Docket Nos. 50-269
50-270
50-287

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Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
422 South Church Street
Post Office Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Our review of recent submittals regarding the reevaluation of emergency core cooling system (ECCS) performance in accordance with 10 CFR 50.46 indicates that certain additional information was required in order for us to complete our evaluation. This letter is to inform you of one part of the evaluations that have been submitted that has required additional information.

In performing the evaluation of single failures of ECCS equipment required by Appendix K to 10 CFR 50, Section I.D.1., the effects of a single failure or operator error that causes any manually-controlled, electrically-operated valve to move to a position that could adversely affect the ECCS must be considered. Therefore, please review your submittals regarding ECCS; and if this consideration has not been specifically reported in the past, your upcoming submittal must address this consideration. Include any proposed plant modifications and changes to the Technical Specifications that might be required in order to protect against any loss of safety function caused by this type of failure. A copy of Branch Technical Position [illegible] from SRP [illegible], "Failure Modes Consideration Standard Review Plan" is attached to provide you with guidance.

This request for generic information was approved by OMB under a blanket clearance number B-180225 (R0072); this clearance expires July 31, 1977.

Sincerely,
Original signed by:
Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosure:
Branch Technical Position [illegible]

cc w/enclosure: See next page

ECCS
(2)

See Commonwealth Edison Company (50-295) for blanket concurrence from TR

OFFICE ➤	DRL:ORB#1	DRL:ORB#1			
SURNAME ➤	LMcDonough:dc	RAPurple			
DATE ➤	4/16/75	4/ /75			

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

Mr. Troy B. Conner
Conner, Hadlock & Knotts
1747 Pennsylvania Avenue, NW
Washington, D. C. 20006

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

BRANCH TECHNICAL POSITION EICSB 18

APPLICATION OF THE SINGLE FAILURE CRITERION TO MANUALLY-CONTROLLED ELECTRICALLY-OPERATED VALVES

A. BACKGROUND

Where a single failure in an electrical system can result in loss of capability to perform a safety function, the effect on plant safety must be evaluated. This is necessary regardless of whether the loss of safety function is caused by a component failing to perform a requisite mechanical motion, or by a component performing an undesirable mechanical motion.

This position establishes the acceptability of disconnecting power to electrical components of a fluid system as one means of designing against a single failure that might cause an undesirable component action. These provisions are based on the assumption that the component is then equivalent to a similar component that is not designed for electrical operation, e.g., a valve that can be opened or closed only by direct manual operation of the valve. They are also based on the assumption that no single failure can both restore power to the electrical system and cause mechanical motion of the components served by the electrical system. The validity of these assumptions should be verified when applying this position.

B. BRANCH TECHNICAL POSITION

1. Failures in both the "fail to function" sense and the "undesirable function" sense of components in electrical systems of valves and other fluid system components should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.

2. Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should include a list of all electrically-operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

3. Electrically-operated valves that are classified as "active" valves, i.e., are required to open or close in various safety system operational sequences, but are manually-controlled, should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless: (a) electrical power can be restored to the valves from the main control room, (b) valve operation is not necessary for at least ten minutes following occurrence of the event requiring such operation, and (c) it is demonstrated

that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

4. When the single failure criterion is satisfied by removal of electrical power from valves described in (2) and (3), above, these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion.

5. The phrase "electrically-operated valves" includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve whose air supply is controlled by an electrical solenoid valve).

C. REFERENCES

1. Memorandum to R. C. DeYoung and V. A. Moore from V. Stello, October 1, 1973.

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

MAR 14 1975

Docket Nos. 50-269
50-270
50-287

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
422 South Church Street
Post Office Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Although your emergency core cooling systems generally satisfy our requirements with regard to long-term cooling, the system configurations have not been specifically evaluated to show that significant changes in chemical concentrations would not occur during the long term after a loss-of-coolant accident (LOCA) and these potential changes have not been specifically addressed by appropriate operating procedures. Accordingly, you should review your system capabilities and operating procedures to assure that boron precipitation would not compromise long-term core cooling capability following a LOCA. This review should consider all aspects of your design, including component qualification in the LOCA environment in addition to a detailed review of operating procedures. You should examine the vulnerability of your design to single failures that would result in any significant boron precipitation.

You should submit this evaluation and associated operating procedures within 30 days of receipt of this letter. These procedures should be promptly effected to assure that boron precipitation would not interfere with the ability of your facility to conform to Criterion (5) of 10 CFR 50.46(b). We will inform you as to the acceptability of your evaluation and associated operating procedures.

While solute concentrations may be subject to control through operating procedures, equipment modifications may be required or desirable to simplify such procedures. Your submittal should include a plan for completing such modifications within six months of the date of this letter.

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

ECCS
2
R.

MAR 14 1975

This request for generic information was approved by GAO under a blanket clearance number B-180225 (R0072); this clearance expires July 31, 1977.

