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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards addl info re departure from nucleate boiling ratio rod bow penalty,in response to NRC 810831 ltr.Also forwards proposed revised Tech Spec pages which reflect new limits based on revised reactor protection sys string error.

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DUKE POWER COMPANY

POWER BUILDING

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VICE PRESIDENT
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October 16, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. J. F. Stolz, Chief
Operating Reactors Branch No. 4

Re: Oconee Nuclear Station
Docket No. 50-269

Dear Sir:

Mr. A. C. Thies's letter of May 25, 1981 transmitted proposed changes to the Oconee Nuclear Station (ONS) Technical Specifications to support the operation of Oconee Unit 1 at full-rated power during Cycle 7. This letter supplements that response, provides a response to the staff letter of August 31, 1981, and supplements my letter of September 10, 1981 by providing figures and text revised to reflect new reactor protection system string errors for Unit 1.

Attachment 1 provides Duke Power Company's response to the staff letter dated August 31, 1981 concerning rod bow penalty.

Attachment 2 provides the revised Technical Specification pages which reflect new limits based on revised RPS string errors and the minor changes to the original submittal dated May 29, 1981. Please note that page 3.8-2 of the original submittal was included in error and the enclosed page 3.8-3 should replace page 3.8-3 of the May 29, 1981 submittal.

Inasmuch as the applicable licensing fees were provided with the May 29, 1981 submittal, no additional fees are enclosed.

Very truly yours,

William O. Parker, Jr.

JLJ/php

A001 s/11

ATTACHMENT 1

Response to NRC Request for Additional Information

DNBR Rod Bow Penalty

492.1 The DNBR rod bow penalty for the Cycle 7 core was calculated based on the design parameters of Batch 9 only, which contains the limiting (maximum radial peaking factor) fuel assembly. The assumed burnup was the maximum value 17,649 $\frac{MWD}{MTU}$ for any assembly in that batch. You are requested to provide a justification that the calculated rod bow penalty for a Batch 9 assembly is the worst case. Consider the DNBR values with the rod bow penalty for the most limiting assembly for each batch in the Cycle 7 core in conjunction with maximum burnup in the respective batch. Your justification should indicate the results of DNBR calculations which indicate batch number, maximum burnup and rod bow penalty, radial peaking factor and estimated DNBR, and DNBR less rod bow penalty.

492.1 Response

Introduction

A series of thermal hydraulic analyses have been completed, and results are presented herein. All methods are consistent with Duke's approved Oconee Nuclear Station Reload Design Methodology Technical Report.[1] This analysis has been done generically to support both Oconee 1 Cycle 7 and future fuel cycle designs and reload report submittals by Duke.

Approach

1. Review Oconee Unit 1, Cycle 7 final fuel cycle design report for the maximum pin power in each batch and the maximum fuel assembly burnup of each batch, per the interim rod bow penalty methodology.[2] Results are reported in Table 1.

2. Perform an Oconee generic thermal-hydraulic analysis to assess the sensitivity of hot channel minimum DNBR to changes in the maximum pin peak (and associated core power distribution). Sensitivities were determined at the reference design conditions, at the lowest pressure condition permitted by the RPS, and at the highest temperature condition permitted by the RPS to ensure that the most limiting case was enveloped. Results (% pin peak reduction versus % DNBR) are reported in Table 2.

3. Calculate the rod bow penalty at 55,000 MWD/MTU, using the interim rod bow procedure, and extend the historic rod bow penalty versus burnup curve to include the batch 4E assembly of Oconee 1 Cycle 7 (Figure 1).

4. Combine the sensitivity (DNBR credit) of step 2 with a conservative rod bow penalty sensitivity of step 3 to form a conservative trade-off of peaking reduction with burnup. Using this trade-off, develop a conservative, generic procedure that can be used for future fuel cylces.

5. Demonstrate, specifically, the acceptability of Oconee Unit 1 Cycle 7.

Results

1. Pin Peaking Trade-off with Batch Burnup

Table 2 presents the pin peaking reduction required to offset a DNBR change. The maximum percent reduction in pin peak per percent change in hot channel minimum DNBR was observed to be -.61. This DNBR credit for peaking reduction was combined with the maximum rate of change of rod bow penalty with respect to batch maximum fuel assembly burnup change (Figure 1) to obtain a pin peaking reduction trade-off against batch maximum fuel assembly burnup, as follows:

$$\frac{\%\ PP}{\Delta BU} = -\frac{.61\ \%\ PP}{\%\ DNBR} \times \frac{.303\ \%\ DNBR}{1000\ MWD/MTU\ change}$$

$$= -\frac{.185\ \%\ \text{pin peaking reduction}}{1000\ MWD/MTU\ \text{change}}$$

where % PP = percent peaking reduction, and

ΔBU = fuel assembly burnup change.

