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July 29, 1983

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Limerick Generating Station, Units 1 and 2 Core Performance Branch and Reactor Systems Branch

Reference: PECO and NRC Telecon Dated 7/15/83

File: GOVT 1-1 (NRC)

Dear Mr. Schwencer:

As discussed in the reference telecon the information contained on the attached draft FSAR page changes will be incorporated into the FSAR, exactly as it appears on the attachments, in the revision scheduled for August, 1983.

Sincerely,

Eugene J. Bradley

Eugene J. Bradley

RJS/gra/53

Copy to: See Attached Service List

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15.4.2.2.2 System Operation

The focal point of this transient is localized to a small portion of the core. Therefore, although reactor controls and instrumentation are assumed to function normally, credit is taken only for the RBM system. A discussion of the transient follows below.

While operating in the power range in a normal mode ~~(except as noted in Section 15.4.2.3.2)~~ of operation, the reactor operator makes a procedural error and withdraws the highest worth control rod until the RBM system inhibits further withdrawal.

Under most normal operating conditions, no operator action is required since the transient which would occur would be very mild. Should the peak linear power design limits be exceeded, the nearest local power range monitor (LPRM) would detect this phenomenon and sound an alarm. The operator must acknowledge this alarm and take appropriate action to rectify the situation.

If the rod withdrawal error is severe enough, the RBM system would sound alarms, at which time the operator would acknowledge the alarm and take corrective action. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will block further withdrawal of the control rod before the fuel reaches the point of boiling transition or the 1%, plastic strain limit imposed on the cladding.

15.4.2.2.3 The Effect of Single Failures and Operator Errors

The effect of operator errors has been discussed above. It was shown that operator errors (which initiated this transient) cannot impact the consequences of this transient due to the highly reliable RBM system. See Section 15.9 for details.

15.4.2.3 <u>Core and System Performance</u>

Insert (A)

~~15.4.2.3.1 Mathematical Model~~

~~For this transient, the reactivity insertion rate is very slow. Therefore, it is adequate to assume that the core has sufficient time to equilibrate (i.e., that both the neutron flux and heat flux are in phase). Making use of the above assumption, this transient is calculated using a steady-state, three-dimensional, coupled nuclear thermal hydraulics computer program. The program is described in detail in Ref 15.4-2. All spatial effects are included in the calculation.~~

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Insert (A)

This event is described in Section S.2.2.1.5 of GESTAR II (Ref. 15.4-3)

For cycle 1, a cycle specific analysis was performed. The consequences of this transient are relatively mild. Neither localized nor gross occurence of boiling transition, nor violation of the 1% plastic strain limit on the cladding, occur. The limiting rod pattern is presented in Figure 15.4-1. A summary of the variation in ΔCPR and MLHGR as a function of withdrawal of the highest worth rod is presented in Table 15.4-2. With a setpoint of 107% the rod is shown to block at 4.5 feet, resulting in a ΔCPR of 0.141 and MLHGR of 16.2 kw/ft.

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The primary output from this code, in addition to the basic nuclear parameters, are: the variation of the linear heat generator rate (LHGR); the variation of the minimum critical power ratio (MCPR), the total reactor power; and the variation of the incore instruments during the transient. A detector response code uses the instrument responses to predict the RBM action under the specified condition for the rod withdrawal error.

The analytical methods and assumptions used in evaluating the consequences of this transient are considered to provide a realistic, yet conservative assessment of the consequences.

15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined to provide a conservative assessment of the consequences.

The conservative assumptions are:

a. The assumed error is a continuous withdrawal of the highest worth rod at its maximum drive speed.

b. The core is assumed to be operating at rated conditions.

c. The reactor is presumed to be in its most reactive state and devoid of all xenon. This ensures that the amount of excess reactivity which must be controlled by the movable control rods is maximum.

d. Furthermore, it is assumed that the operator has fully inserted the highest worth rod prior to its removal, and selected the remaining control rod pattern so that thermal limits are approached in the fuel bundles in the vicinity of the rod to be withdrawn (see Figure 15.4-1). It should be emphasized that this control rod configuration would be highly abnormal, and could only be achieved by deliberate operator action or by numerous operator errors.

e. The operator is assumed to ignore all warnings during the transient.

f. Of the four LPRM strings nearest to the control rod being withdrawn, the two highest reading LPRM during the transient are assumed to have failed.

g. One of the two instrument channels is assumed to be bypassed and out of service. The A and C LPRM chambers input to one channel, while the B and D chambers input

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to the other. The channel with the greatest response is assumed to be bypassed.

The conservative assumptions indicated above provide a high degree of assurance that the transient, as analyzed, bounds all RWEs that could possibly occur. Table 15.4-2 presents the other parameters used in the analysis of this transient.

15.4.2.3.2.1 RBM System Operation

The RBM system minimizes the consequences of an RWE by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. The highest trip level is set so that the safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. Settings are 106%, 98% and 90% of initial, steady-state, operating power at 100% flow. The trip levels are automatically varied with reactor coolant flow to protect against fuel damage at lower flows. The variation is set to assure that no fuel damage will occur at any indicated coolant flow. The operator may encounter any number (up to three) of trip points, depending on the starting power of a given control rod withdrawal. The lower two points may be passed up (reset) by manual operation of a pushbutton. The reset permissive is actuated (and indicated by a light) when the RBM reaches 2% power less than the trip point. The operator should then assess his local power and either reset or select a new rod. The highest (power) trip point may not be reset.