Sincerely,

Original signed by
R. A. Purple
Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

cc: See next page

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OFFICE ➤	RL:ORB#1	RL:ORB#1				
SURNAME ➤	LMcDonough:dc	RAPurple				
DATE ➤	3/13/75	3/ /75				

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

Mr. Troy B. Conner
Conner, Hadlock & Knotts
1747 Pennsylvania Avenue, NW
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201 South Spring Street
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Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
422 South Church Street
Post Office Box 2178
Charlotte, North Carolina 28201

Gentlemen:

This is in response to your letter of January 28, 1975, relating to the re-evaluation of the Oconee ECCS performance.

We request that the information relating to containment back pressure calculations identified in the Safety Evaluation dated December 27, 1974, be included in your submittal of the re-evaluation of the Oconee ECCS performance.

Sincerely,

Original signed by R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Readtor Licensing

cc: See next page

ECCS
(2)

OFFICE	RL:OR-1	RL:OR-1				
SURNAME	LMcDonough:dc	RAPurple				
DATE	2/7/75	2/ /75				

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

Mr. Troy B. Conner
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Docket

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

DEC 27 1974

Docket Nos. 50-269, 50-270 and 50-287

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
422 South Church Street
Post Office Box 2178
Charlotte, North Carolina 28201

Gentlemen:

The Commission has issued the enclosed Order for Modification of License for the Oconee Nuclear Power Station Units 1, 2, and 3 pertaining to your proposed Technical Specifications which were submitted pursuant to Section 50.46 and Appendix K of 10 CFR Part 50. The enclosed Safety Evaluation Report contains the bases for our action.

A copy of the Order is being filed with the Office of the Federal Register for publication. This Order is effective immediaately.

Within 30 days after receipt of this letter, we request that you inform us of your submittal date for the reevaluation required by paragraph 1 of Section III of the Order.

You will note that the order requires that a reanalysis in conformance with the order must be submitted along with any request for authorization for any core reloading. In order to provide for sufficient time for our review of your reanalysis, you should assure that such submittal is provided at least 45 days prior to your schedule of initiation of operation following approval of such reloading. You should also note, that since your current analysis was based upon your presently authorized fuel loading patterns, any modification of fuel design or core configuration which affects the basis for the analysis will require staff approval.

Sincerely,

/S/

Robert A. Purple, Chief
Operating Reactors Branch #1
Directorate of Licensing

Enclosures:
1. Order
2. Safety Evaluation Report
3. Status Report by the Directorate of Licensing in the Matter of Babcock and Wilcox Model Conformance to 10 CFR 50, Appendix K

ECCS

cc w/enclosures:
Mr. William L. Porter
Duke Power Company
422 South Church Street
Post Office Box 2178
Charlotte, North Carolina 28201

Mr. Troy B. Conner
Conner, Hadloc, & Knotts
1747 Pennsylvania Avenue, NW.
Suite 1050
Washington, D.C. 20006

Oconee County Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable Reese A. Hubbard
County Supervisor of Oconee County
Walhalla, South Carolina 29621

cc w/enclosures and incoming:
Mr. Randolph Hendricks
State Clearinghouse and
Information Center
116 West Jones Street
Raleigh, North Carolina 27603

Mr. Dave Hopkins
U.S. Environmental Protection Agency
Region IV Office
1421 Peachtree Street, NE.
Atlanta, Georgia 30309

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
DUKE POWER COMPANY	)	Docket Nos. 50-269
	)	50-270
(Oconee Nuclear Power Station,	)	50-287
Units 1, 2, and 3)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

The Duke Power Company (the licensee) is the holder of facility licenses DPR-38, DPR-47 and DPR-55, which authorize operation of the Oconee Nuclear Power Station, Units 1, 2, and 3, respectively, in Oconee County, South Carolina. These licenses provide, among other things, that they are subject to all rules, regulations and orders of the Commission now or hereafter in effect.

II.

Pursuant to the requirements of the Commission's regulations in 10 CFR § 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors", on August 5, 1974, the licensee submitted an evaluation of ECCS cooling performance calculated in accordance with an evaluation model developed by the Babcock and Wilcox Company ("the vendor"), along with certain proposed technical specifications necessary to bring reactor operation into conformity with the results of the evaluation.

The evaluation model developed by the vendor has been analyzed by the regulatory staff for conformity with the requirements of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models". The regulatory staff's evaluation of the vendor's model is described in two previously published documents: Status Report by the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K, issued October 15, 1974, and a Supplement to the Status Report, issued November 13, 1974. Based on its evaluation, the regulatory staff has concluded that the vendor's evaluation model was not in complete conformity with the requirements of Appendix K and that certain modifications described in the above-mentioned documents were required in order to achieve such conformity. The regulatory staff assessments were reviewed by the Commission's Advisory Committee on Reactor Safeguards in meetings held on October 26, 1974, and November 14, 1974.

In its Report to the Chairman of the AEC, dated November 20, 1974, the Advisory Committee has concluded that "the four light-water reactor vendors have developed Evaluation Models which, with additional modifications required by the Regulatory Staff, will conform to Appendix K to Part 50".