This result has been increased by 25 percent to maintain future margin. Therefore, the final trade-off is -0.23% pin peak/ 1000 MWD/MTU burnup change.

2. Generic Application to Oconee Nuclear Station Reload Cycles (Procedure)

Use the existing interim rod bow procedure as has been done in the past, but, in addition, do the following:

- Determine the maximum fuel assembly burnup of each batch and the maximum pin peak of each batch over the cycle.
- Determine whether the differences in the maximum fuel assembly burnups of batches N+1, N+2, . . . compared to the maximum fuel assembly burnup of batch N is adequately compensated by its lower pin power (using -0.23% pin peaking per 1000 MWD/MTU).
- If compensated, the process is complete.
- If not adequately compensated, use the higher maximum fuel assembly burnup of batch N+1 or N+2 or . . . for the rod bow penalty.

3. Oconee 1 Cycle 7 Assessment

Table 1 confirms that the net rod bow penalty for batch 9 fuel (0.2%) is most limiting. Thus the Reload Report and Technical Specifications are acceptable as written. (A minor error on page 6-1 of the Reload Report is noted however: batch 9 maximum fuel assembly burnup is 17669, not 17649. The 0.2 percent net rod bow penalty does envelope 17669, as seen from Figure 1.)

References

1. Oconee Nuclear Station Reload Design Methodology Technical Report, Duke Power Company, NFS-1001, Revision 4, June 1981.

2. Letter, L. S. Rubenstein to J. H. Taylor, "Evolution of Interim Procedure Calculating DNBR Reductions Due to Rod Bow", US NRC, October 18, 1979.

TABLE 1

Oconee 1 Cycle 7

Batch	Burnup	Max Pin Peak	Δ BU	Δ PP	Criterion	Conclusion
9	17669	1.428				
8	31339	1.348	13,670	-5.67%	$\leq$-3.1%	Conservative as is
7B	34709	1.315	17,040	-7.9%	$\leq$-3.9%	Conservative as is
4E	50871	0.896	33,202	-37%	$\leq$-7.6%	Conservative as is

Where "Δ BU" is is the change in batch maximum fuel assembly burnup relative to batch 9.

"Δ PP" is the reduction in batch maximum pin peak relative to batch 9, and

"Criterion" is the minimum permissible reduction in pin peak for the batch (using -.23% pin peaking reduction per 1000 MWD/MTU).

The conclusion, "conservative as is", states that the net minimum DNBR determined for batch 9 is more limiting (lower) than the net minimum DNBR for batches 8, 7B, and 4E.

TABLE 2

Oconee Generic
Pin Peaking Reduction Required to Offset a DNBR Change

Reference Design Condition

Case	Pin $Peak_0$	Pin $Peak_1$	$DNBR_0$	$DNBR_1$	% PP/% DNBR**
1	1.714*	1.628	2.049	2.269	-.47
2	1.628	1.543	2.269	2.504	-.51
3	1.543	1.43*	2.504	2.842	-.54
4	1.43	1.35*	2.842	3.104	-.61
5	1.35	0.896*	3.104	5.286	-.48

Low Pressure Condition (1800 psia)

Case	Pin $Peak_0$	Pin $Peak_1$	$DNBR_0$	$DNBR_1$	% PP/% DNBR
1	1.714	1.628	1.297	1.578	-.23
2	1.628	1.543	1.578	1.853	-.30
3	1.543	1.430	1.853	2.216	-.37
4	1.430	1.350	2.216	2.486	-.46
5	1.350	0.896	2.486	4.717	-.37

