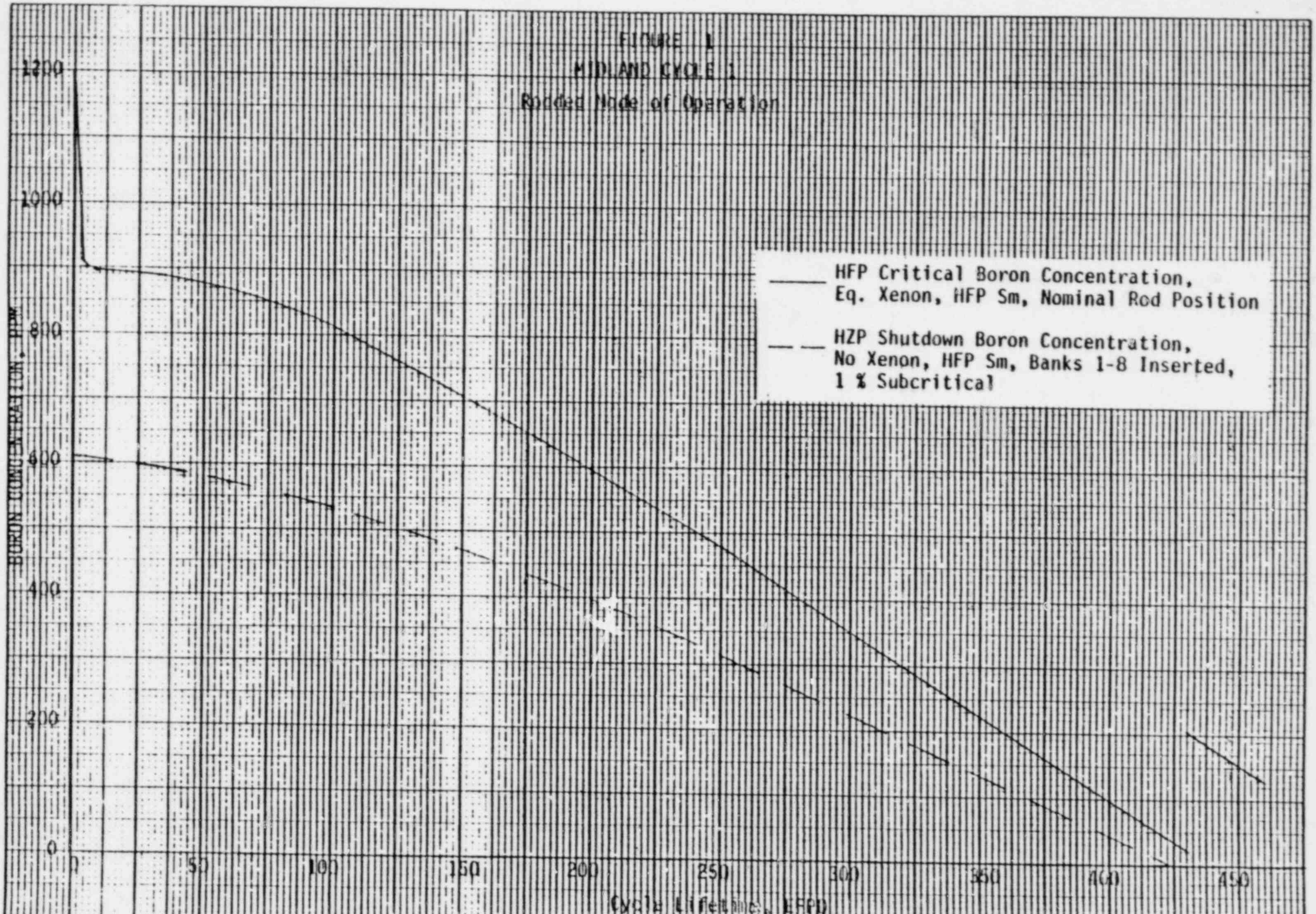
FIGURE 1
MIDLAND CYCLE 1
Rodded Mode of Operation
HFP Critical Boron Concentration, Eq. Xenon, HFP Sm, Nominal Rod Position
HZP Shutdown Boron Concentration, No Xenon, HFP Sm, Banks 1-8 Inserted, 1 % Subcritical
BORON CONCENTRATION, PPM
1200
1000
800
600
400
200
0
0
50
100
150
200
250
300
350
400
450
Cycle Lifetime, EFPD