Since the licensee's evaluation of ECCS cooling performance is based upon the vendor's evaluation model, the licensee's evaluation is similarly deficient. The regulatory staff has assessed the effect of the changes required in the evaluation model upon the results of the evaluation of ECCS performance for

Oconee facilities submitted on August 5, 1974 and September 20, 1974. This is described in the Safety Evaluation Report of the Oconee Nuclear Station Units 1, 2, and 3, Docket Nos. 50-269, 50-270 and 50-287, dated December 27, 1974. On the basis of its review, the regulatory staff has determined that changes in operating conditions for the plant, in addition to those proposed in the licensee's submittal of September 20, 1974 and August 5, 1974, are necessary to assure that the criteria set forth in § 50.46(b) are satisfied. These additional changes, which are set forth in Appendix A to the Safety Evaluation Report, consist of modifications to the linear heat generation rate. These further restrictions will assure that ECCS cooling performance will conform to all of the criteria contained in 10 CFR § 50.46(b), which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling.

These further restrictions were established on the basis of studies of the effect of model changes on the previously submitted evaluations. The regulatory staff believes that these restrictions should be verified by a re-analysis based upon an approved evaluation model, in conformity with 10 CFR § 50.46 and Appendix K. During the interim, before an evaluation in conformity with the requirements of 10 CFR § 50.46 can be submitted and evaluated, the regulatory staff has concluded that continued conformance to the requirements of the Commission's Interim Acceptance Criteria,* and conformance to the restrictions contained in the licensee's September 20, 1974 and August 5, 1974 submittals, together with the additional limitations set forth in Appendix A of the Staff Safety Evaluation Report, will provide

*Interim Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors, 36 F.R. 12247, June 29, 1971, as amended 36 F.R. 24082, December 18, 1971

reasonable assurance that the public health and safety will not be endangered. These additional restrictions are set forth as Appendix A to this Order.

III.

In view of the foregoing and, in accordance with the provisions of § 50.46(a)(2)(v), the Acting Director of Licensing has found that the evaluation of ECCS cooling performance submitted by the licensee is not consistent with the requirements of 10 CFR § 50.46(a)(1) and that the further restrictions set forth in this Order are required to protect the public health and safety. The Acting Director of Licensing has also found that the public health, safety, and interest require that the following Order be made effective immediately. Pursuant to the Atomic Energy Act of 1954, as amended, the Commission's regulations in 10 CFR §§ 2.204, 50.46, and 50.54.

IT IS ORDERED THAT:

1. As soon as practicable, but in no event later than six months from the date of publication of this order in the FEDERAL REGISTER, or prior to any license amendment authorizing any core reloading, whichever occurs first, the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, § 50.46. Such evaluation may be based upon the vendor's evaluation model as modified in accordance with the changes described in the Staff Safety Evaluation Report of the Oconee Nuclear Power Station, dated December 27, 1974. The evaluation

shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

2. Effective immediately, reactor operation shall continue only within the limits of:

(a) The requirements of the Interim Acceptance Criteria, the Technical Specifications, and license conditions imposed by the Commission in accordance with the requirements of the Interim Acceptance Criteria, and

(b) The limits of the proposed Technical Specifications submitted by the licensee on September 20, 1974 and August 5, 1974, as modified by the further restrictions set forth in Appendix A, attached hereto.

The license shall conform operation to the foregoing limitations until such time as the proposed Technical Specifications required to be submitted in accordance with paragraph 1 above are approved or modified and issued by the Commission. Subsequent notice and opportunity for hearing will be provided in connection with such action.

IV.

Within thirty (30) days from the date of publication of this Order in the FEDERAL REGISTER the licensee may file a request for a hearing with respect to this Order. Within the same thirty (30) day period any other person whose interest may be affected may file a request for a hearing with respect to this Order in accordance with the provisions of 10 CFR § 2.714 of the Commission's Rules of Practice. If a request for a hearing is filed within the time prescribed herein, the Commission will issue a notice of hearing or an appropriate order.

For further details with respect to this action, see (1) the licensee's submittals dated September 20, 1974 and August 5, 1974 and vendor's topical reports referenced in the licensee's submittals, which describe the vendor's evaluation model, (2) the Status Report by the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K, (3) Supplement 1 thereto dated November 13, 1974, (4) the Safety Evaluation Report dated December 27, 1974, and (5) Report of the Advisory Committee on Reactor Safeguards dated November 20, 1974. All of these items are available at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., and at the Oconee County Library, 201 South Spring Street, Walhalla, South Carolina 29691. A single copy each of items (2) through (5) may be obtained upon request addressed to the U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing, Regualtion.

Dated at Bethesda, Maryland this 29 day of December, 1974.