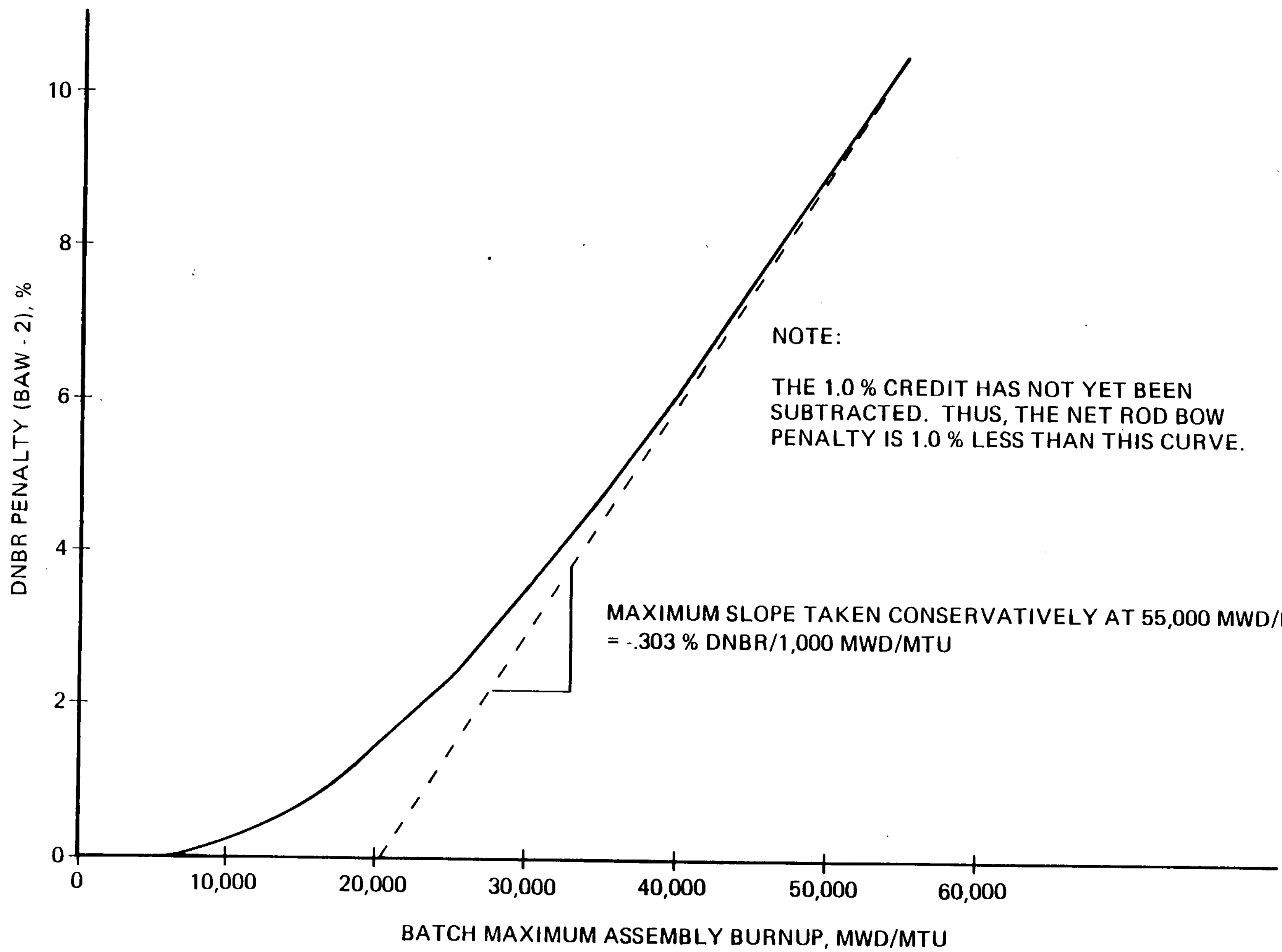
High Temperature Condition (618F Reactor Outlet)

Case	Pin $Peak_0$	Pin $Peak_1$	$DNBR_0$	$DNBR_1$	% PP/% DNBR
1	1.43	1.35	2.160	2.413	-.48

* 1.714 is the design pin peak. 1.43, 1.35, and .896 are specific Oconee 1 Cycle 7 batch maximum pin peaks.

$$** \ \frac{\%\ PP}{\%\ DNBR} = \frac{\%\ \text{Change in pin peak}}{\%\ \text{Change in DNBR}} = \frac{(\text{pin peak}_1 - \text{pin peak}_0)/\text{pin peak}_0}{(DNBR_1 - DNBR_0)/DNBR_0}$$

FIGURE 1
OCONEE GENERIC
ROD BOW PENALTY VS. BURNUP



DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

Proposed Technical Specification Revision

<u>Pages</u>

2.1-3
2.1-7
2.1-10
2.3-2
2.3-3
2.3-8
3.8-3

The maximum thermal power for three-pump operation is 89.899 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.929 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