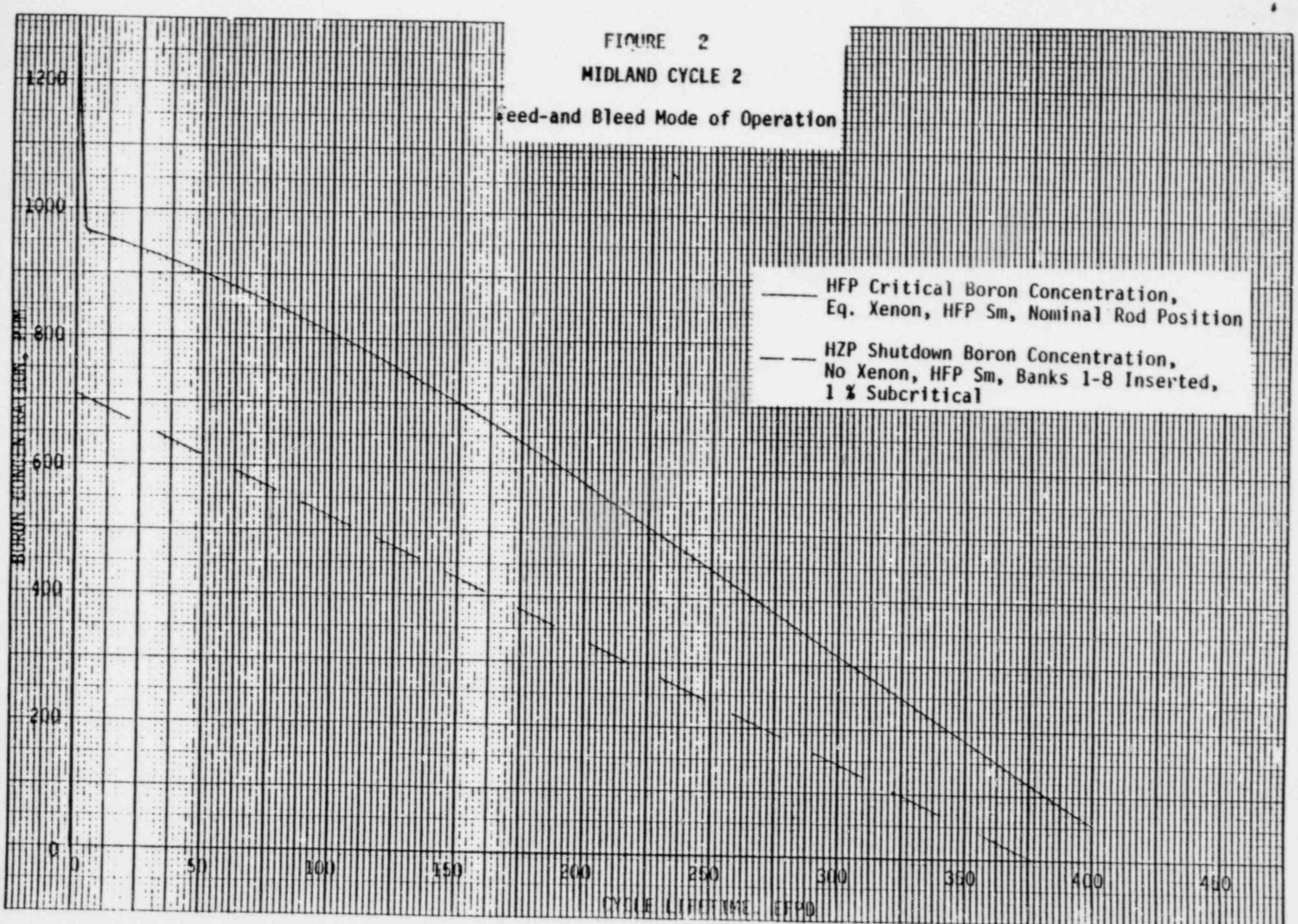
FIGURE 2
MIDLAND CYCLE 2
Feed-and Bleed Mode of Operation
HFP Critical Boron Concentration,
Eq. Xenon, HFP Sm, Nominal Rod Position
HZP Shutdown Boron Concentration,
No Xenon, HFP Sm, Banks 1-8 Inserted,
1 % Subcritical
BORON CONCENTRATION, PPM
1200
1000
800
600
400
200
0
0
50
100
150
200
250
300
350
400
450
CYCLE LIFETIME, EFPD