FOR THE ATOMIC ENERGY COMMISSION

/S/

Edson G. Case, Acting Director
Directorate of Licensing

APPENDIX A

OPERATING RESTRICTIONS

The Regulatory staff has reviewed the methods used by Babcock and Wilcox to derive the LOCA-related operating limits for its plants. The review considered the basic calculation method, the range of operating conditions calculated, the types of uncertainties and their magnitude, and the instrumentation provided to monitor plant operation. Based on this review, we conclude that sufficient monitoring instrumentation is present to provide assurance that the plant may be operated within LOCA-related operating restrictions. We further conclude that operation of Oconee Units 2 and 3 within the restrictions shown on Figures A-1 through A-3, which were a part of the August 5, 1974 proposed Technical Specifications from the licensee, will assure that the heat generation limits of Figure A-6 will not be exceeded. For Unit 1, Figure A-4 already incorporates both criteria. For Oconee Unit 1, we further conclude that the heat generation limits of Figure A-6 will not be exceeded if Unit 1 is operated within the Technical Specifications for cycle 2, provided that the following additional operating restrictions pursuant to the authority contained in 10 CFR 50.46 are imposed:

1. The power level cutoff indicated in Figure 3.5-2-1A1 of the licensee's September 20, 1974 submitted shall be reduced from 94 percent of rated power. The power level cutoff is defined as the maximum power at which the reactor can operate without regard to the reactivity held by xenon.

2. Power level shall not be greater than 92 percent (power level cutoff) unless one of the following requirements is met:

   a. Quadrant tilt is less than or equal to 2.5 percent and the xenon reactivity is within 10 percent of the value for operation at steady-state rated power.

   b. Quadrant tilt is greater than 2.5 percent and the xenon reactivity is within 5 percent of the value for operation at steady-state rated power.