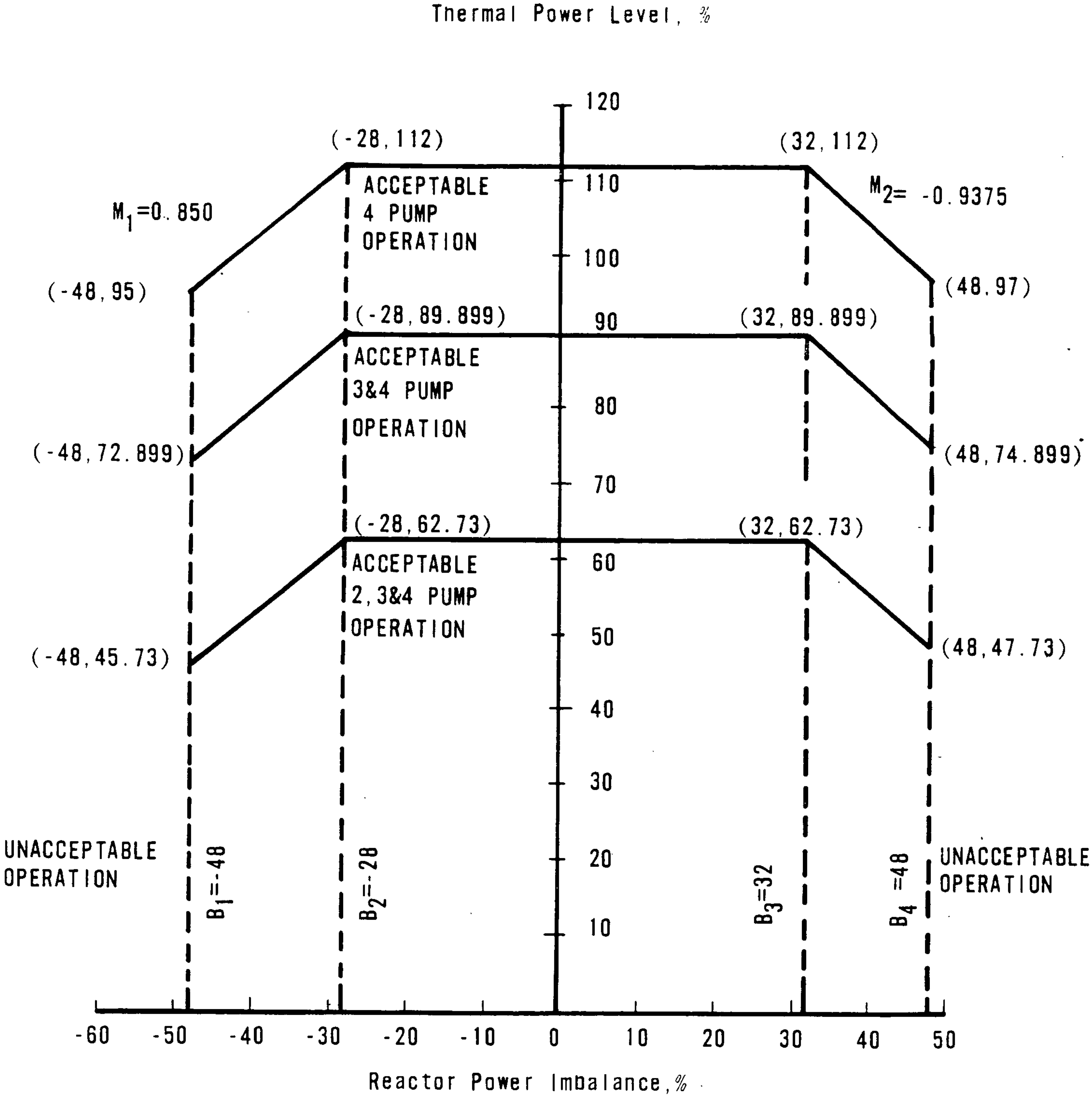
For Figure 2.1-3A, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30.

References

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-1000, March, 1970.

(2) Oconee 1, Cycle 4 - Reload Report - BAW-1447, March, 1977.

(3) Oconee 1, Cycle 7 - Reload Report - BAW-1660, March, 1981.

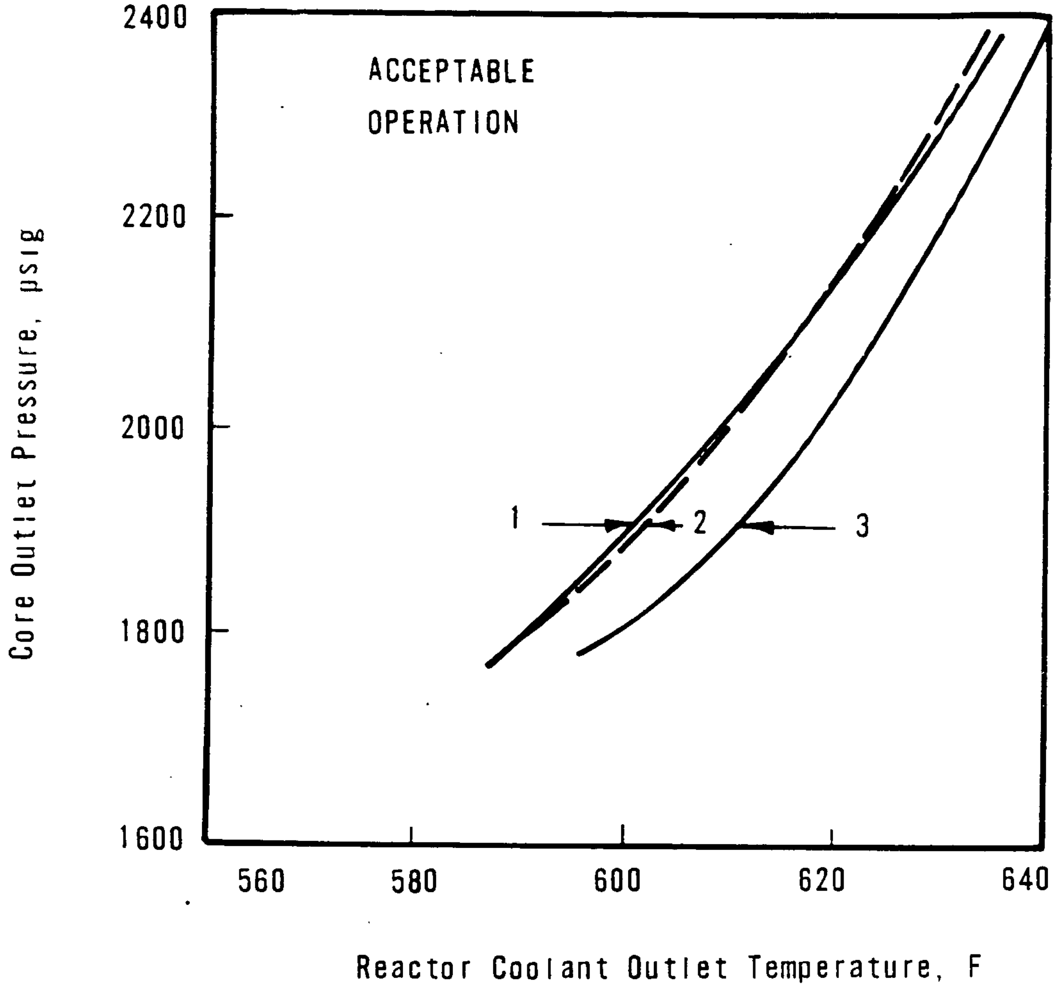


CORE PROTECTION
SAFETY LIMITS
UNIT 1

DUKE POWER

OCONEE NUCLEAR STATION

Figure 2.1-2A



CURVE	COOLANT FLOW, GPM	POWER, %	PUMPS OPERATING	TYPE OF LIMIT
1	374,880 (100%)*	112	4	DNBR
2	280,035 (74.7%)	89.899	3	DNBR
3	183,690 (49.0%)	62.73	2	QUALITY

*106.5% OF FIRST CORE DESIGN FLOW

CORE PROTECTION
SAFETY LIMITS
UNIT 1

OCONEE NUCLEAR STATION
Figure 2.1-3A

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

Overpower Trip Based on Flow and Imbalance

The power level trip setpoint produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accomodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissable power level, and for every power level there is a minimum permissable low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.92% and reactor flow rate is 74.7% or flow rate is 70.09% and power level is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.43% and reactor flow rate is 49.0% or flow rate is 45.79% and the power level is 49%.