15.4.2.3.3 Results

The consequences of this transient are relatively mild. Neither localized nor gross occurrence of boiling transition, nor violation of the 1% plastic strain limit on the cladding, occur. ~~The variation in the MCPR and MLHGR, as a function of withdrawal of the highest worth rod, is presented in Figures 15.4-2 and 15.4-3 respectively.~~ The bundles presented in Figures 15.4-2 and 15.4-3 represent the envelope of the MCPR and the MLHGR for each two-foot interval during the transient. Variation in the total reactor power is also shown in these figures. Although these figures show the change in thermal limits from the fully inserted to the fully withdrawn position, the control rod is automatically blocked at 5 feet, even under the worst set of assumptions. ~~The variation in the signal response of the two independent channels is shown in Figures 15.4-4 and 15.4-5.~~ With a setpoint of 106% the rod is shown to block at 5 feet, resulting in ΔMCPR of 0.148 and MPLHGR of 14.84 kW/ft.

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15.4.2.3.4 Consideration of Uncertainties

~~The conservative assumptions, which assure that this transient has been conservatively analyzed, have been previously discussed in Section 15.4.2.3.2.~~

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this transient, since this is a localized transient with very little change in the gross core characteristics. Typically, the increase in total core power is less than 5% and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this transient, since no radioactive material is released from the fuel.

15.4.3 CONTROL ROD MALOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

This transient is covered by the evaluations cited in Sections 15.4.1 and 15.4.2.

15.4.4 ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and System Operation

15.4.4.2.1 Sequence of Events

Table 15.4-3 lists the sequence of events for Figure 15.4-6.

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TABLE 15.4-1

SEQUENCE OF EVENTS FOR CONTROL ROD WITHDRAWAL ERROR IN POWER RANGE

TIME-SEC	
~~0~~	~~Core is assumed to be operating at rated conditions~~
0	Operator selects and withdraws the highest worth control rod
~~1.0~~	~~The total core power and the local power in the vicinity of the control rod increase~~
≤ 3.0	The LPRM system indicates excessive localized peaking, but warning is ignored.
~~5.0~~	~~The operator ignores warning and continues withdrawal~~
~15.0	The RBM system indicates excessive localized peaking, but warning is ignored
~~15.0~~	~~The operator ignores warning and continues withdrawal~~
≤30.0	The RBM system initiates a rod block inhibiting further withdrawal
~~40.0~~	~~Reactor core stabilizes at higher core power level~~
~ 60.0	Operator re-inserts control rod to reduce core power level
~ 80.0	Core stabilizes at ~~rated~~ normal conditions

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TABLE 15.4-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR CONTROL ROD WITHDRAWAL TRANSIENT

Reactor power, MWt	3293
Average core exposure, MWd/t	6000
Xenon state	None
Average linear heat generation rate, kW/ft	5.34
Maximum linear heat generation rate, kW/ft	1.34 x 10+1
Location of maximum LHGR bundle	(21-38)
Minimum CPR	1.298
Location of minimum CPR bundle	(21-38)
Highest worth control rod	(26-35)
Rod withdrawal speed, in./sec	3.6
Core coolant flow rate lb m/hr	1.0 x 10+8
Core coolant inlet enthalpy, Btu/lb	5.261 x 10+2
Core average steam volume fraction	3.71 x 10-1
Reactor coolant pressure, average, psia	1.035 x 10+3
Control rod pattern	Figure 15.4-1
RBM trip setpoint, %	106%

REPLACE THIS TABLE WITH FOLLOWING PAGE

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TABLE 15.4-~~1~~ 2

ROD WITHDRAWAL ERROR SUMMARY

Rod Block Set Point (%)	Rod Position (Feet Withdrawn)	ΔCPR	MLHGR (Kw/Ft)
104	3.5	0.101	15.3
105	4.0	0.122	15.9
106	4.0	0.122	15.9
107	4.5	0.141	16.2
108	5.5	0.178	16.3
~~109~~	~~6.0~~	~~0.202~~	~~16.4~~
~~110~~	~~7.0~~	~~0.241~~	~~17.0~~

~~Set Point Selected = 107~~

"NEW TABLE"