3. Operation shall be within the control rod withdrawal limits as shown in Figure A-4.

4. Operation shall be within the power imbalance envelope as shown in Figure A-5.

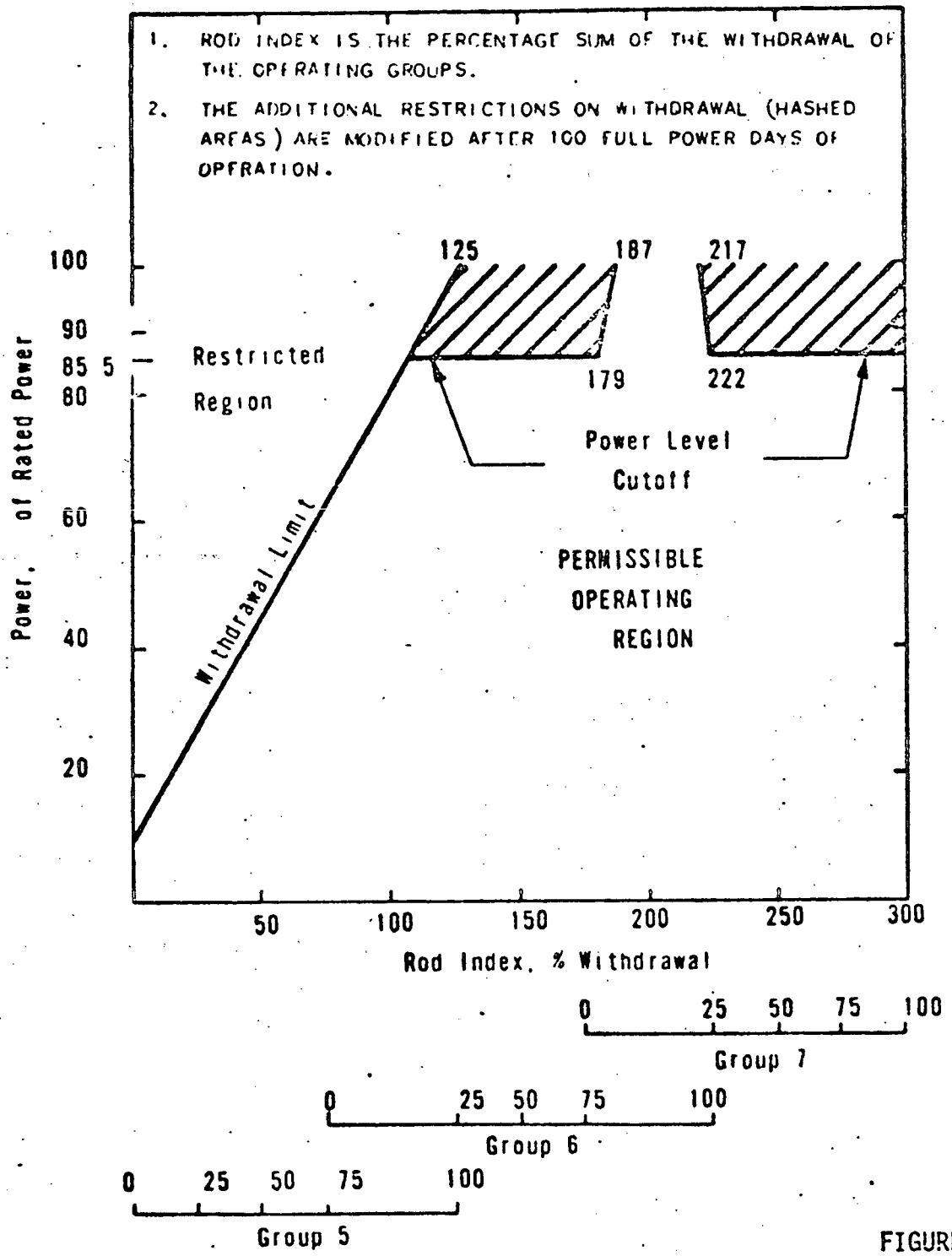

FIGURE A-1

CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION - UNITS 2, 3

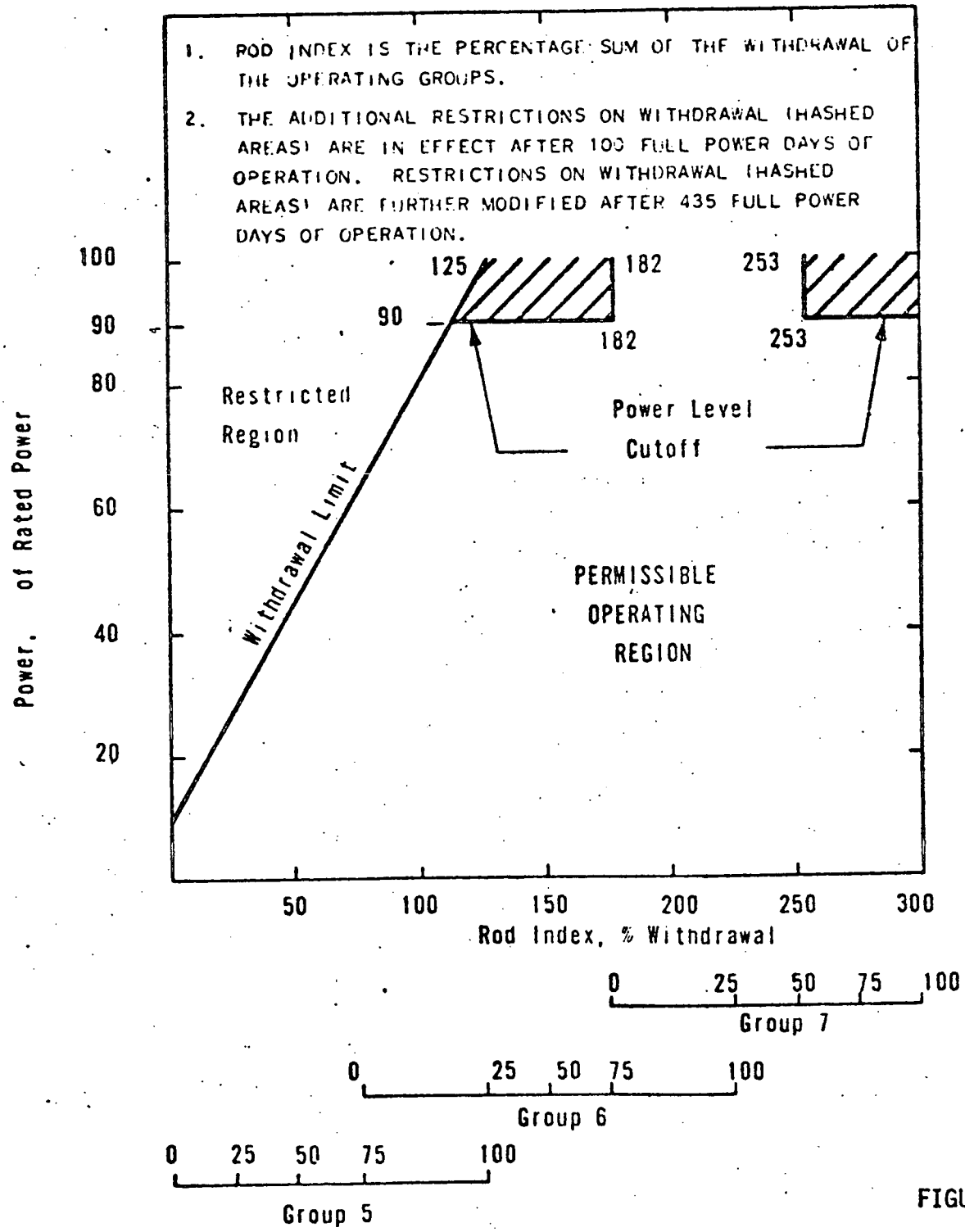

FIGURE A-2

CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION - UNITS 2, 3

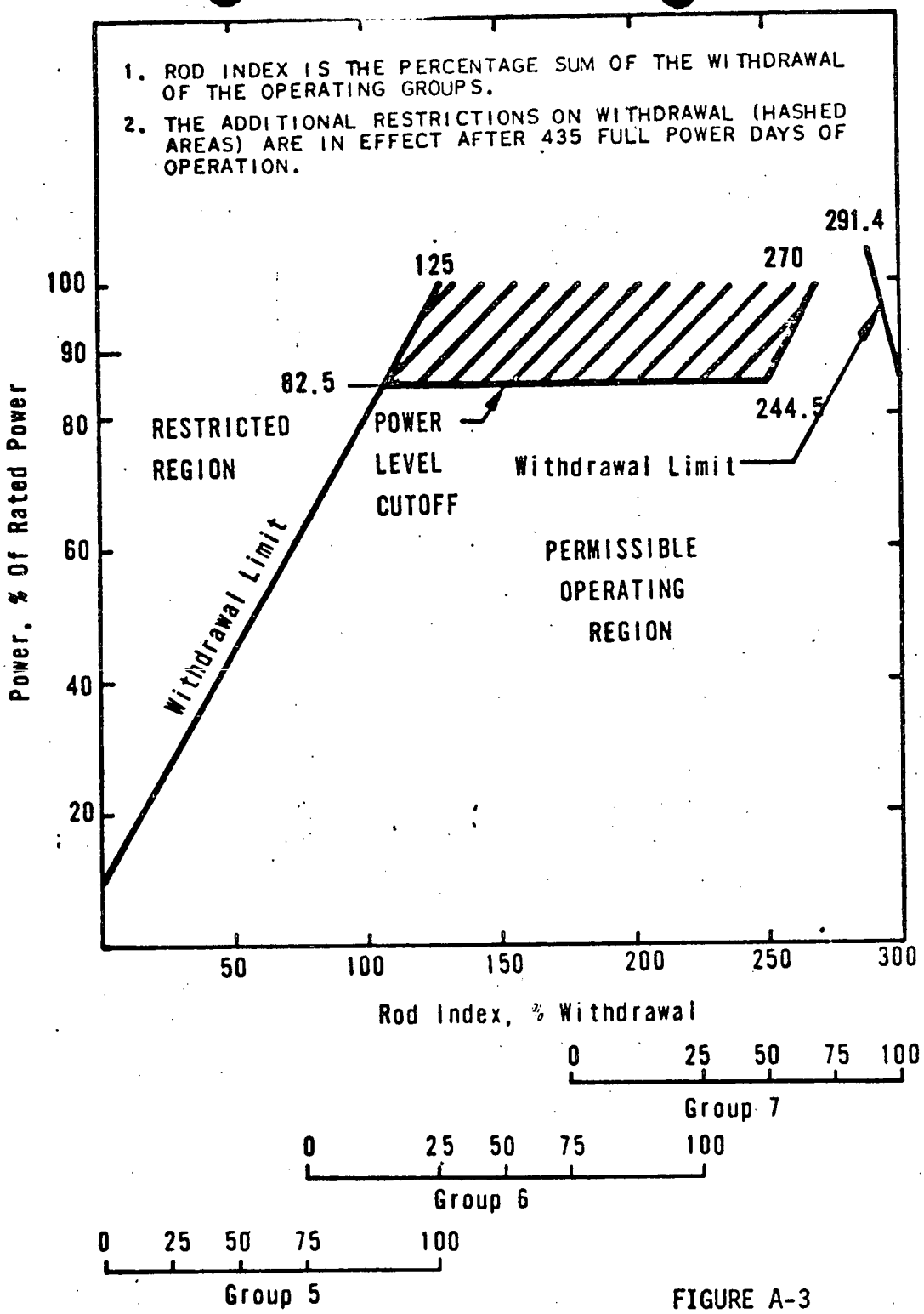

FIGURE A-3

CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION UNITS 2, 3

1. Rod index is the percentage sum of the withdrawal of the operating groups.
2. The withdrawal limits are modified after 250 ± 5 full power days of operation.