   The flux-to-flow ratios account for the maximum calibration and instrument errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.07% - Unit 1 for 1% flow reduction.
1.08% - Unit 2
1.08% - Unit 3

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}$-4706) trip
(1800) psig ($11.14\ T_{out}$-4706)
(1800) psig ($11.14\ T_{out}$-4706)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2, 3)

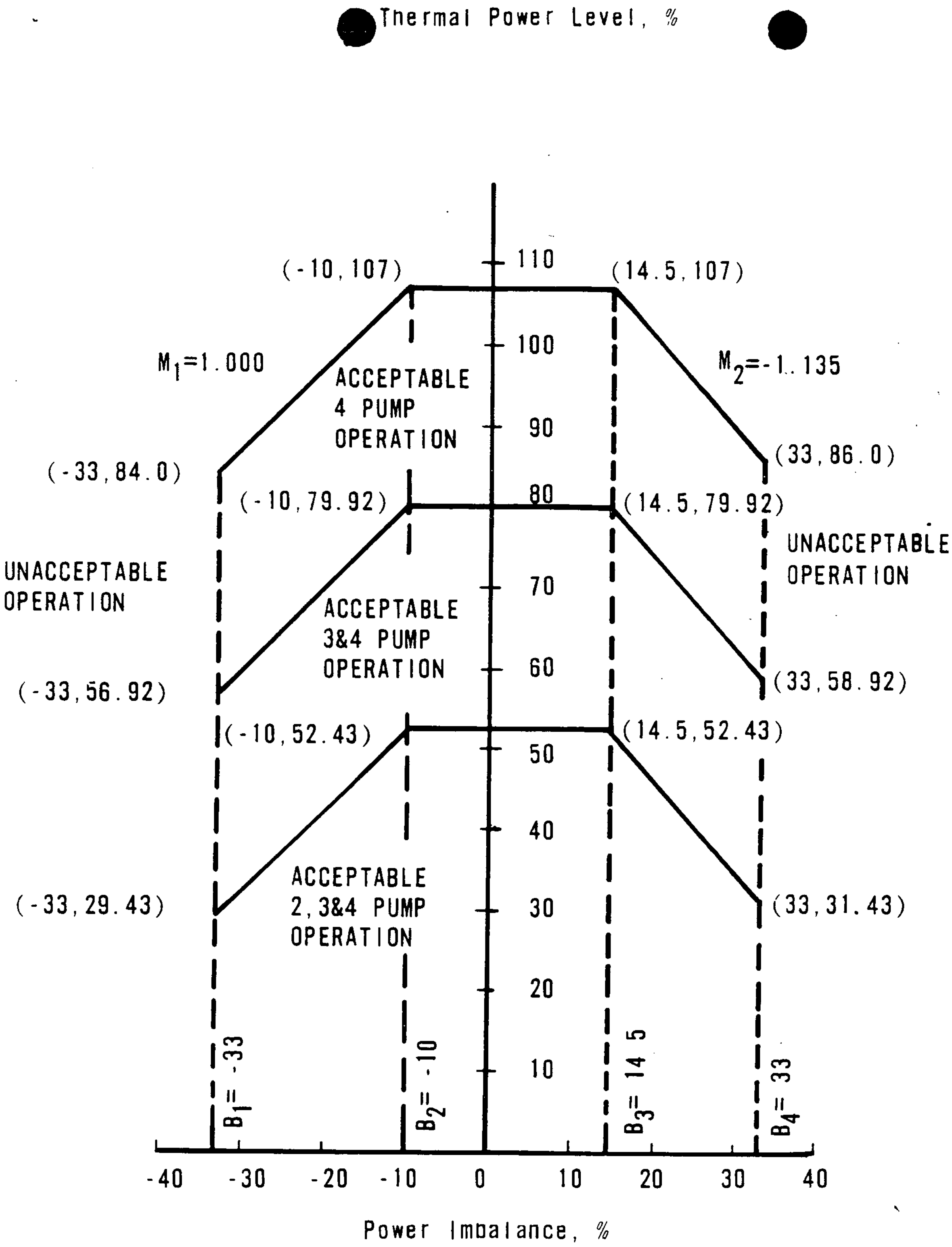
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618^{o}F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperature in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620^{o}F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.3-2A

Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above 1835 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should afuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.(3)

The off-site doses for the fuel handling accident are within the guidelines of 10CFR100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.8.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

REFERENCES

(1) FSAR, Section 9.7
(2) FSAR, Section 14.2.2.1
(3) FSAR, Section 14.2.2.1.2