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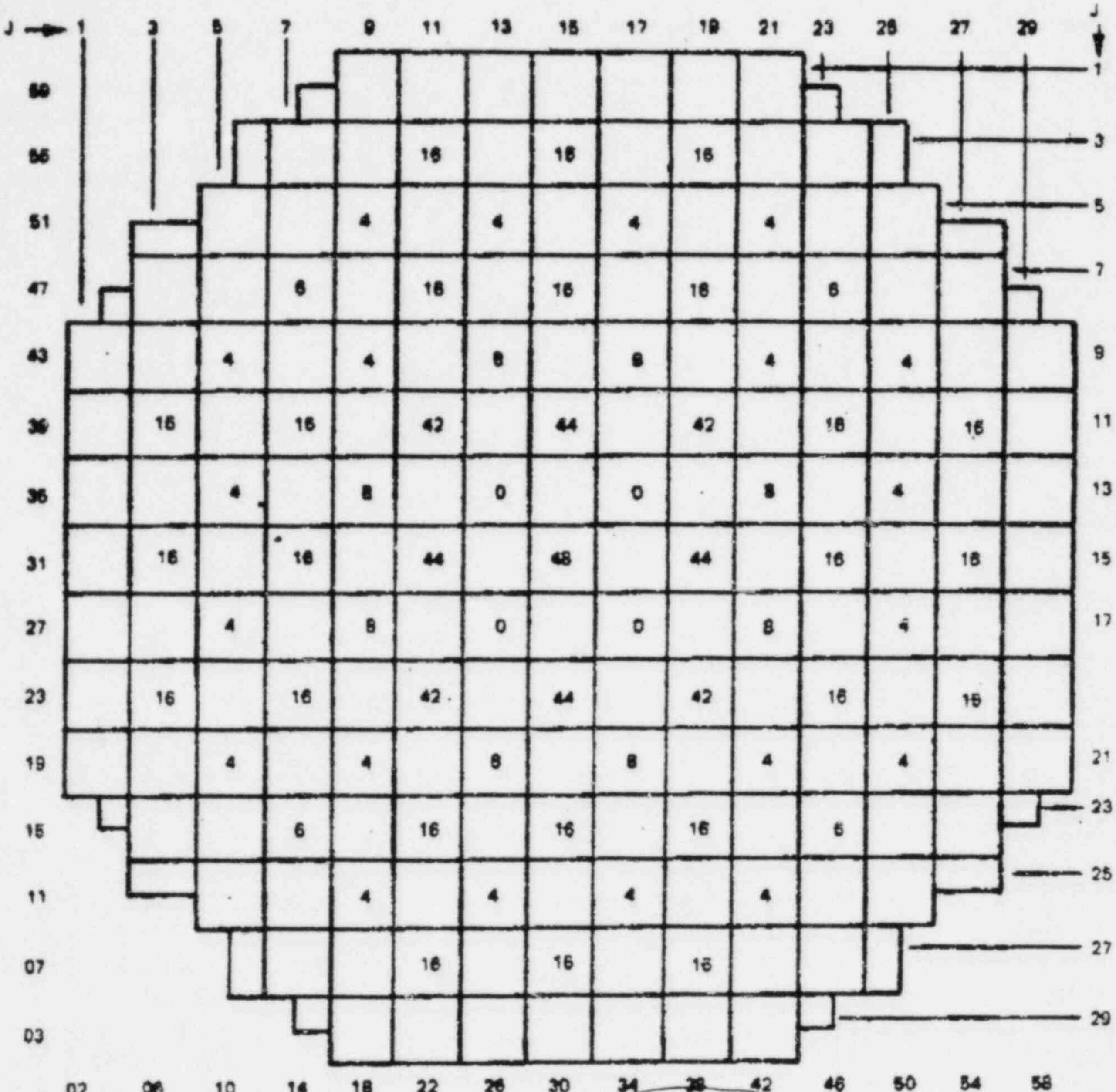
764 ASSEMBLIES
185 CONTROL RODS



*ON LIMITS BUNDLE (21-38)

REPLACE THIS FIGURE WITH FIG. 15.4-1 ATTACHED

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ROD PATTERN FOR ROD WITHDRAWAL ERROR ANALYSIS, ROD (26-35)

FIGURE 15.4-1

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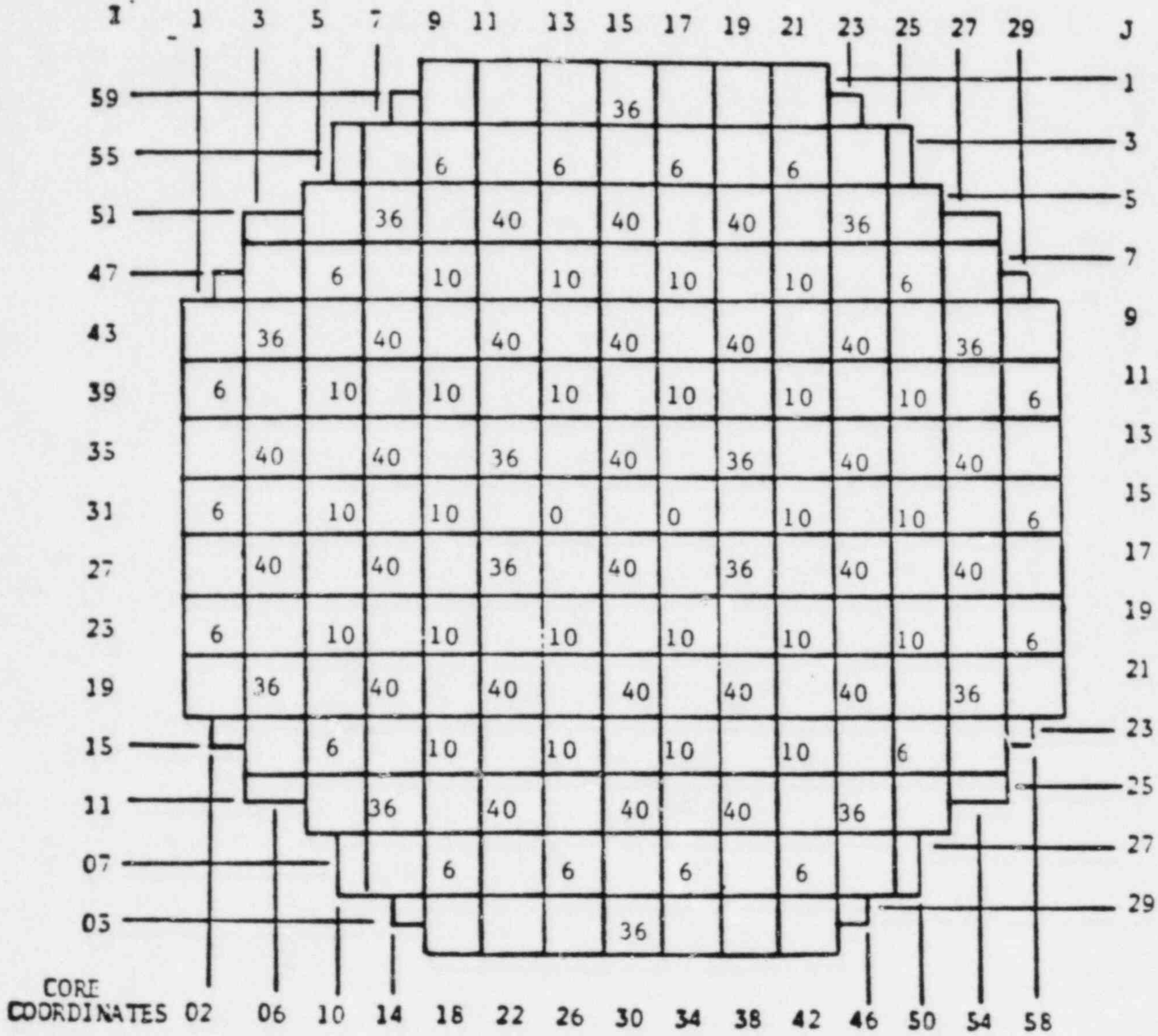
FIGURE 15.4-1