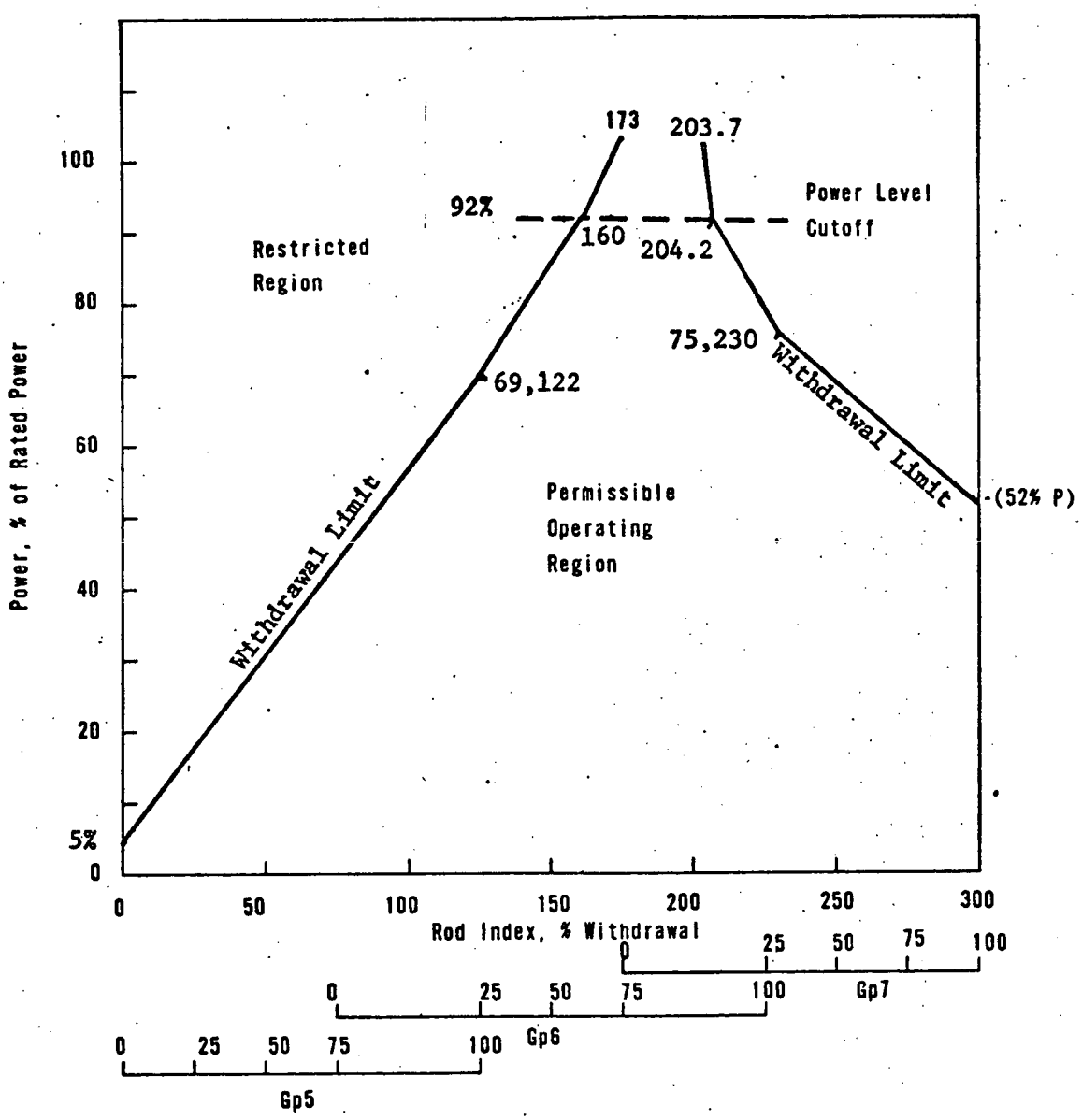

FIGURE A-4

CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION

UNIT 1

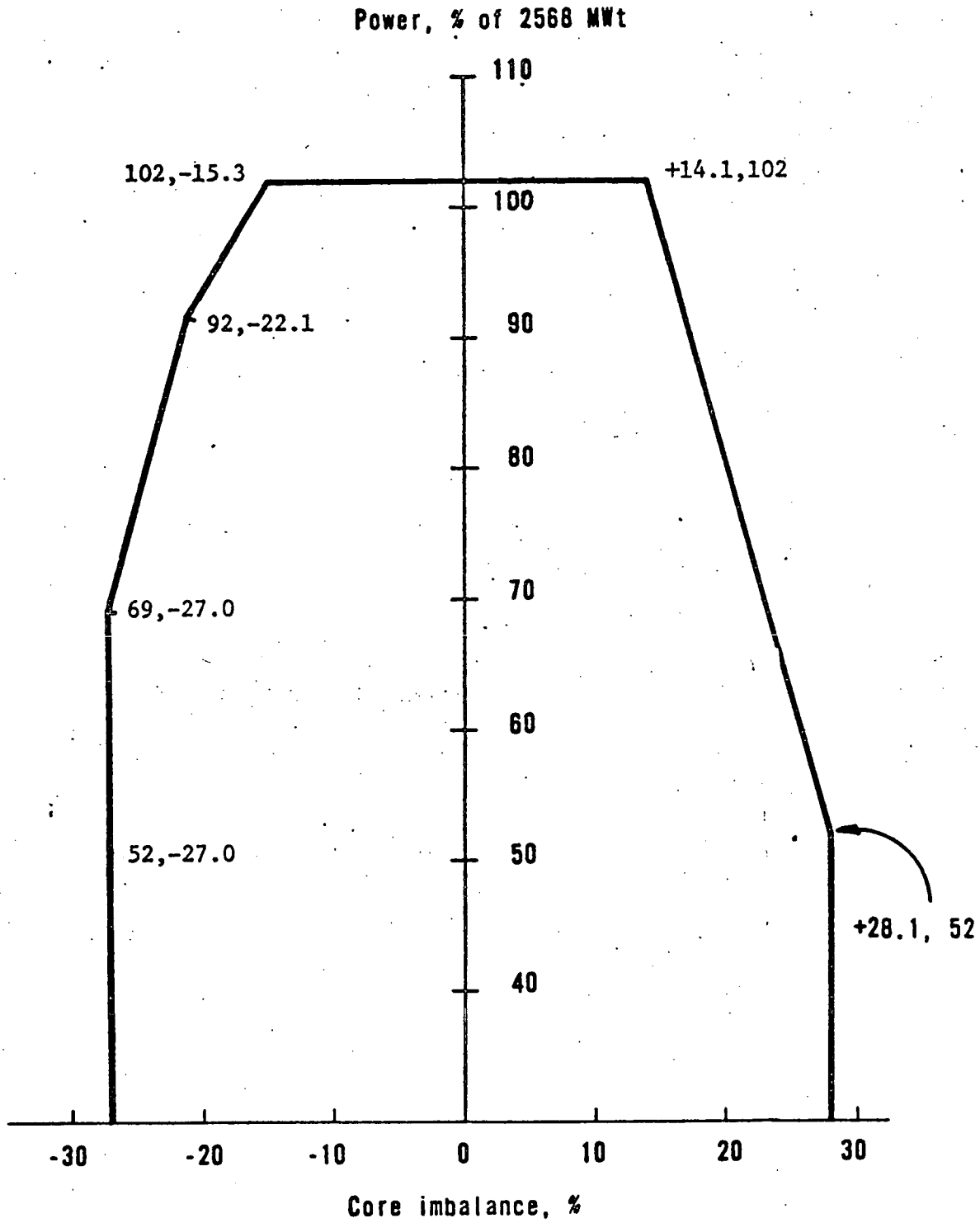