LIMITING CONTROL ROD PATTERN FOR ROD WITHDRAWAL ERROR



"NEW FIGURE"

~~15.4-6~~

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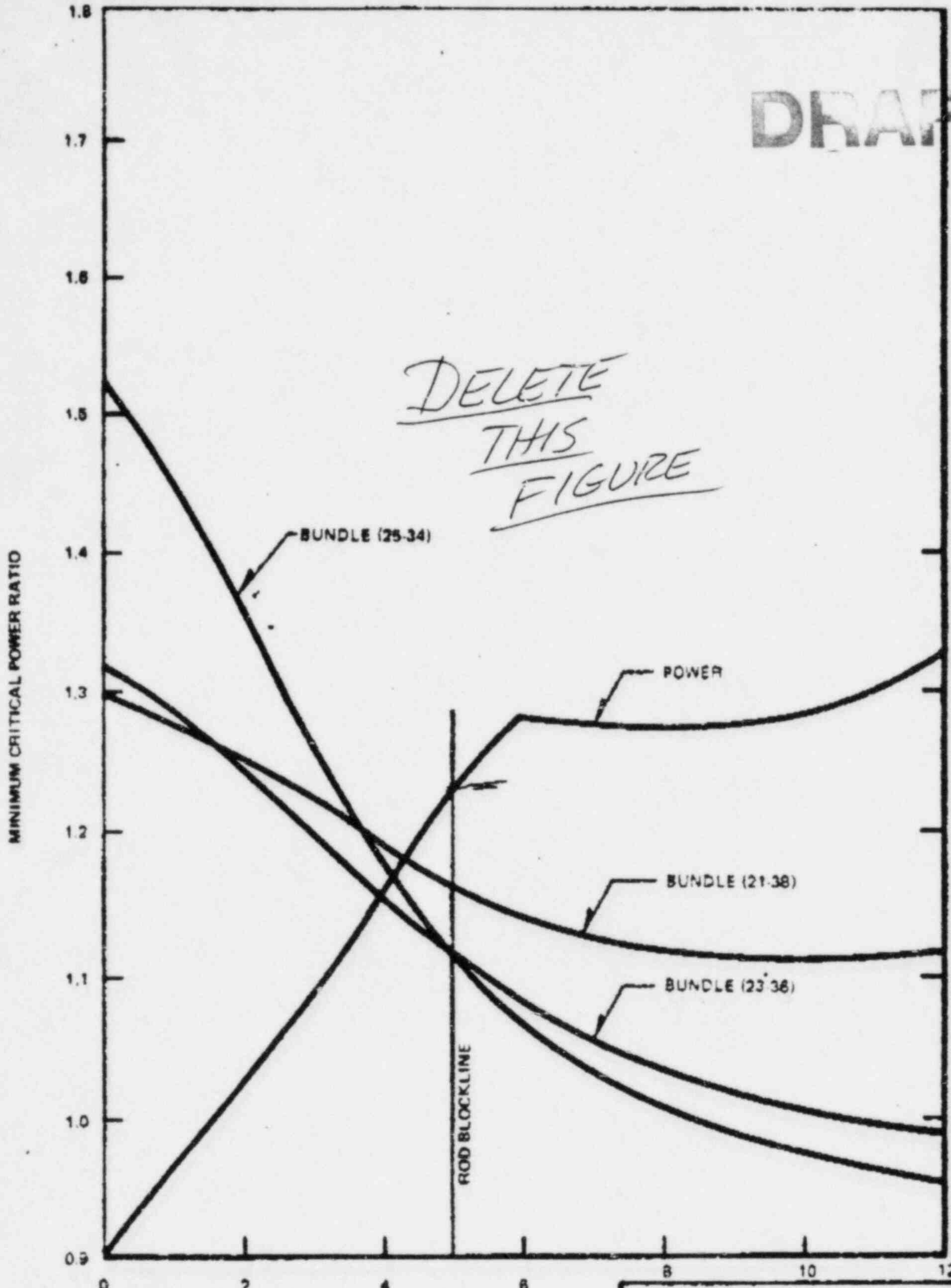

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VARIATION OF FRACTION OF RATED POWER & MCPR WITH THE DISTANCE OF ROD (26-35) WITHDRAWAL DURING CONTINUOUS ROD WITHDRAWAL ERROR

FIGURE 15.4-2

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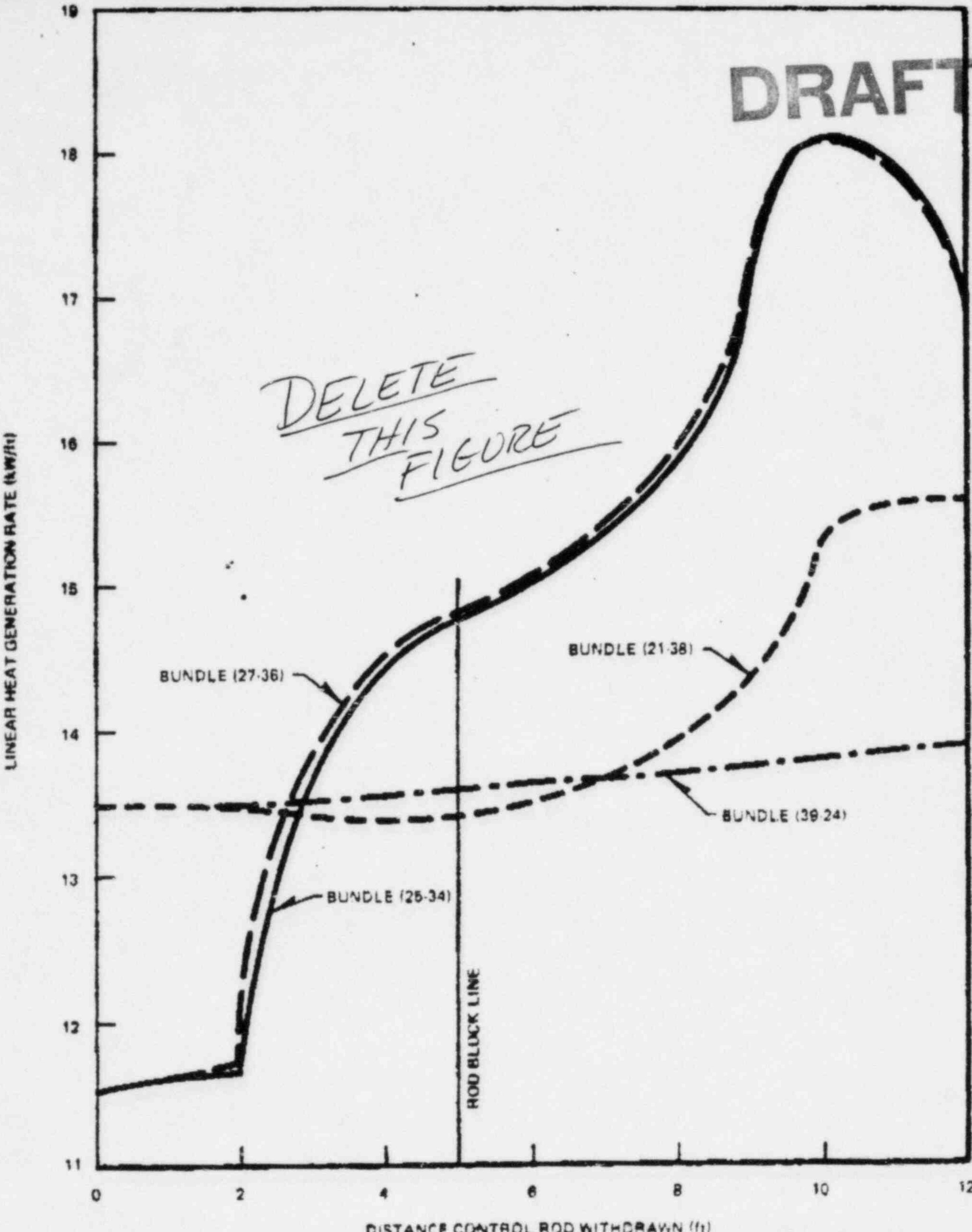

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VARIATION OF MAXIMUM LINEAR HEAT GENERATION RATE WITH THE DISTANCE OF ROD (26-35) WITHDRAWAL DURING CONTINUOUS ROD WITHDRAWAL ERROR

FIGURE 15.4-3

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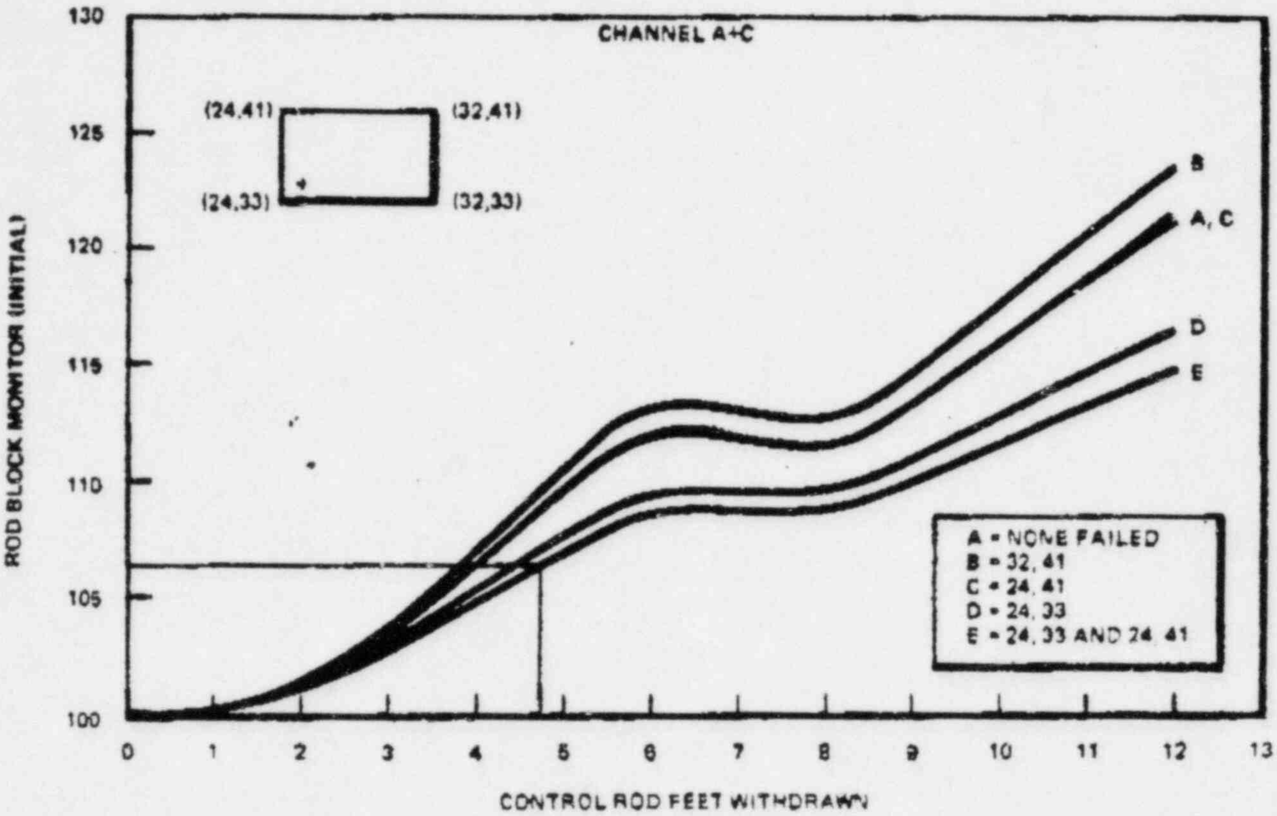

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ROD BLOCK MONITOR RESPONSE TO CONTROL ROD MOTION, CHANNEL A+C

FIGURE 15.4-4

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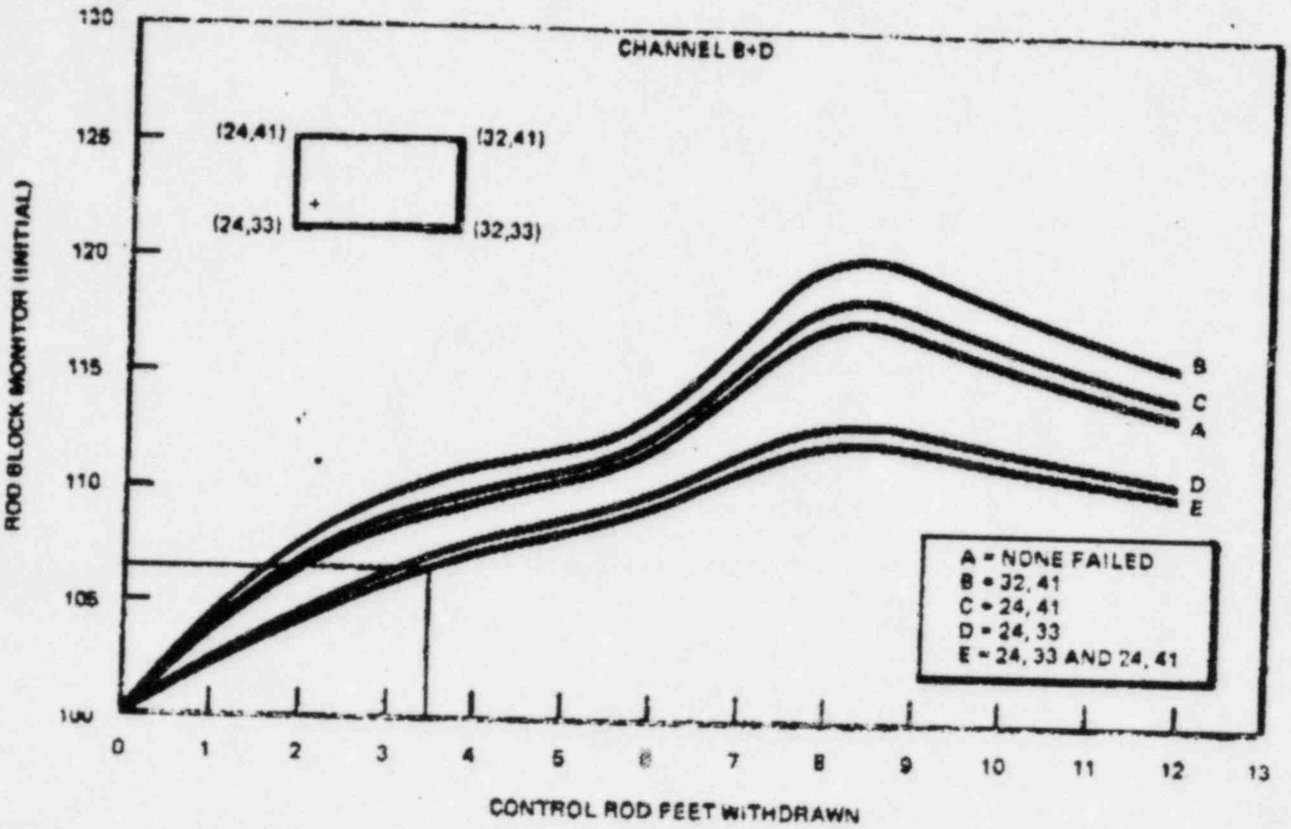

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ROD BLOCK MONITOR RESPONSE TO CONTROL ROD MOTION, CHANNEL B+D

FIGURE 15.4-5

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also put in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following initial core loading.

15.4.7.1.2 Frequency Classification

This accident occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed the bundle is misplaced in the worst possible location, and the plant is operated with the mislocated bundle. This accident is categorized as an infrequent incident based upon the following data.

Expected Frequency: 0.004 events/operating cycle

The above number is based upon past experience. The only misloading accidents that have occurred in the past were in reload cores where only two errors are necessary. Therefore, the frequency of occurrence for initial cores is even lower since three errors must occur concurrently.

15.4.7.2 Sequence of Events and System Operation

The postulated sequence of transients for the misplaced bundle accident (MBA) is presented in Table 15.4-5.

Fuel loading errors, undetected by incore instrumentation following fueling operations, may result in undetected reductions in thermal margins during power operations. No detection is assumed and, therefore, no corrective operator action or automatic protection system functioning occurs.

15.4.7.2.1 The Effect of Single Failures and Operator Errors

This analysis already represents the worst case (i.e., operation of a misplaced bundle with three SAF or SOE) and there are no further operator errors which can make the accident results any worse. It is felt that this section is not applicable to this accident. Refer to Section 15.9 for further details.

15.4.7.3 Core and System Performance

This event is discussed in Section 5.2.5.4 of GESTAR II (Ref. 15.4-3). A summary of the input parameters for this analysis is given in Table 15.4-6 and Figure 15.4-8.

Results of analyzing the worst fuel bundle loading error are reported in Table 15.4-6. As can be seen, MCPR remains well above the point where boiling transition would be expected to occur, and the MLHGR does not exceed the 1% plastic strain limit for the cladding. Therefore, no fuel damage occurs as a result of this accident.

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TABLE 15.4-6

INPUT PARAMETERS AND INITIAL CONDITIONS FOR FUEL BUNDLE LOADING ERROR

1.	Power, % rated	100
2.	Flow, % rated	100
3.	Estimated MCPR operating limit	1.31
4.	MLHGR operating limit, kW/ft	13.4
5.	Average core exposure, MWd/t	0.0
6.	Location of minimum CPR bundle	(25-34)
7.	Location of maximum LHGR bundle	(21-46)
8.	Control rod pattern	Figure 15.4-8

NOTE: Core conditions are assumed to be normal for a hot, operating core at BOC.

REPLACE WITH "NEW TABLE" ATTACHED

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TABLE 15.4-6

INITIAL CONDITIONS AND RESULTS
OF FUEL BUNDLE LOADING ERROR*

Reactor Power, % rated	100
Core Flow, % rated	100
<u>For Largest ΔCPR</u>:	
Core Exposure, MWD/ST	7810
Location of Error	(15,15)
Minimum CPR with Fuel Loading Error	1.29
Minimum CPR Operating Limit	1.24
Minimum CPR Safety Limit	1.06
<u>For Largest ΔMLHGR</u>:	
Core Exposure, MWD/ST	5000
Location of Error	(11,04)
Initial LHGR (Assumed at Operating Limit), KW/FT	13.4
LHGR with Fuel Loading Error, KW/FT	16.97

* Core conditions are assumed to be normal for a hot, operating core.

"NEW TABLE"

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TABLE 15.4-7

MISPLACED BUNDLE ANALYSIS

2.19 w/o enriched bundle at location (25-34), adjacent to highest reading LPRM, replaced with natural uranium, 0.711 w/o enriched bundle from location (23-60) at zero exposure. The four bundles around the LPRM are then assumed to be put on the CPR limit by the operator.

ESTIMATED MCPR(1) LIMIT	MCPR(2) M.P.B.A.	ΔMCPR	ΔMCPR / MCPR LIMIT %
~~1.3100~~	1.1688	-0.1412	-10.78

2.19 w/o enriched bundle at location (21-46), replaced with natural uranium, 0.711 w/o enriched bundle from location (23-60) at zero exposure. The four bundles surrounding the LPRM are then assumed to be put on the MLHGR limit by the operator.

ESTIMATED MLHGR[1] LIMIT	MLHGR[2] M.P.B.A.	ΔMLHGR	ΔMLHGR / MLHGR LIMIT %
13.4	~~14.92~~ 16.97	1.52	11.32

(1) Instrumented location
(2) Non-instrumented location

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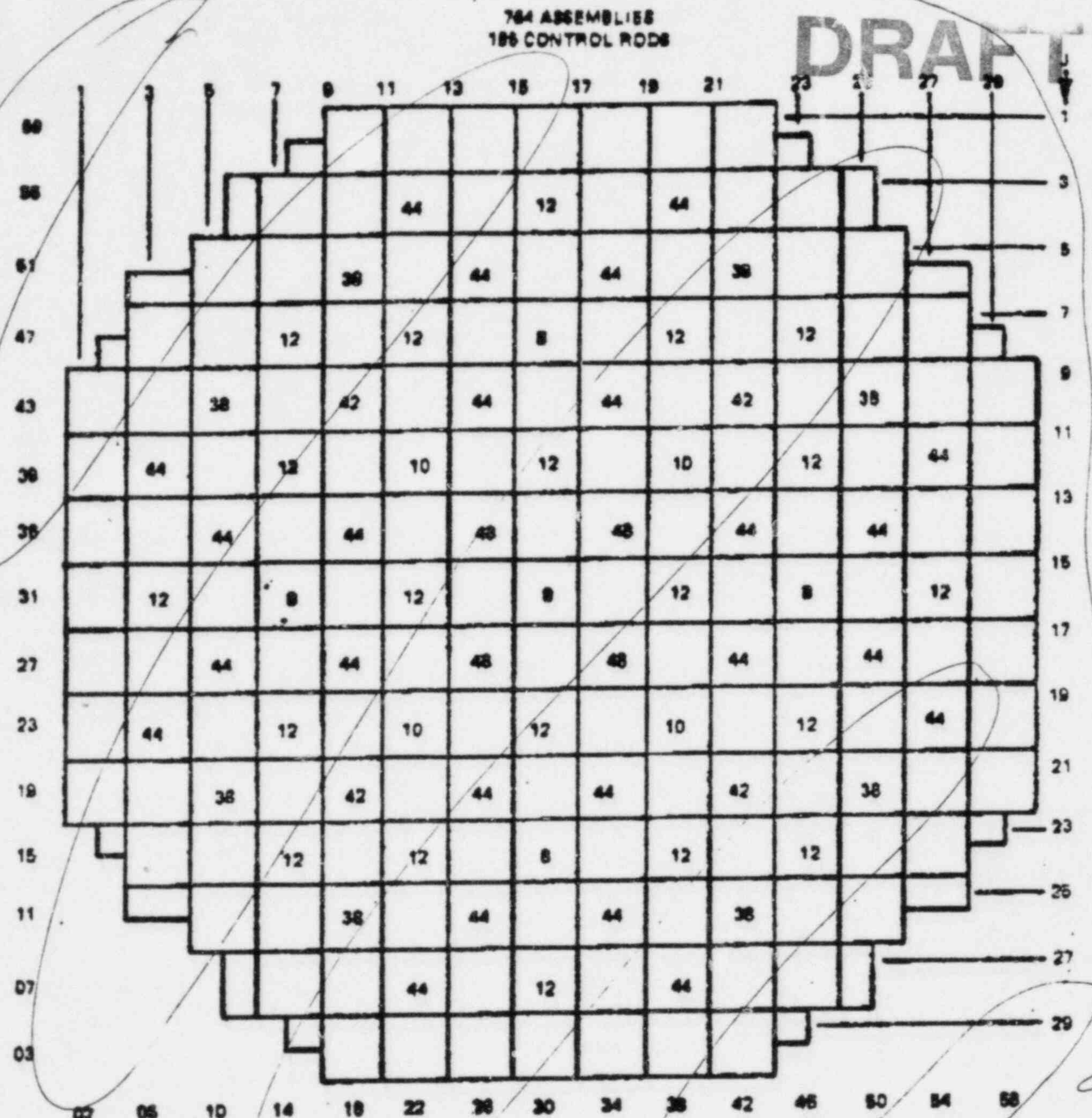

	Bundle ε_∞	Moved From	To
MCPR	0.711	(23-60)	(26-34)
	2.19	(26-34)	(23-60)
MLHGR	0.711	(23-60)	(21-46)
	2.19	(21-46)	(23-60)

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CRITICAL ROD PATTERN AND FUEL BUNDLE EXCHANGE LOCATIONS FOR MISPLACED BUNDLE ACCIDENT (0.0 GWd/t)

FIGURE 15.4-8

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TABLE 15.4-10

INCREMENTAL CONTROL ROD WORTHS USING BPWS[1]

CORE CONDITION	CONTROL ROD GROUP[2]	BANKED AT NOTCH	CONTROL ROD (X, Y)	DROPS FROM-TO	INCREASE IN Keff
BOC-1 Sequence A G1 through G4 W/D all others at 0	7	12	26-35	0->48	.004658
BOC-1 Sequence A G1 through G4 W/D all others at 0	8	12	26-43	0->48	.002518
BOC-1 Sequence A G1 through G4 W/D G5 through G8 at 12 G10 at 0	9	4	30	0->8	.002154
BOC-1 Sequence A G1 through G4 W/D G5 through G8 at 12 G9 at 0	10	4	22-31	0->8	.002141

(1) The following assumptions were made to ensure that the rod worths were conservatively high for the BPWS:

a. BOC
b. Hot startup
c. No xenon

(2) For definition of rod groups, see Figures 4.3-27 and 4.3-28.
The worst case for each rod group is given.

REVISE TO INCORPORATE INFORMATION ON "NEW TABLE" ATTACHED

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TABLE 15.4-~~9~~ 10

INCREMENT WORTH OF THE MOST
REACTIVE ROD USING BPWS

Core Condition	Control Rod Group	Banked At Notch	Control Rod (I,J)	Drops From-To	Increase In K sub (eff)
Sequence B 0 GWD/ST Groups 1-6 Withdrawn	8	12	(26,55)	0 → 48	0.0097

NOTE: The following assumptions were made to ensure that the rod worths were conservatively high for the Banked Position Withdrawal Sequence (BPWS).

a) BOC
b) Hot Startup
c) No Xenon

"NEW" TABLE