FIGURE A-5

OPERATIONAL POWER IMBALANCE ENVELOPE

UNIT 1

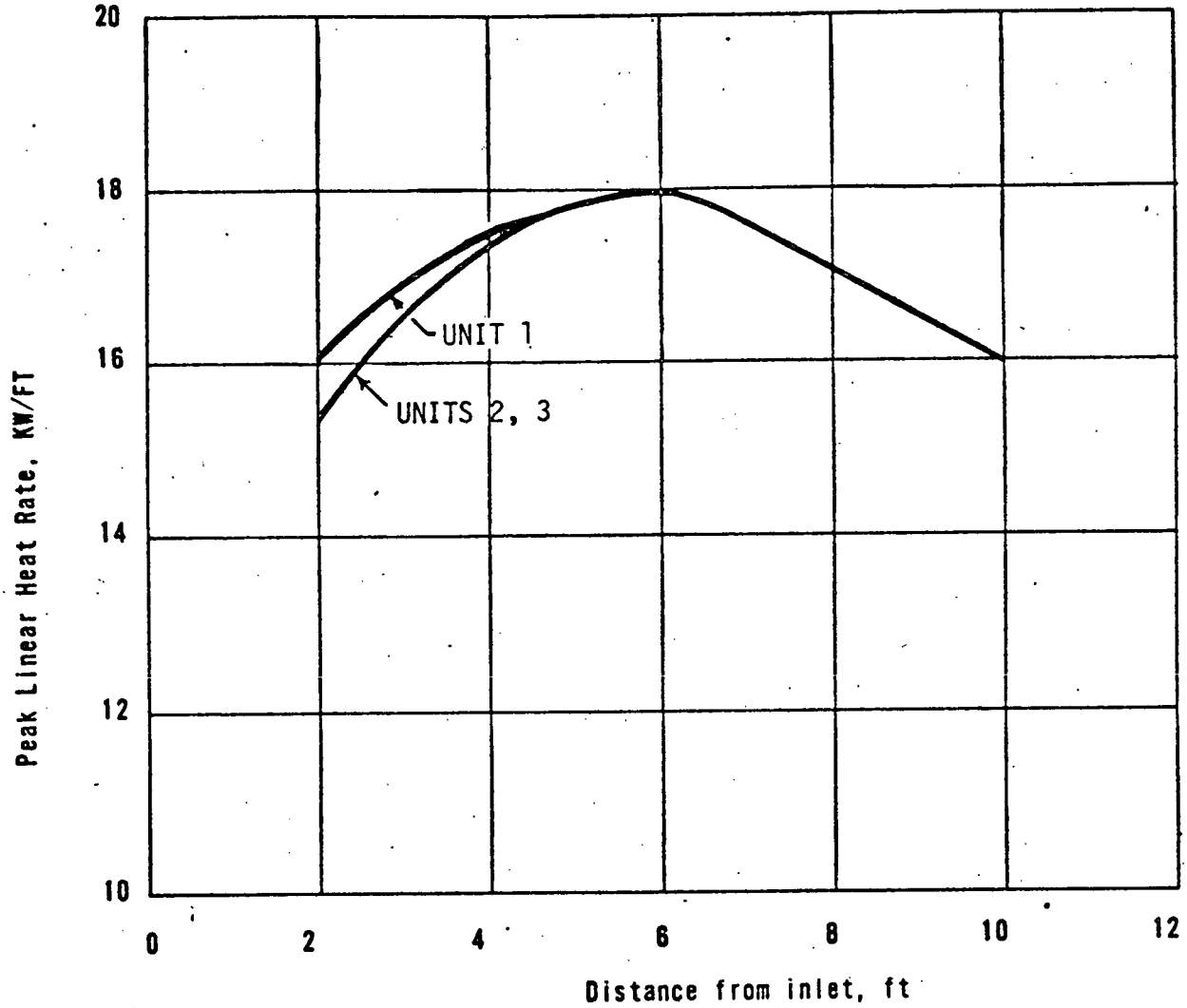

FIGURE A-6

LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE