

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO: 7514

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FROM: Buke Power Company Charlotte, NC W O Parker Jr			DATE OF DOC 7-9-75	DATE REC'D 7-14-75	LTR XX	TWX	RPT	OTHER
TO: Mr Giambusso			ORIG one signed	CC	OTHER	SENT NRC PDR XX SENT LOCAL PDR XX		
CLASS	UNCLASS XXXXX	PROP INFO	INPUT	NO CYS REC'D 1	DOCKET NO: 50-269/270/287			

DESCRIPTION:

Ltr re our 12-27-75 order for modification
.....trans the following:

ACKNOWLEDGED
DO NOT REMOVE

PLANT NAME: **Oconeec 1-3**

ENCLOSURES:

Proposed amdt to OL/Change to Tech Specs:
Consisting of revised ECCS acceptance Criteria
&-addl & supporting info for the change.....

(40 cys encl rec'd)

FOR ACTION/INFORMATION 7-14-75 ehf

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1 - ASLB	NEWMARK/BLUME/AGBABIAN	
1 - Newton Anderson		
1 - ACRS HOLDING/SENT		

TO C.A. Sheppard

DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

REGULATORY DOCKET FILE COPY

TELEPHONE: AREA 704
373-4083

July 9, 1975

Mr. Angelo Giambusso, Director
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

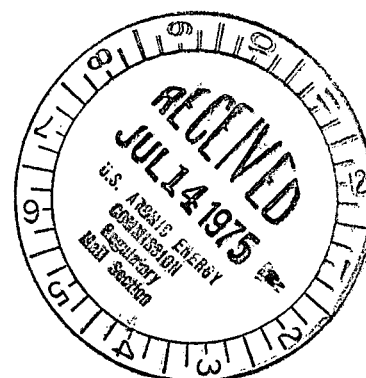
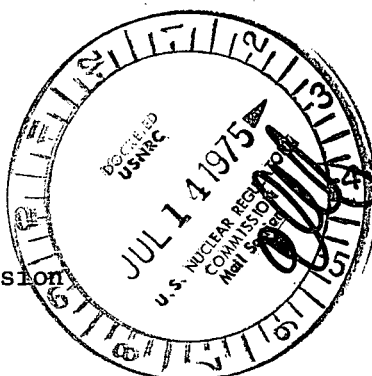
Re: Oconee Nuclear Station
Docket Nos. 50-269, 50-270, 50-287

Dear Mr. Giambusso:

Pursuant to the Commission's December 27, 1974 Order for Modification of License for Oconee Nuclear Station, Units 1, 2 and 3, a re-evaluation of ECCS cooling performance has been performed. This re-evaluation utilized a calculational model conforming to the provisions of 10 CFR 50, Section 50.46. Based on this re-evaluation, Duke Power Company is submitting herewith proposed changes to Oconee Technical Specifications 3.1.3.5, 3.1.7, 3.5.2.3 and 3.5.2.5. The proposed revisions are indicated in the attached replacement pages (Attachment 1). The proposed changes to Specifications 3.1.3.5 and 3.5.2.3, which are necessary to incorporate minimum ejected rod worth criteria into the rod withdrawal limits, are comparable to those submitted on May 9, 1975 and are included here for completeness.

The evaluation model utilized in performing the re-evaluation of ECCS cooling performance is described in Babcock and Wilcox (B&W) non-proprietary Topical Report BAW-10104, "B&W's ECCS Evaluation Model." Non-proprietary Topical Report BAW-10103, "ECCS Evaluation of B&W's 177 FA Lowered Loop NSS," describes the results of the re-evaluation for a generic B&W unit of the Oconee class. Since the generic parameters used in the re-evaluation are conservative for units of this type, and the parameters associated with Oconee 1, 2 and 3 are bounded by those utilized in the generic analysis, BAW-10103 provides a conservative evaluation of ECCS performance for the Oconee units. In the case of Oconee 1, portions of the analysis presented in BAW-10103 were also performed utilizing specific Oconee 1 parameters. This analysis, and the results thereof, are described in Attachment 2.

The results presented in BAW-10103 and in Attachment 2 demonstrate the conformance of the Oconee units to the criteria of 10CFR50, Section 50.46



Mr. Angelo Giambusso

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under the operating conditions specified in the attached proposed technical specifications.

In addition to the above, and in accordance with Mr. Robert A. Purple's letter of June 13, 1975 to Mr. A. C. Thies, the following information is also provided:

1. Break Spectrum and Partial Loop Operation

The requested information for operation with four reactor coolant pumps is presented in BAW-10103 and in Attachment 2. For operation with three or two reactor coolant pumps, the proposed Technical Specifications are based on ECCS limits for four pump operation and on ejected rod worth criteria. Additional information concerning the limits for three and two pump operation will be provided by August 1, 1975.

2. Potential Boron Precipitation

The requested information was provided by Mr. A. C. Thies' letters of April 16, 1975 and May 30, 1975 in response to Mr. Purple's letter of March 14, 1975.

3. Single Failure Analysis

A single failure analysis for manually-controlled, electrically-operated ECCS valves has been performed and is provided as Attachment 3 hereto. Valves 1CF-1, 2CF-1 and 3CF-1, and 1CF-2, 2CF-2 and 3CF-2, core flood tank discharge isolation valves, identified in Attachment 3 are ECCS valves which are currently required by Technical Specifications to have power disconnected during operation - see Technical Specification 3.3.3(c). Based on the information provided in Attachment 3, it is concluded that no station modifications or changes to the Technical Specifications are necessary in order to protect against a single failure of a manually-controlled, electrically-operated ECCS valve.

4. Submerged Valves

A review of equipment arrangement to determine if any valve motors within the Reactor Building will become submerged following a LOCA has been performed and those valves which may be affected are identified in Attachment 4. As can be seen, flooding of the valve motors identified would have no effect on short term or long term ECCS functions, or on containment isolation.

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5. Containment Pressure

The containment pressure used to evaluate the performance capability of the ECCS has been calculated in accordance with the methods described in BAW-10104 and the results thereof are presented in BAW-10103.

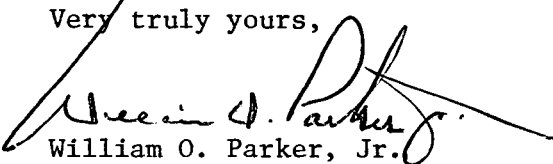
Also, as requested in the staff's Safety Evaluation Report which accompanied the Commission's December 27, 1974 Order, and in accordance with Mr. Purple's letter of February 10, 1975 to Mr. A. C. Thies, as-built passive containment heat-sink data have been compiled and are given in Attachment 5.

Timely approval of the attached proposed Technical Specifications is requested. In the interim, in accordance with the Commission's December 27, 1974 Order for Modification of License, operation of the Oconee units is continuing 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 to the Order.

Forty copies of this letter and attachments are enclosed.

Very truly yours,



William O. Parker, Jr.

DCH:vr

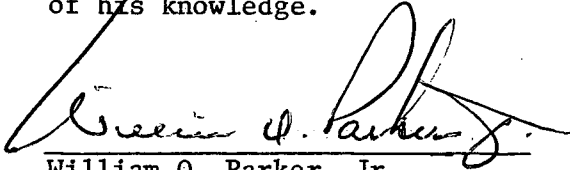
Enclosures

Mr. Angelo Giambusso

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WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47 and DPR-55, and that all statements and matters set forth therein are true and correct to the best of his knowledge.



William O. Parker, Jr.
Vice President

ATTEST:



Dorothea B. Stroupe
Assistant Secretary

Subscribed and sworn to before me this 9th day of July, 1975.



Notary Public

My Commission Expires:

December 3, 1979

Attachment 1

Proposed Technical Specification Revisions
Replacement Pages

3.1.3 Minimum Conditions for Criticality

Specification

- 3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.
- 3.1.3.2 Reactor coolant temperature shall be above DTT + 10°F.
- 3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.
- 3.1.3.4 The reactor shall be maintained subcritical by at least 1%Δk/k until a steam bubble is formed and a water level between 80 and 396 inches is established in the pressurizer.
- 3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. The regulating rods shall then be positioned within their position limits defined by Specification 3.5.2.5 prior to deboration.

Bases

At the beginning of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525°F, the consequences are acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1%Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT + 10°F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident. (3)

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

- (1) FSAR, Section 3
- (2) FSAR, Section 3.2.2.1.4
- (3) FSAR, Supplement 3, Answer 14.4.1

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

The moderator temperature coefficient shall not be positive at power levels above 95 percent of rated power.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4} \Delta k/k/^{\circ}F$ corrected to 95% rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4} \Delta k/k/^{\circ}F$. The moderator coefficient is expected to be zero or negative prior to completion of startup tests.

When the hot zero power value is corrected to obtain the hot full power value, the following corrections will be applied.

A. Uncertainty in isothermal measurement

The measured moderator temperature coefficient will contain uncertainty on the account of the following:

1. $\pm 0.2^{\circ}F$ in the ΔT of the base and perturbed conditions.
2. Uncertainty in the reactivity measurement of $\pm 0.1 \times 10^{-4} \Delta k/k$.

Proper corrections will be added for the above conditions to result in a conservative moderator coefficient.

B. Doppler coefficient at hot zero power

During the isothermal moderator coefficient measurement at hot zero power, the fuel temperature will increase by the same amount as the moderator. The measured temperature coefficient must be increased by $0.16 \times 10^{-4} (\Delta k/k)/^{\circ}F$ to obtain a pure moderator temperature coefficient.

C. Moderator temperature change

The hot zero power measurement must be reduced by $.09 \times 10^{-4} (\Delta k/k)/^{\circ}F$. This corrects for the difference in water temperature at zero power ($532^{\circ}F$) and 15% power ($580^{\circ}F$) and for the increased fuel temperature effects at 15% power. Above this power, the average moderator temperature remains $580^{\circ}F$. However, the coefficient, α_m , must also be adjusted for the interaction of an average moderator temperature with increased fuel temperatures. This correction is $-.001 \times 10^{-4} \Delta \alpha_m / \Delta \%$ power. It adjusts the 15% power α_m to the moderator coefficient at any power level above 15% power. For example, to correct to 100% power, α_m is adjusted by $(-.001 \times 10^{-4}) (85\%)$, which is $-.085 \times 10^{-4} \Delta \alpha_m$.

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall be not less than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

- a. If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.
- b. If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.
- c. If a control rod cannot meet the requirements of Specification 4.7.1, the rod shall be declared inoperable.
- d. If a control rod is found to be improperly programmed per Specification 4.7.2, the rod shall be declared inoperable until properly programmed.
- e. Operation with more than one inoperable rod in the safety or regulating rod groups shall not be permitted.
- f. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position, an evaluation shall be initiated immediately to verify the existence of 1% $\Delta k/k$ hot shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are fully withdrawn, whichever occurs first. Simultaneously, a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

- g. If within one (1) hour of determination of an inoperable rod, it is not determined that a $1\% \Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.
- h. Following the determination of an inoperable rod, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.
- i. If a control rod in the regulating or safety rod groups is declared inoperable, power shall be reduced to 60 percent of the thermal power allowable for the reactor coolant pump combination.
- j. If a control rod in the regulating or axial power shaping groups is declared inoperable, operation above 60 percent of rated power may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 3.5.2.2.a and the withdrawal limits of Specification 3.5.2.5.c.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Power Tilt

- a. Whenever the quadrant power tilt exceeds 4 percent, except for physics tests, the quadrant tilt shall be reduced to less than 4 percent within two hours or the following actions shall be taken:
 - (1) If four reactor coolant pumps are in operation, the allowable thermal power shall be reduced by 2 percent of full power for each 1 percent tilt in excess of 4 percent below the power level cutoff (see Figures 3.5.2-1A1, 3.5.2-1B1, 3.5.2-1B2, 3.5.2-1B3, 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3).
 - (2) If less than four reactor coolant pumps are in operation, the allowable thermal power shall be reduced by 2 percent of full power for each 1 percent tilt below the power allowable for the reactor coolant pump combination as defined by Specification 2.3.
 - (3) Except as provided in 3.5.2.4.b, the reactor shall be brought to the hot shutdown condition within four hours if the quadrant tilt is not reduced to less than 4 percent after 24 hours.
- b. If the quadrant tilt exceeds 4 percent and there is simultaneous indication of a misaligned control rod per Specification 3.5.2.2, reactor operation may continue provided power is reduced to 60 percent of the thermal power allowable for the reactor coolant

pump combination.

- c. Except for physics tests, if quadrant tilt exceeds 9 percent, a controlled shutdown shall be initiated immediately and the reactor shall be brought to the hot shutdown condition within four hours.
- d. Whenever the reactor is brought to hot shutdown pursuant to 3.5.2.4.a(3) or 3.5.2.4.c above, subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the power range high flux setpoint allowable for the reactor coolant pump combination are restricted by a reduction of 2 percent of full power for each 1 percent tilt for the maximum tilt observed prior to shutdown.
- e. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15 percent of rated power.

3.5.2.5 Control Rod Positions

- a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.
- b. Operating rod group overlap shall be $25\% \pm 5\%$ between two sequential groups, except for physics tests.
- c. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1A1 and 3.5.2-1A2, (Unit 1), 3.5.2-1B1, 3.5.2-1B2 and 3.5.2-1B3 (Unit 2), and 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 (Unit 3) for four pump operation and on Figures 3.5.2-2A (Unit 1), 3.5.2-2B (Unit 2), and 3.5.2-2C (Unit 3) for three or two pump operation. If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall then be attained within two hours. The minimum shutdown margin required by specification 3.5.2.1 shall be maintained at all times.
- d. Except for physics tests, power shall not be increased above the power level cutoff as shown on Figures 3.5.2-1A1, (Unit 1) 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2), and 3.5.2-1C1, 3.5.2-1C2, 3.5.2-1C3 (Unit 3), unless the following requirements are met.
 - (1) The xenon reactivity shall be within 10 percent of the value for operation at steady-state rated power.
 - (2) The xenon reactivity shall be asymptotically approaching the value for operation at the power level cutoff.
- e. Power level for Unit 1 shall not be greater than the power level cutoff unless one of the following requirements are met.

- (1) 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.
- (2) 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.

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A, 3.5.2-3B, and 3.5.2-3C. If the imbalance is not within the envelope defined by Figure 3.5.2-3A, 3.5.2-3B, and 3.5.2-3C, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager.

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A, 3.5.2-3B, and 3.5.2-3C is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final acceptance criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

- a. Nuclear uncertainty factors
- b. Thermal calibration
- c. Fuel densification effects
- d. Hot rod manufacturing tolerance factors

The 25% + 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

<u>Group</u>	<u>Function</u>
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.5% $\Delta k/k$ (Unit 1) or 0.65% $\Delta k/k$ (Units 2 and 3) at rated power. These values have been shown to be safe by the safety analysis (2,3,4) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod positions limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.5% $\Delta k/k$ (Unit 1) or 0.65% $\Delta k/k$ (Units 2 and 3) ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5,6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

**Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position limits in Specification 3.5.2.5c ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant tilt and axial imbalance monitoring in Specifications 3.5.2.4 and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

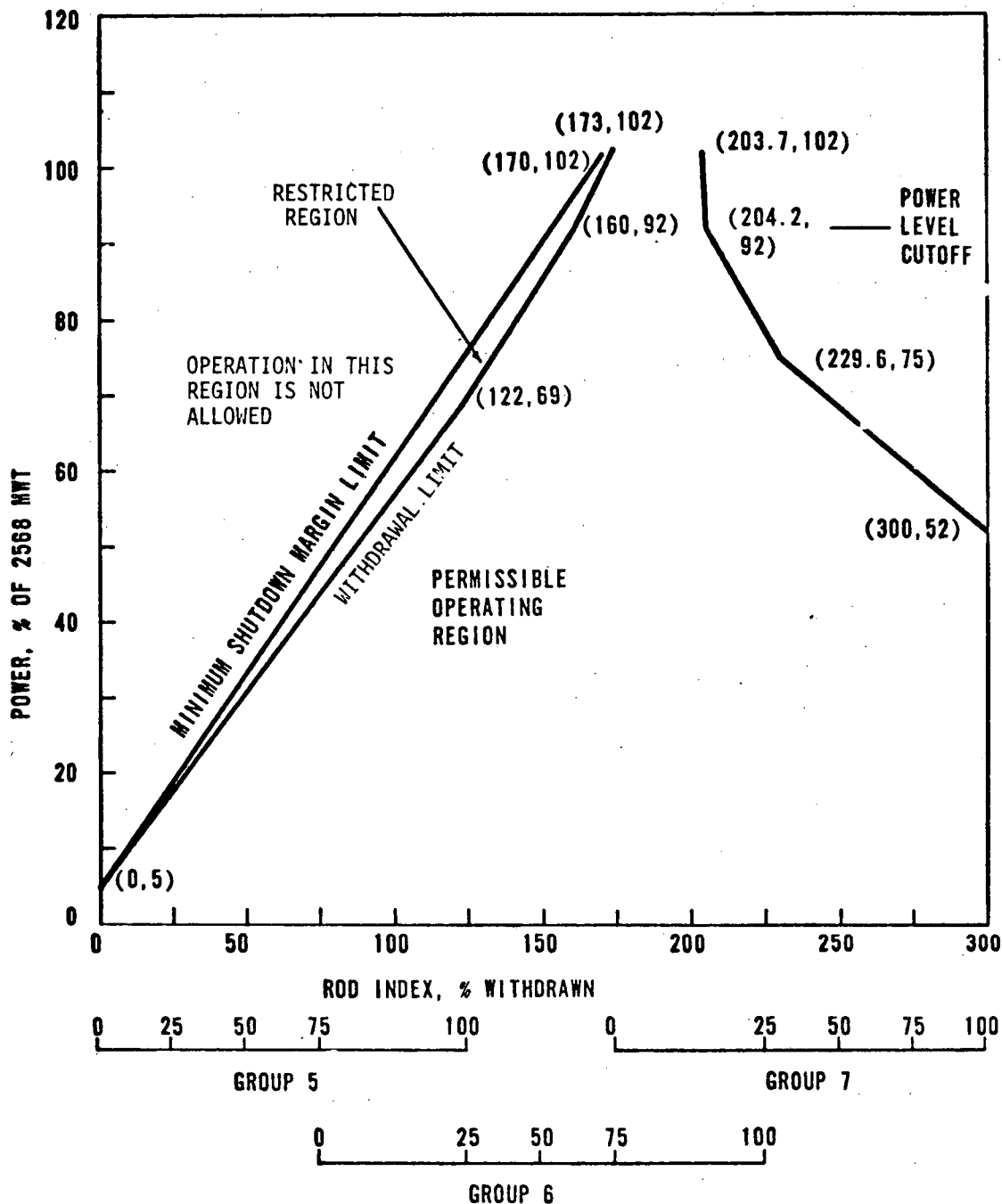
Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptance rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specification 3.5.2.5d and e to prevent excessive power peaking by transient xenon. The xenon reactivity must be beyond the "undershoot" region and asymptotically approaching its equilibrium value at rated power.

REFERENCES

- ¹FSAR Section 3.2.2.1.2
- ²FSAR Section 14.2.2.2
- ³FSAR SUPPLEMENT 9
- ⁴B&W FUEL DENSIFICATION REPORT
 - BAW-1409 (UNIT 1)
 - BAW-1396 (UNIT 2)
 - BAW-1400 (UNIT 3)

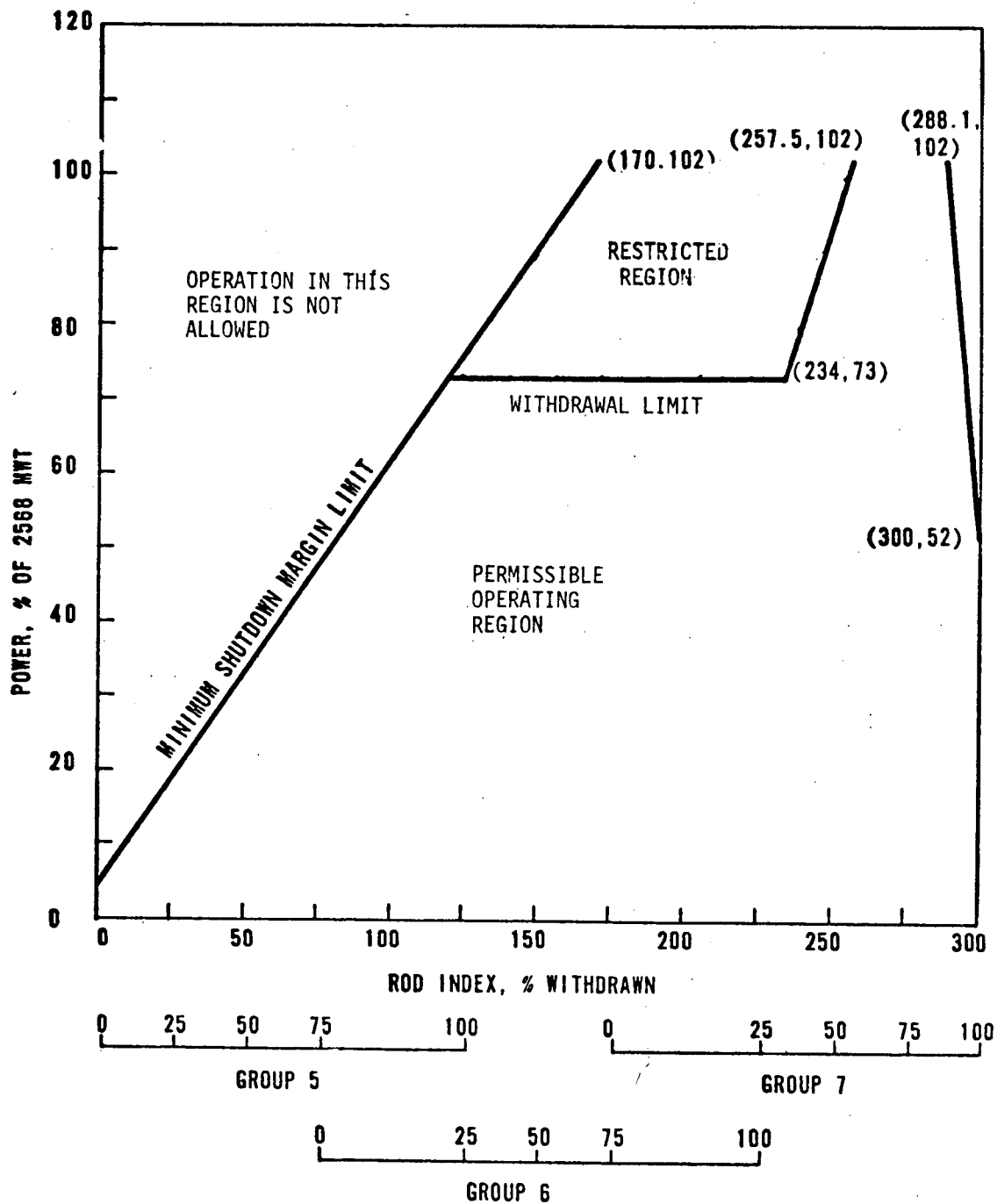
ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE TO THE PERIOD FROM 50±5 EFPD TO 250±5 EFPD.



Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7



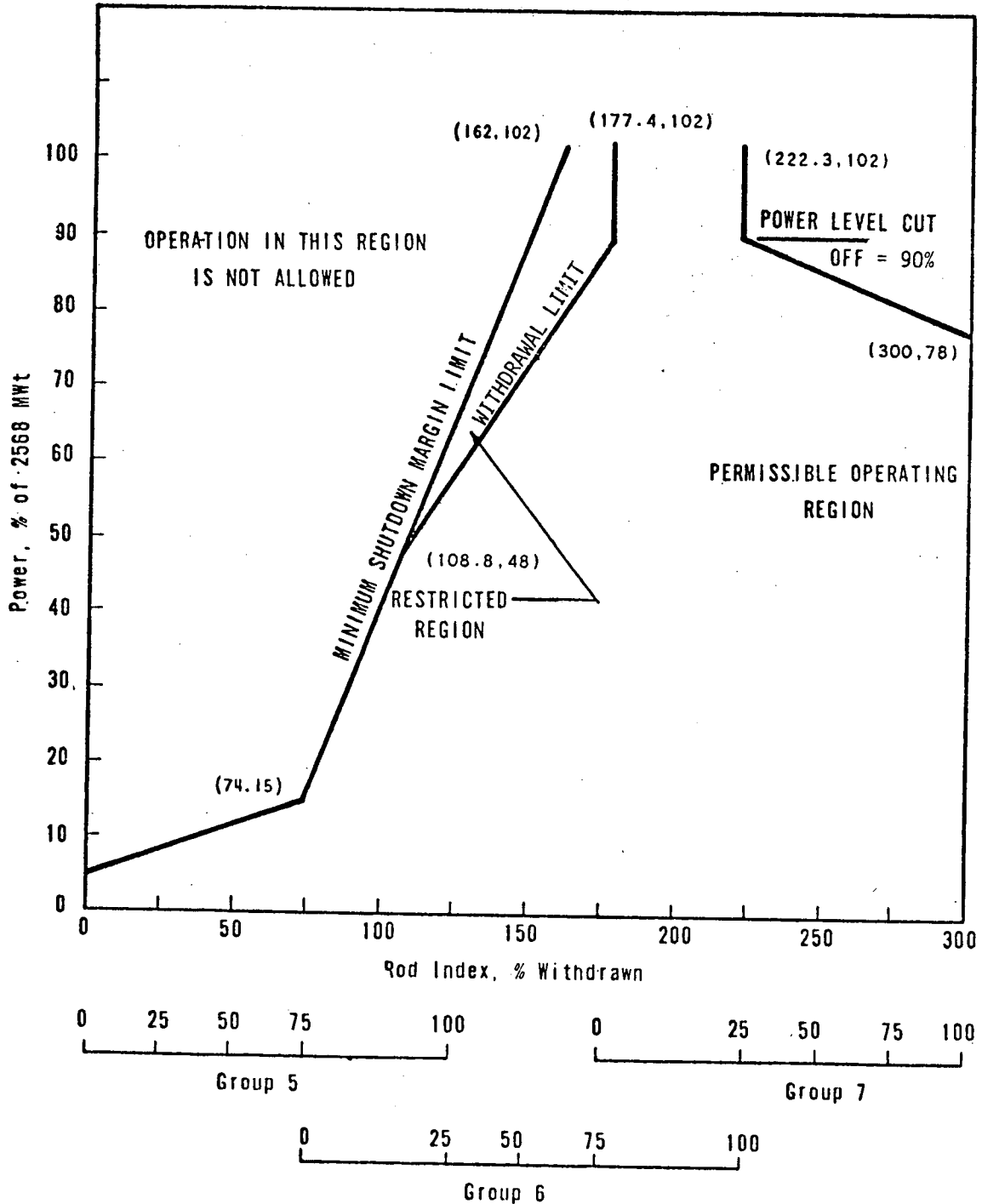
ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE TO THE PERIOD AFTER 250±5 EFPD



Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING
THE PERIOD FROM 100 EFPD TO 250±10 EFPD.



Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

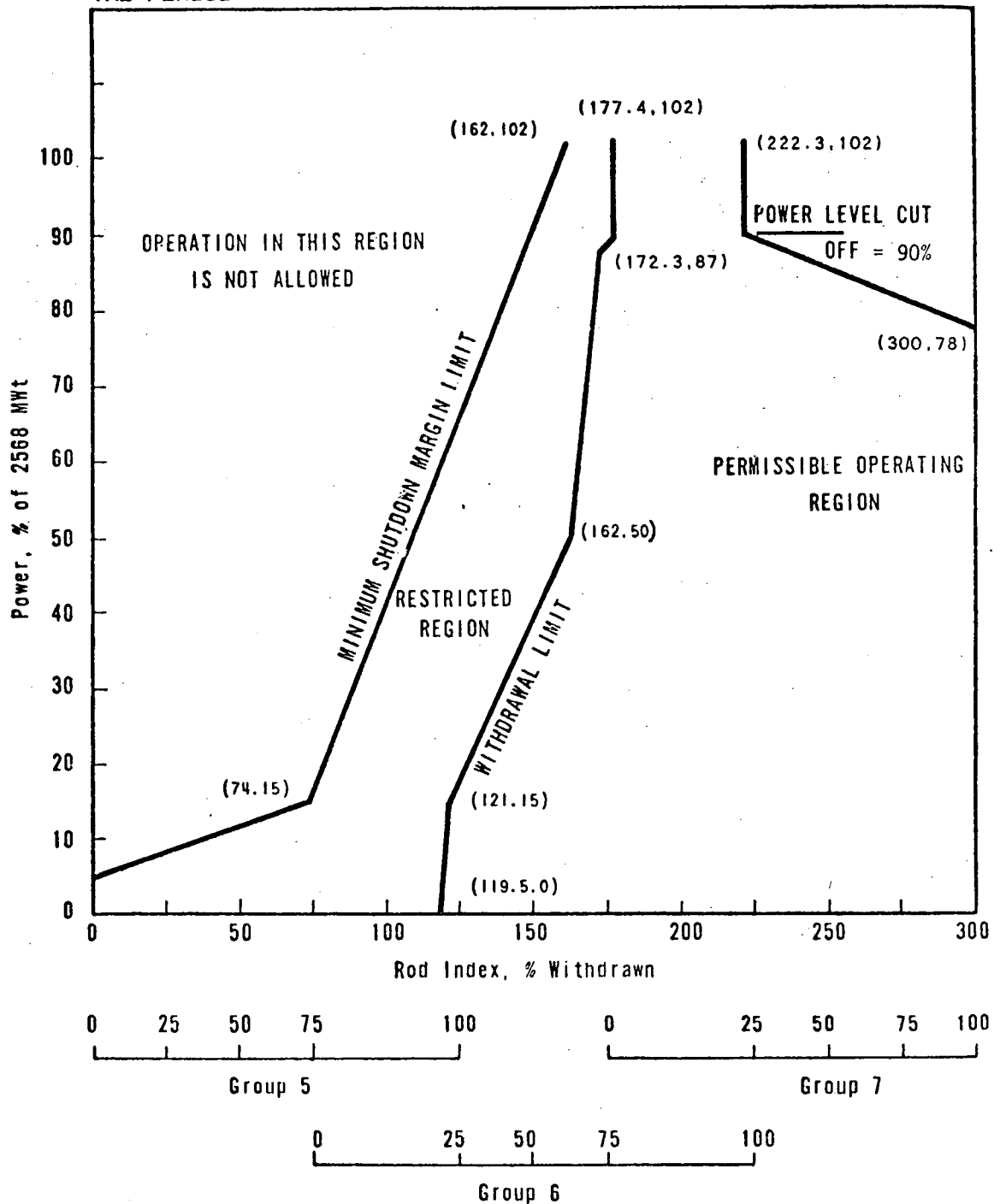
UNIT 2
ROD POSITION LIMITS FOR
4 PUMP OPERATION



OCONEE NUCLEAR STATION

Figure 3.5.2-1B1

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING
THE PERIOD FROM 250±10 EFPD TO 435±10 EFPD.



Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

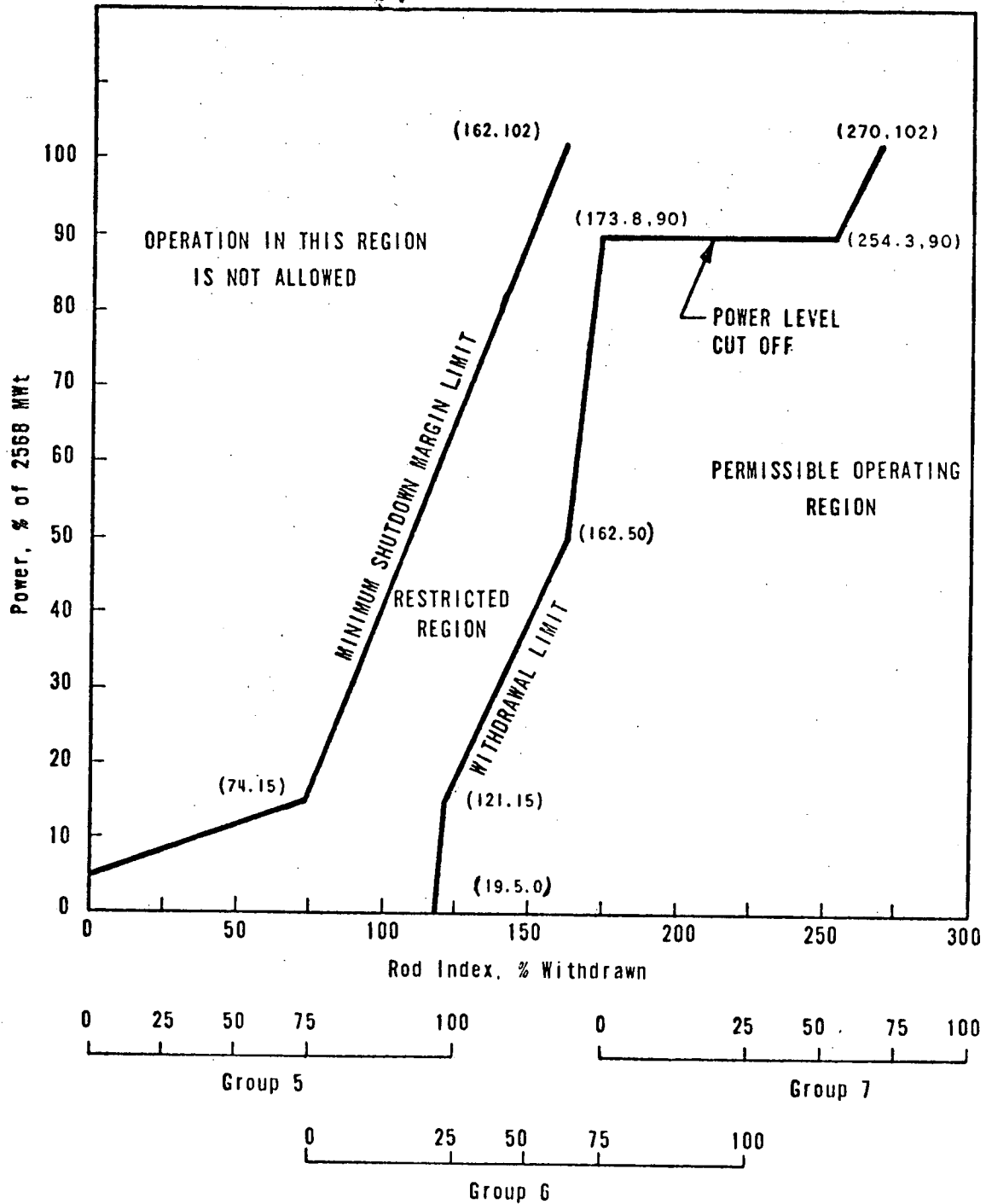
UNIT 2
ROD POSITION LIMITS FOR
4 PUMP OPERATION



OCONEE NUCLEAR STATION

Figure 3.5.2-1B2

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING
THE PERIOD AFTER 43,410 EFPD.



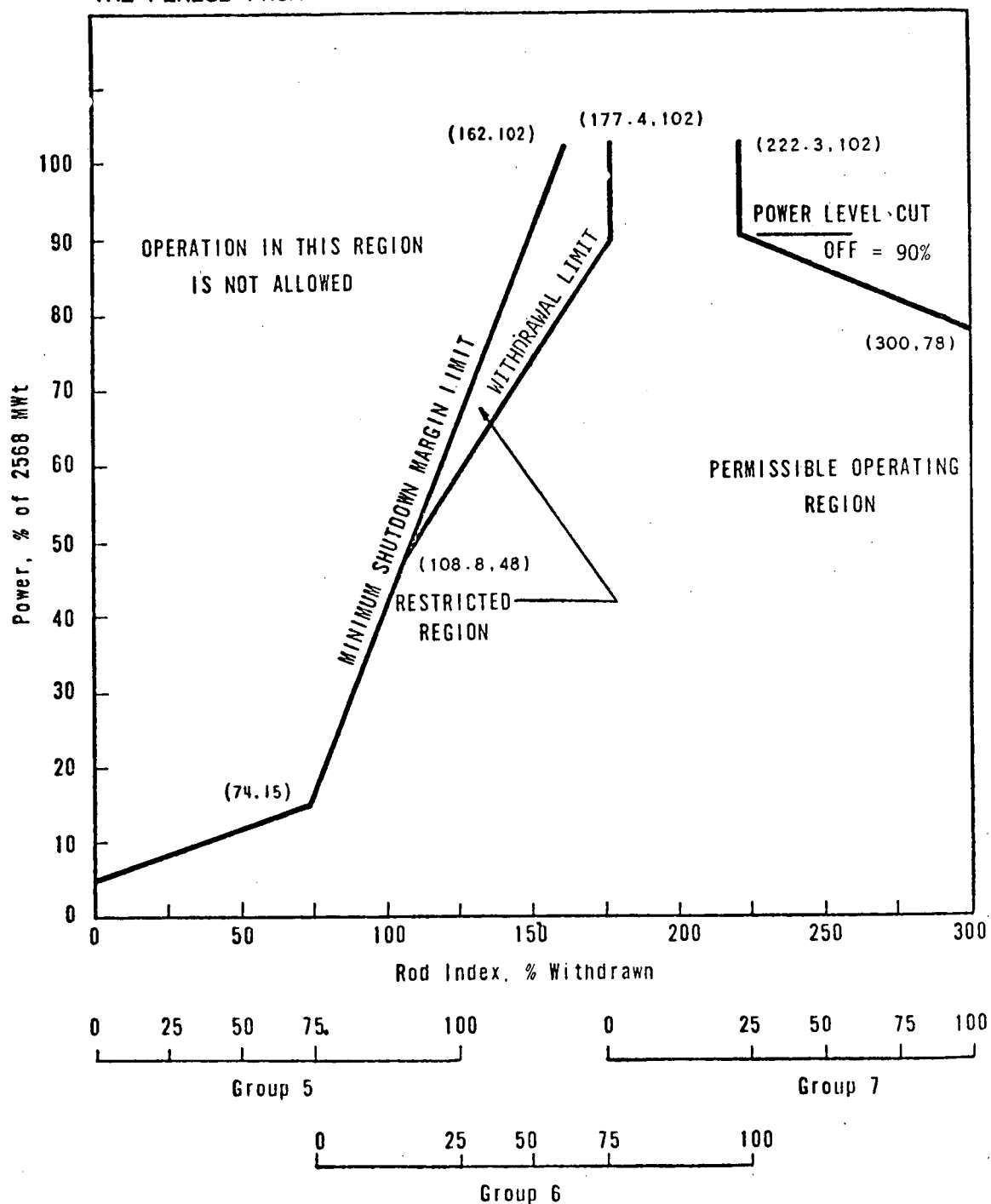
Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

UNIT 2
ROD POSITION LIMITS FOR
4 PUMP OPERATION



OCONEE NUCLEAR STATION

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING
THE PERIOD FROM 100 EFPD TO 250±10 EFPD.



Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

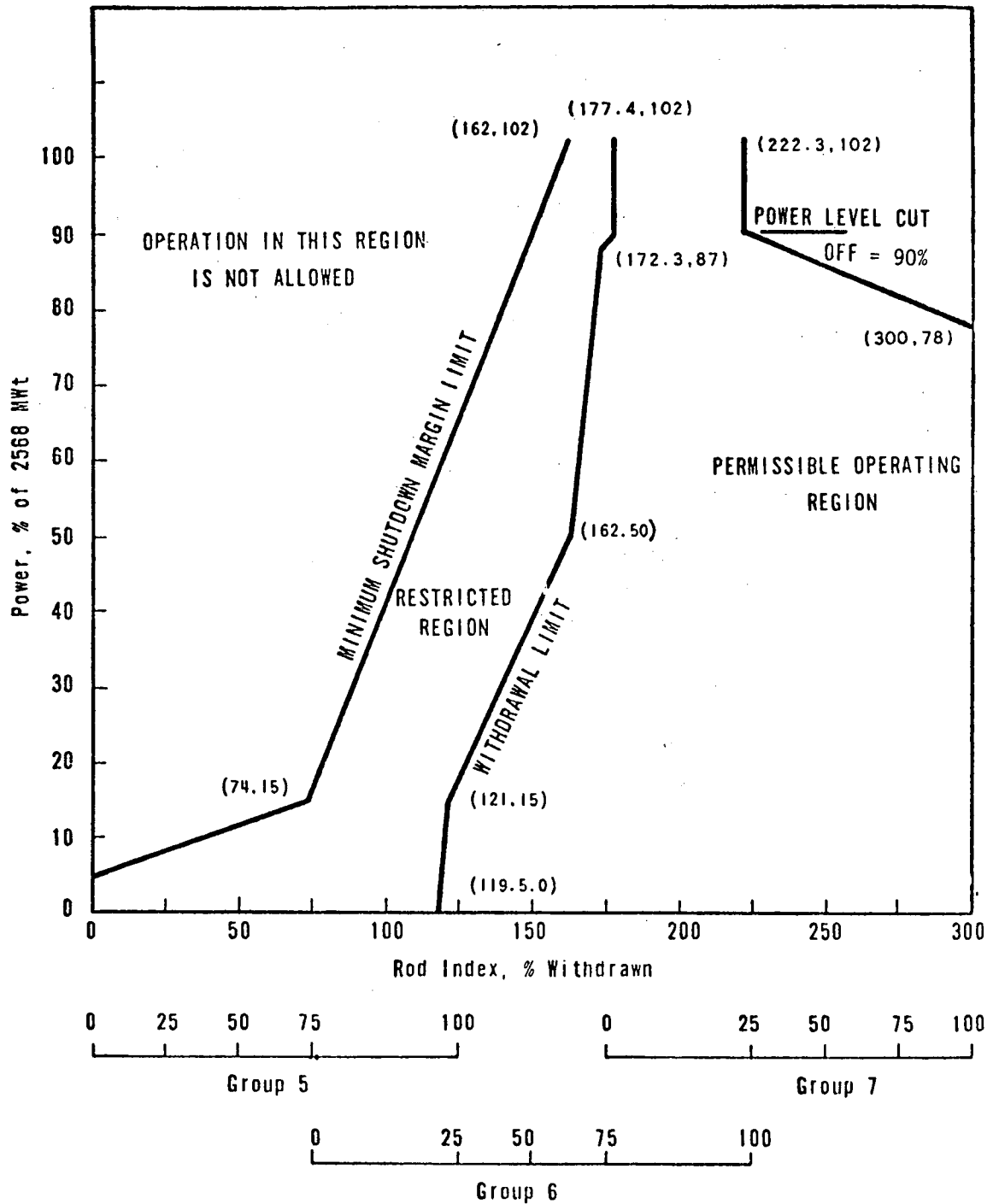
UNIT 3
ROD POSITION LIMITS FOR
4 PUMP OPERATION



OCONEE NUCLEAR STATION

Figure 3.5.2-1C1

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 250±10 EFPD TO 435±10 EFPD.



Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

UNIT 3
ROD POSITION LIMITS FOR
4 PUMP OPERATION

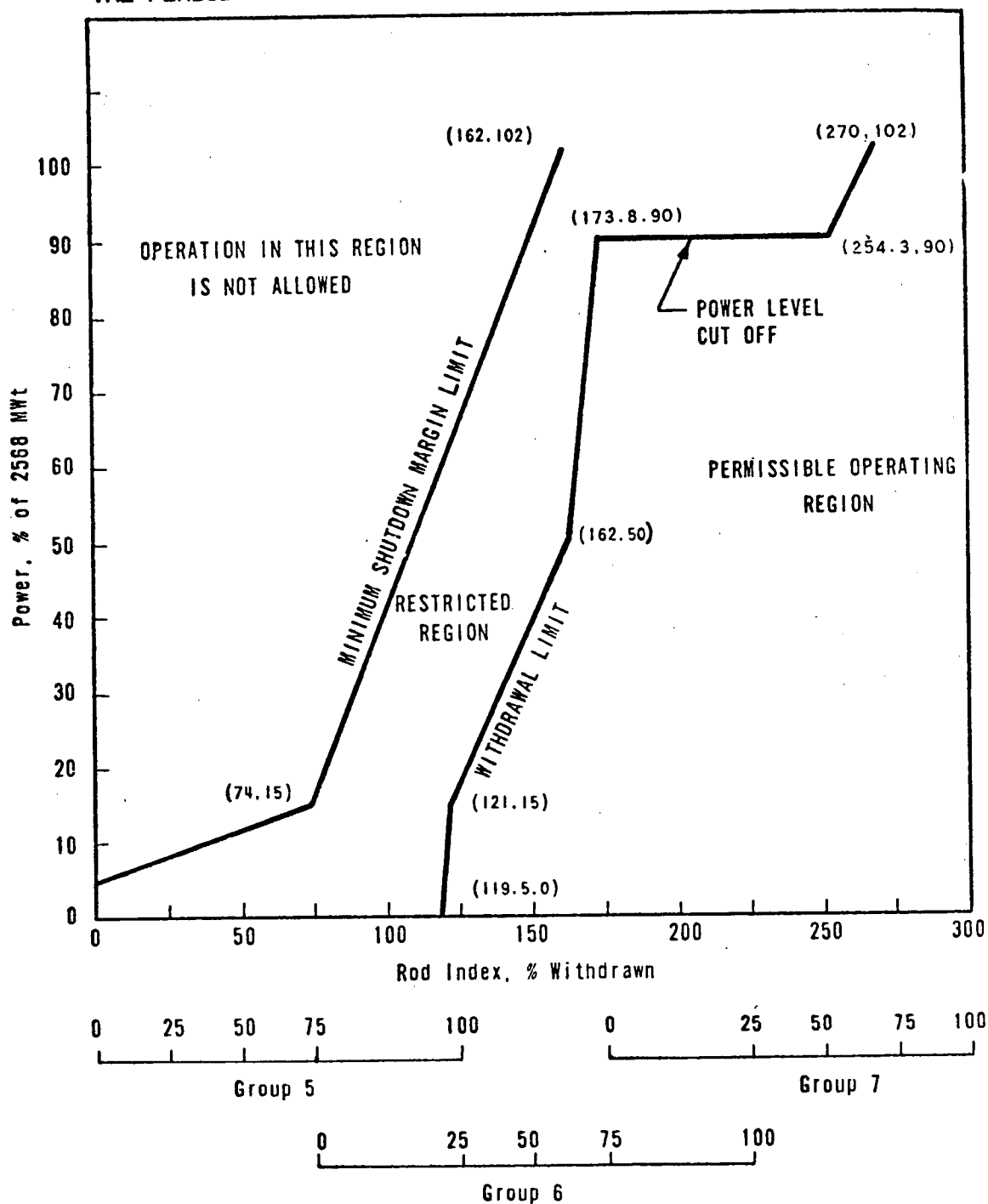


OCONEE NUCLEAR STATION

3.5-16a

Figure 3.5.2-1C2

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING
THE PERIOD AFTER 435 ± 10 EFPD.



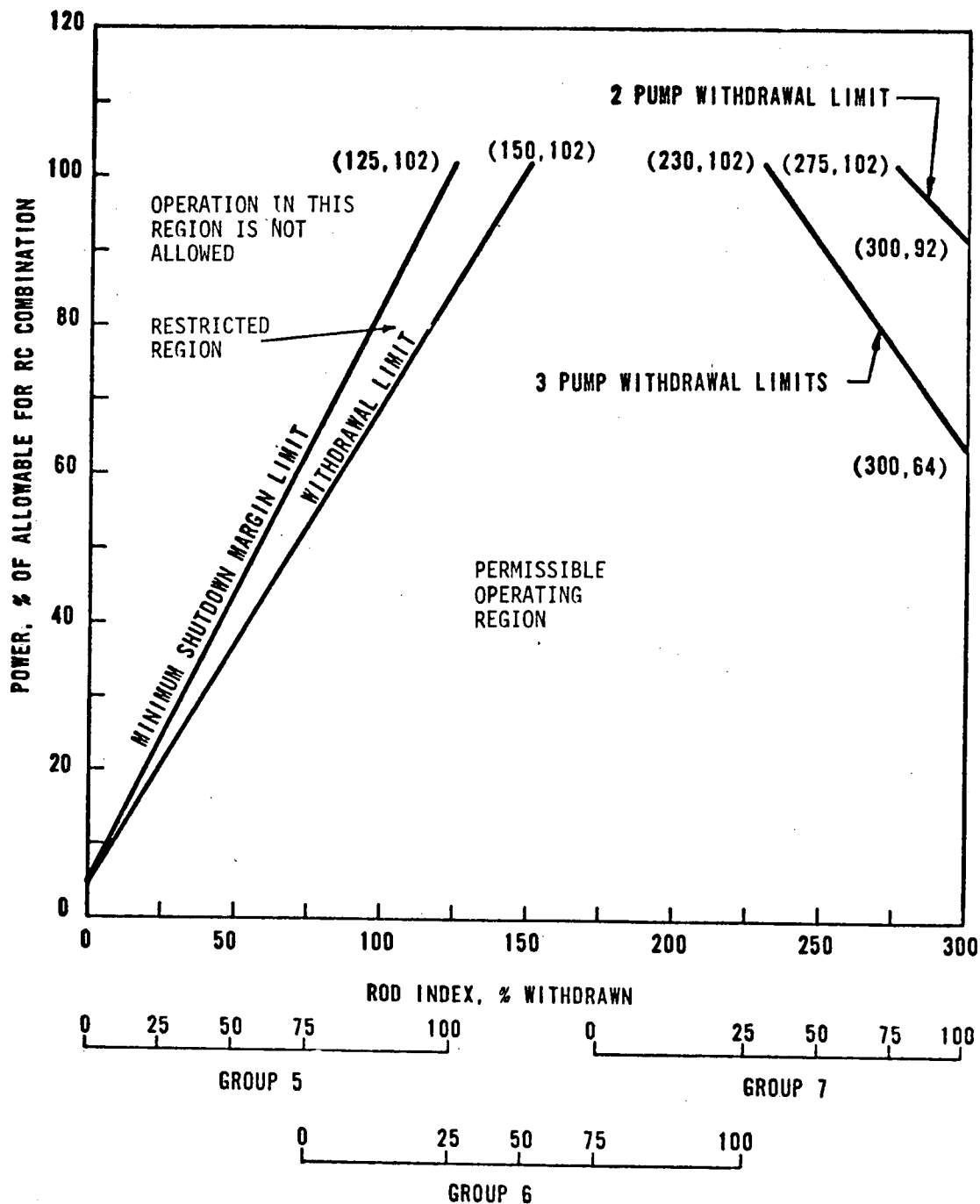
Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

UNIT 3
ROD POSITION LIMITS FOR
4 PUMP OPERATION



OCONEE NUCLEAR STATION

ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE TO
THE PERIOD AFTER 50±5 EFPD

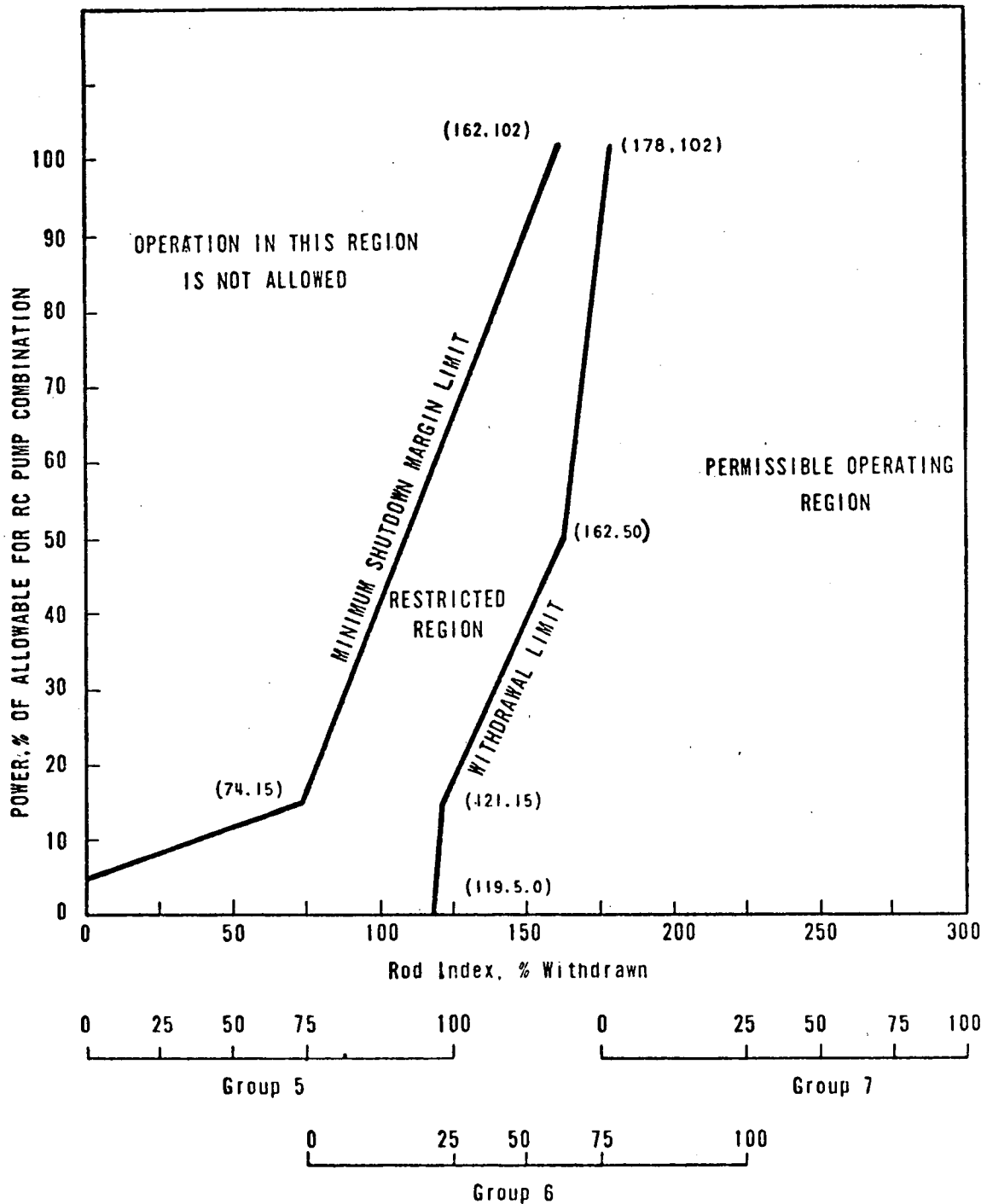


Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7



UNIT 1
ROD POSITION LIMITS FOR
2 AND 3 PUMP OPERATION
OCONEE NUCLEAR STATION

ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION



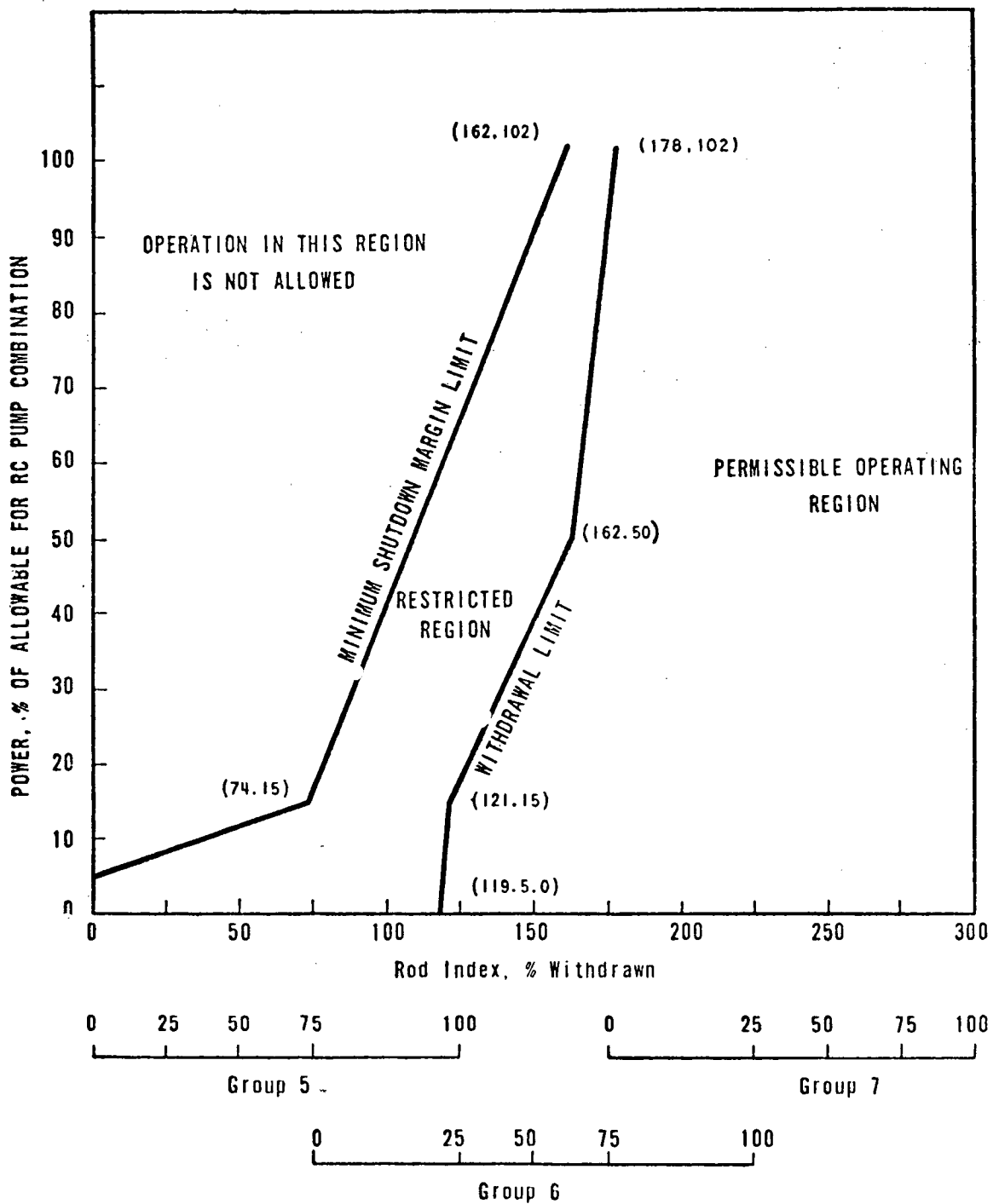
Rod index is the percentage sum of the withdrawal of Groups 5, 6 and 7

UNIT 2
ROD POSITION LIMITS FOR
2 AND 3 PUMP OPERATION



OCONEE NUCLEAR STATION

ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION



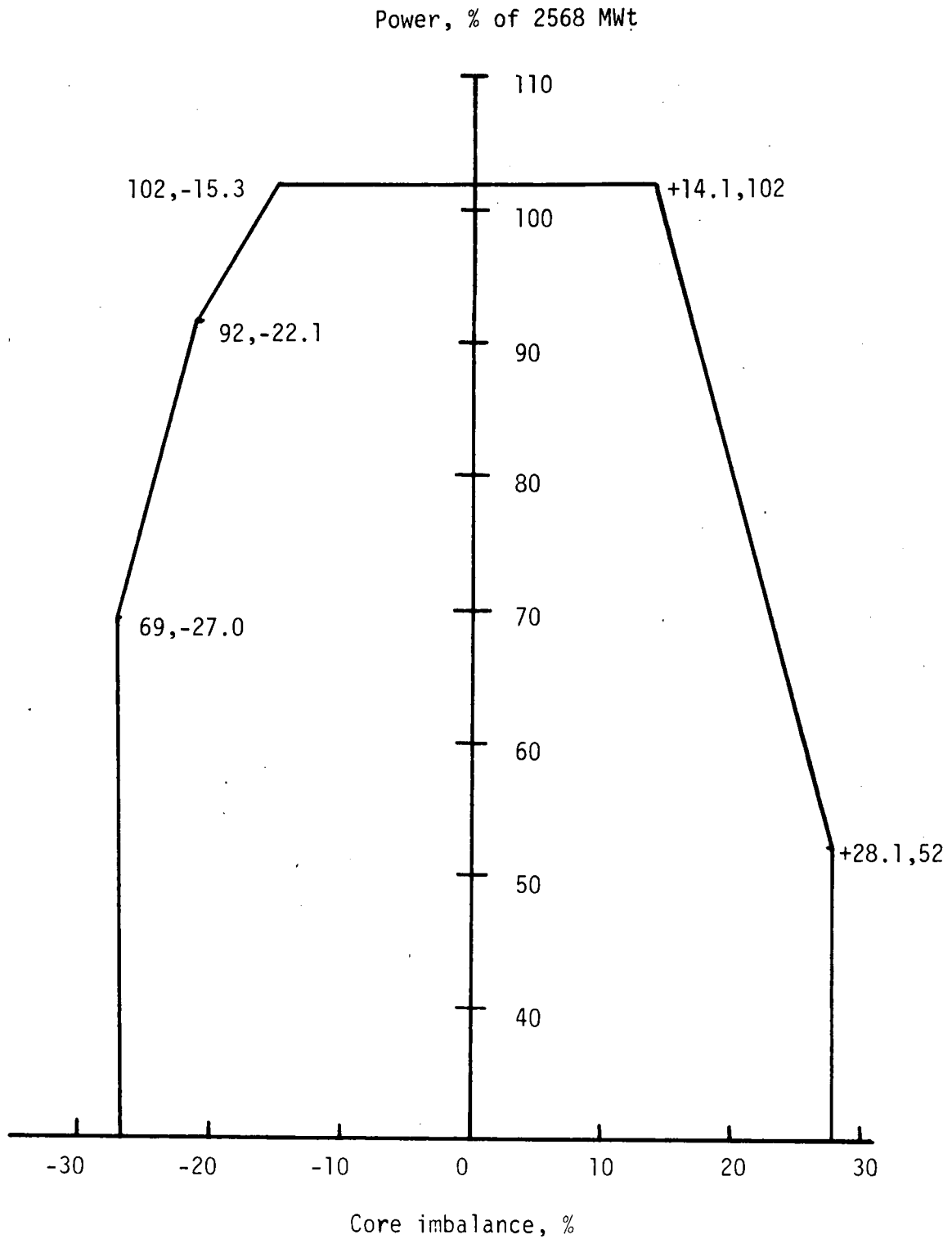
Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

UNIT 3
ROD POSITION LIMITS FOR
2 AND 3 PUMP OPERATION



OCONEE NUCLEAR STATION

Figure 3.5.2-2C

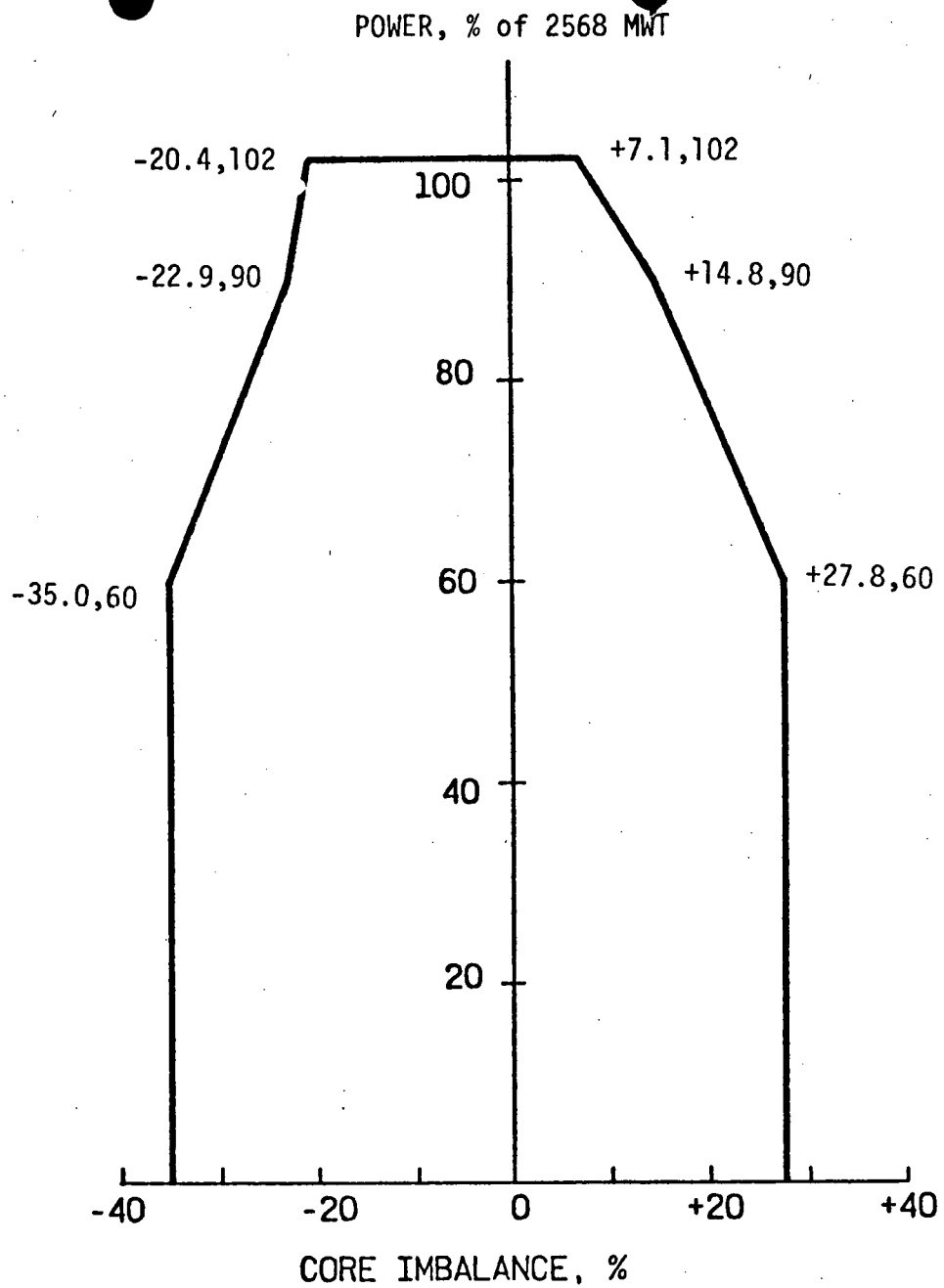


OPERATIONAL POWER IMBALANCE ENVELOPE



UNIT 1
OCONEE NUCLEAR STATION

Figure 3.5.2-3A



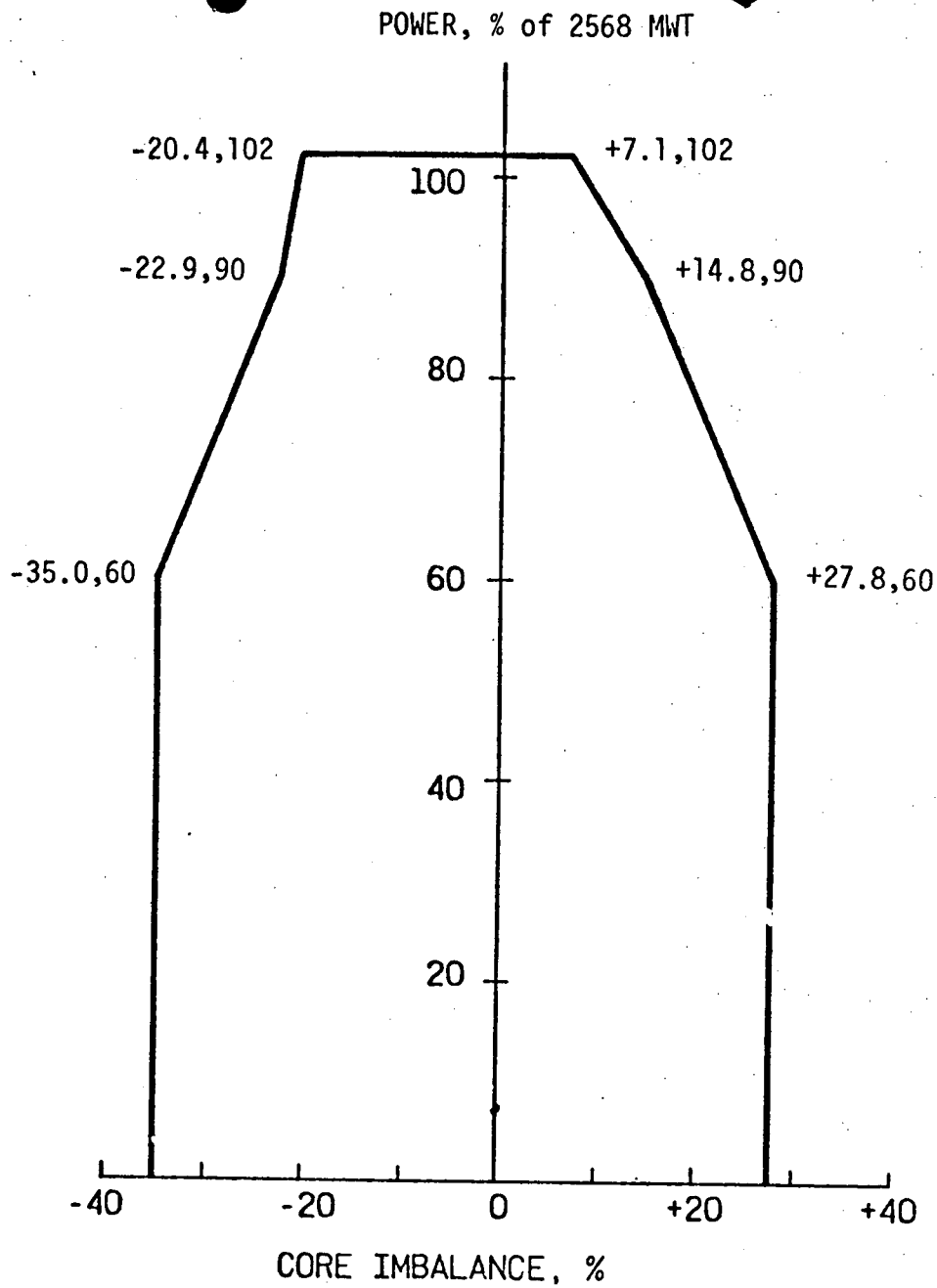
UNIT 2
OPERATIONAL POWER
IMBALANCE ENVELOPE



OCONEE NUCLEAR STATION

3.5-22

Figure 3.5.2-3B

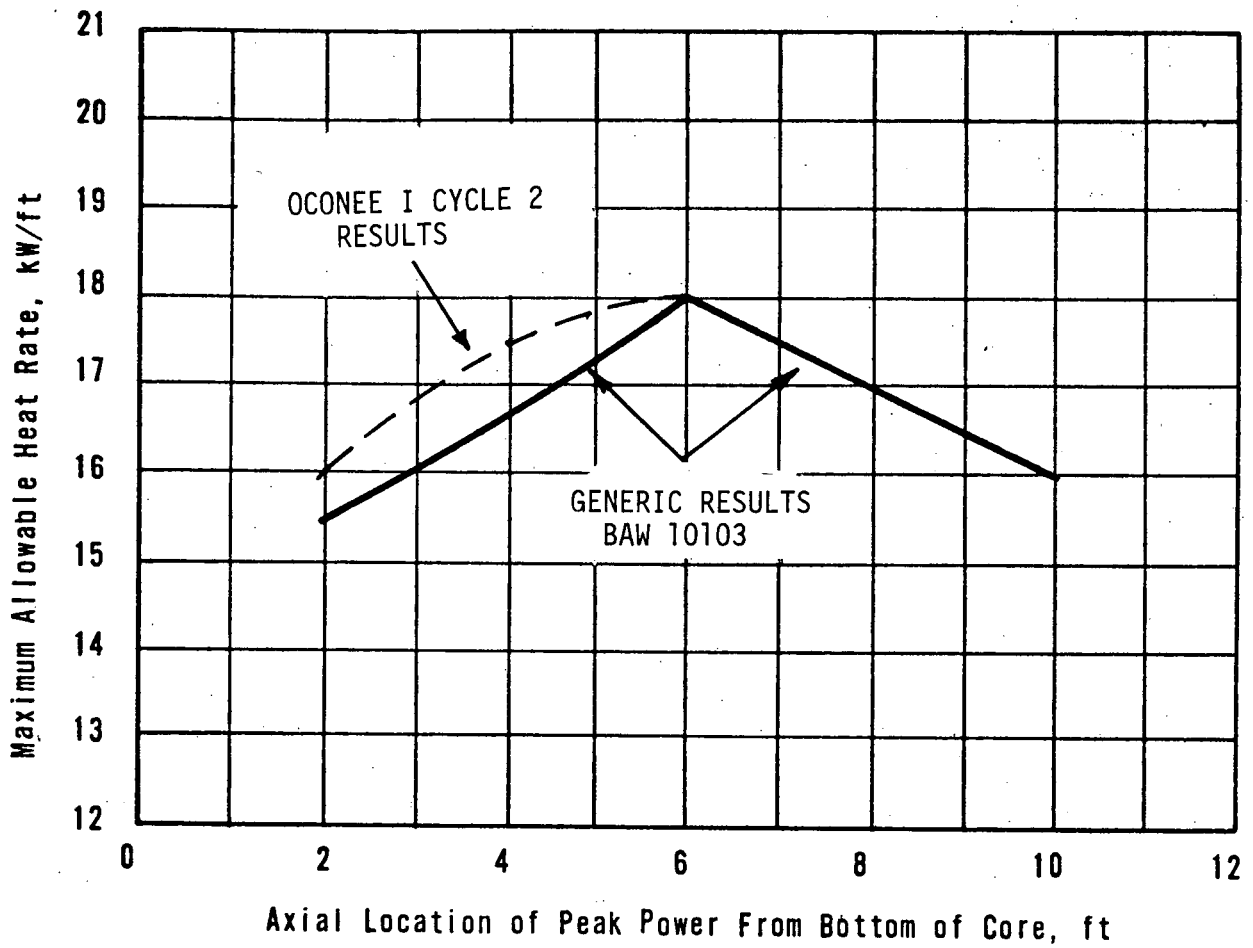


UNIT 3
OPERATIONAL POWER
IMBALANCE ENVELOPE



OCONEE NUCLEAR STATION

Figure 3.5.2-3C



LOCA LIMITED MAXIMUM
ALLOWABLE LINEAR HEAT RATE



OCONEE NUCLEAR STATION

3.5-24

Figure 3.5.2-4

Attachment 2

Oconee 1 ECCS Evaluation

OCONEE I, CYCLE 2
LOCA LIMITS

1. Introduction

In Babcock and Wilcox Topical Report BAW-10103¹, generic LOCA limits applicable to the 177 FA class plants with lowered-loop arrangement (Category 1 plants) were developed. LOCA limits specific to Oconee 1, Cycle 2 fuel are developed in this attachment as were developed in the Oconee 1, Cycle 2 reload report, BAW-1409. The analysis is performed using the B&W evaluation model documented in BAW-10104².

2. Summary and Conclusions

The LOCA limits developed herein are specific to Oconee I, Cycle 2 fuel. The analysis performed to generate these limits meet all the requirements of 10CFR50, Section 50.46. Section 4 of this attachment gives the allowable peak linear heat rates as a function of elevation within the core. The results of the analysis are tabulated below and are shown graphically on Figure 1.

<u>Elevation From Bottom of Core, ft</u>	<u>Allowable Linear Heat Rate, kw/ft</u>	<u>Peak Cladding Temperature, F</u>	<u>Local Metal-Water Reaction, %</u>
2.0	16.0	1930	3.40
4.0	17.5	1978	3.17
*6.0	*18.0	*2146	*5.46
*8.0	*17.0	*2110	*5.19
*10.0	*16.0	*1931	*2.93

*These numbers are from the generic LOCA limits analysis presented in BAW-10103.

3. Input

The method of selection of the input for the CRAFT³, REFLOOD⁴, CONTEMP⁵, and THETA⁶ codes for the generic evaluation of the Category 1 plants is discussed in Section 4 of BAW-10103¹. Also presented in that section is the procedure used in the evaluation of a LOCA. This section will describe the deviations from the input chapter in BAW-10103. To facilitate the ECCS evaluation specific for the Oconee 1, Cycle 2 fuel the following assumptions are made.

- (a) Power Level - Prior to a postulated pipe rupture, the plant is assumed to be operating in a steady-state condition at 102 percent of rated power, 2568 MWt.

- (b) Stored Energy - The initial average fuel temperature takes into account the densification phenomenon. The Batch 4 fuel is used to generate the LOCA limits as described in the Cycle 2 Reload Report, BAW-1409.
- (c) Minimum Capacity - The performance of the LPI system is modeled conservatively at 90 percent of the design flow rate. Also, the CFT resistance used is 6.5 which is a higher value than that obtained by testing. The HPI has not been modeled for the purposes of core cooling in the analysis.
- (d) Backpressure - The determination of minimum backpressure during the reflood phase of a LOCA is based on conservative assumptions. The actual input parameters and calculational methods are discussed in Section 3.3.

3.1 CRAFT

The input description and calculation procedures outlined in Section 4.2 of BAW-10103, with the exception of the value quoted for the core flooding line resistance, remains valid. The core flooding line resistance used is 6.5 which is a value that is higher than the measured value for Oconee I. Use of a higher value for the core flooding line resistance is conservative as this will lead to a longer refill period and lower flooding rates and will result in a higher peak cladding temperature than expected.

3.2 REFLOOD

Since the value for the core flooding line resistance has changed from that reported in BAW-10103, the effect of the hot wall time delay must be reexamined. As shown in Section 4.3 of BAW-10103, the rate of available water storage volume is 119 ft³/sec. The injection rate of ECC water is 108 ft³/sec. Since the storage volume rate exceeds the injection rate, there will be no additional ECC water spillage due to the hot wall phenomena. Therefore, the hot wall time delay has no effect on the length of refill.

The remainder of the input selection description given in Section 4.3 of BAW-10103 is unchanged by the specific analysis of Oconee I.

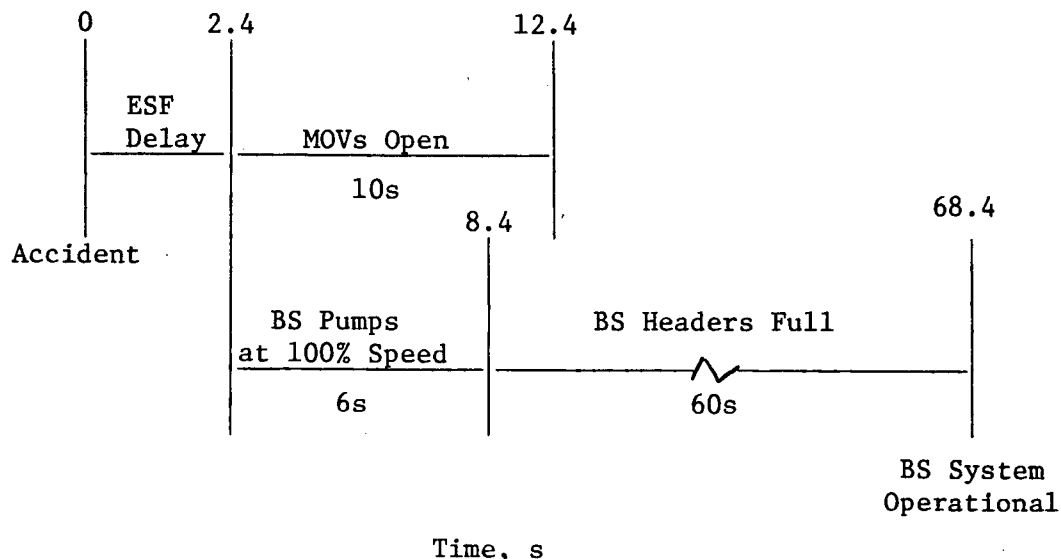
3.3 CONTEMPT

The CONTEMPT input description in Section 4.4 of BAW-10103 was formulated on a generic basis. The entire section, with the exception of the procedure used to generate mass and energy releases, is revised herein. The following assumptions were used for the actual CONTEMPT analysis:

- (1) Initially, the Reactor Building is at 110°F, 13.7 psia, and 100 percent relative humidity. Data taken from the Oconee 1 Reactor Building from December 1973 to June 1974 indicate that

the pressure is nearly always -1.0 psig and varies (though rarely) up to +1.0 psig. The temperatures averaged over the same period were 94°F for the steam generator cavities and 129°F for the dome space. Since the dome makes up over half the building free volume, the temperature for CONTEMPT was weighted equally between these two numbers and rounded down. Values for relative humidity are not readily available, so the conservative value of 100 percent was assumed.

- (2) The outside air temperature is 40°F.
- (3) The containment free volume, including a 5 percent conservatism, is 2,010,000 ft³. Calculations consisting of subtracting the volumes of the internal structures and installed equipment from the volume determined from the building walls and ceiling inside dimensions show the SAR value is less than 3 percent low. Thus, 5 percent is a suitably conservative value for ECCS evaluation.
- (4) All heat removal systems are actuated. The delay times are based on an assumption of no loss of off-site power (see Section 4.6 of BAW-10103 for single failure position). This results in no delay for the fan coolers. The time delay to the sprays is 65 seconds from the start of the accident. A breakdown of the starting time which yields the quickest activation time for the spray systems, is shown below.



The coolant water for these systems is assumed to be at 40°F, and the sprays are considered 100 percent efficient. The 40°F temperature is derived from a lower Technical Specification limit for the conditions in the BWST. The total fan cooler heat removal (HR) rate as a function of Reactor Building atmosphere temperature is

$$HR \text{ (BTU/s)} = 3.0 (0.9T^2 - 76.9T + 1670)$$

where 3.0 represents the number of fan coolers in the containment. The fan coolers are the largest units for the Category 1 Reactor Buildings. The coolant water is assumed to be 40°F, and the coolers are taken as clear and unfouled. The spray flow rate is 1800 gpm for each of the two spray systems. The design flow for the sprays is 1500 gpm. The spray efficiency is taken as one, so that the water is completely heated to the containment vapor temperature as it falls.

- (5) Complete mixing of the spilled ECCS water with the containment atmosphere is considered. The HPI injecting in the broken loop is considered part of the spilled ECCS water.
- (6) Rainout of suspended water is assumed, and a high heat transfer coefficient of 1000 BTU/h-ft² - F between the liquid and vapor regions is used.
- (7) The building is modeled with five heat sinks:

- (a) The Reactor Building walls and dome including the concrete wall, steel liner, and anchors:

Exposed area, ft ²	82,950
Paint thickness, ft	0.00083
Steel thickness, ft	0.04384
Concrete thickness, ft	3.25

- (b) Painted internal steel:

Exposed area, ft ²	218,272
Paint thickness, ft	0.00083
Steel thickness, ft	0.03125

- (c) Unpainted internal steel:

Exposed area, ft ²	31,728
Steel thickness, ft	0.03125

- (d) Unpainted stainless steel:

Exposed areas, ft ²	10,000
Steel thickness, ft	0.03125

The total area of sinks b, c, and d is calculated from Figure 4-14 of BAW-10104 using the (5 percent margin factor not applied) building free volume. The surface area of slab d was obtained by selecting the maximum stainless steel area from a survey of six Category 1 plants.

The unpainted steel is obtained by applying the same percentage used in BAW-10103 and is given in slab c. The area of the painted steel is found by subtracting the areas of sinks c and d from the total area found from Figure 4-14 of BAW-10104. Appropriate stainless steel properties are used for slab d.

(e) Internal concrete:

Exposed area, ft ²	160,000
Paint thickness, ft	0.00083
Concrete thickness, ft	1.0

(8) The following thermophysical properties are used.

<u>Material</u>	<u>Thermal Conductivity</u> BTU/h-ft ² - °F	<u>Heat Capacity,</u> BTU/ft ² - °F
Concrete	0.92	22.62
Steel	27.0	58.8
Stainless Steel	9.1836	54.263
Paint	0.6215	40.42

(9) The condensing heat transfer coefficients given in Section 4.3.6.1 of BAW-10104 are used:

(a) At the end of blowdown, assume a maximum heat transfer coefficient four times higher than that calculated by Aerojets' Tagami correlation:

$$h_{\max} = 72.5 (Q/Vt_p)^{0.62}$$

where h_{\max} = maximum heat transfer coefficient,
BTU/h-ft² - F,

Q = primary coolant energy deposit to
containment at end of blowdown, BTU,

V = net free containment volume, ft³,

t_p = time interval to end of blowdown, s.

Before the end of blowdown, assume a linear increase from

$$h_{\text{initial}} = 8 \text{ BTU/h-ft}^2 \text{ } ^\circ\text{F}$$

to the peak value specified above.

(b) During the long-term stagnation phase of the accident, characterized by low turbulence in the containment atmosphere, assume condensing heat transfer coefficients equal to 1.2 times the one obtained from the Uchida data. The Uchida heat transfer coefficients are shown in Table A-1 of BAW-10095.

(c) During the transition in phase of the accident between the end of blowdown and the long-term, post-blowdown phase, a reasonably conservative exponential transition with a decay constant of 0.0255 is used.

3.4 THETA 1-B

Section 4.5 of BAW-10103 gives the input description for the generic analysis. The writeup given in that section remains applicable with the exception of the quoted TAFY⁷ parameters. The revised parameters specific to the Oconee I, Batch 4 fuel are shown below.

Fuel Parameters-THETA Input

Fuel Diameter, in	0.370753 ^(a)
Cladding ID, in	0.377673 ^(a)
Cladding OD, in	0.430140
Cladding Surface Roughness, in	0.00003
Fuel Surface Roughness, in	0.00007
Poisson's Ratio	0.35
Gas Composition, moles	
Air	0.0027555
Helium	0.04681285
Krypton	0.00000041
Xenon	0.00000041

(a) 18 kw/ft linear heat rate

3.5 Single Failure

The single failure used in this analysis is identical to that presented in Section 4.6 of BAW-10103.

4. LOCA Limits

4.1 Analytical Method

The allowable linear heat rate (kw/ft) limits as a function of elevation within the core are developed using the method and input assumptions outlined in Section 4 of BAW-10104 and Section 3 of this attachment. The results of the sensitivity analysis performed in Section 5 of BAW-10103 are used as outlined in Section 7.1 of that report. Since BAW-10103 was a generic report applicable to all Category 1 plants, the results of the sensitivity studies in Section 5 remain valid for the Oconee I NSS.

The LOCA limits analysis is performed by analyzing the worst break with different axial power distributions. In Section 6 of BAW-10103, the worst break for the Category 1 plants is shown to be the 8.55 ft² DE break at the pump discharge with a $C_D = 1.0$. The power shape is varied in CRAFT so that the maximum axial peaking factor (1.7) is located at the point of interest. The power shapes used are shown in Figure 7-1 of BAW-10103. Separate CRAFT analyses are run for each power shape studied. The peak linear heat rates analyzed in CRAFT are greater than or equal to the allowable linear heat rates that are developed. This approach is conservative as noted in Section 7.1 of BAW-10091⁸.

Using the CRAFT predictions at the end of blowdown, separate REFLOOD runs are made for each axial location studied. The resultant flooding rates are used to develop heat transfer coefficients during the reflood portion of the accident. For the peak power locations at or below 6 feet, the modified FLECHT correlation is used as discussed in Section 4.3.6.5 of BAW-10104. Above 6 feet, the REFLECHT⁹ correlation is used.

THETA is run for each axial location using a model representative of the hot spot modeled in CRAFT. The power level in THETA is lowered from the CRAFT peak power level until the criteria of 10CFR50.46 are met.

4.2 Allowable kw/ft

Figure 1 shows the results of the LOCA limits analysis for the Oconee I fuel. For elevations below 6 feet, specific limits based on Oconee I, Batch 4 fuel were generated. For elevations 6 feet and above, the generic limits are used. This is conservative because, as shown on Figure 1, the specific analysis would generate higher limits.

Figures 2 through 9 show the results of the LOCA limits analysis at the 2 and 4 foot elevations. For elevations above 4 feet, the parameters of interest for the LOCA limits analysis is presented in Figures 7-10 and 7-21 of BAW-10103. The results of the LOCA limits analysis are tabled below.

	<u>Core Elevation, Ft</u>				
Allowable Peak Linear Heat Rate, kw/ft	<u>2</u> 16.0	<u>4</u> 17.5	<u>6*</u> 18.0	<u>8*</u> 17.0	<u>10*</u> 16.0
Peak Cladding temp of unrupt node,/time,F/s	1930/43.5	1978/43.0	2146/61.0	2110/132.0	1931/135.0
Peak cladding temp of rupt node/Time,F/s	1882/43	1975/43	2066/45.0	1743/68.0	1642/45.0
Initial Pin Pressure PSIA	1910	1725	1300	1450	760
Rupture Time, s	10.90	12.39	15.55	15.01	39.2
Local Metal-Water Reaction, %	3.40	3.17	5.46	5.19	2.93

*Numbers from Table 7-1 of BAW-10103

5. Compliance to 10CFR50.46

The analysis presented in this attachment and the pertinent sections of BAW-10103 demonstrate compliance to the five requirements of 10CFR50.46:

- (a) Peak Cladding Temperature - The maximum peak cladding temperature calculated is 2146°F.
- (b) Maximum Cladding Oxidation - The maximum local cladding oxidation calculated is 5.46 percent.
- (c) Maximum Hydrogen Generation - Compliance to the 1 percent whole-core chemical reaction limit is demonstrated in Section 8 of BAW-10103, where the whole-core reaction is conservatively calculated to be 0.557 percent. This is unaltered by the specific analysis presented in Section 4 of this attachment for elevations below 6 feet. The method of calculating whole-core chemical reaction is performed at the elevation which yields the highest local metal-water reaction for a given kw/ft. This evaluation shows that only elevations of 6 feet or higher are used.
- (d) Coolable Geometry - The discussion given in Section 9 of BAW-10103 is applicable to the Oconee I NSS and shows that the core geometry will remain amenable to cooling.
- (e) Long-Term Cooling - Section 10 of BAW-10103 and Mr. A. C. Thies' letters of April 16 and May 30, 1975 in response to Mr. R. A. Purple's letter of March 14, 1975 demonstrate that long-term cooling is ensured because of the redundant pumped injection systems used on the Oconee I plant, and that precipitation of salts will not occur.

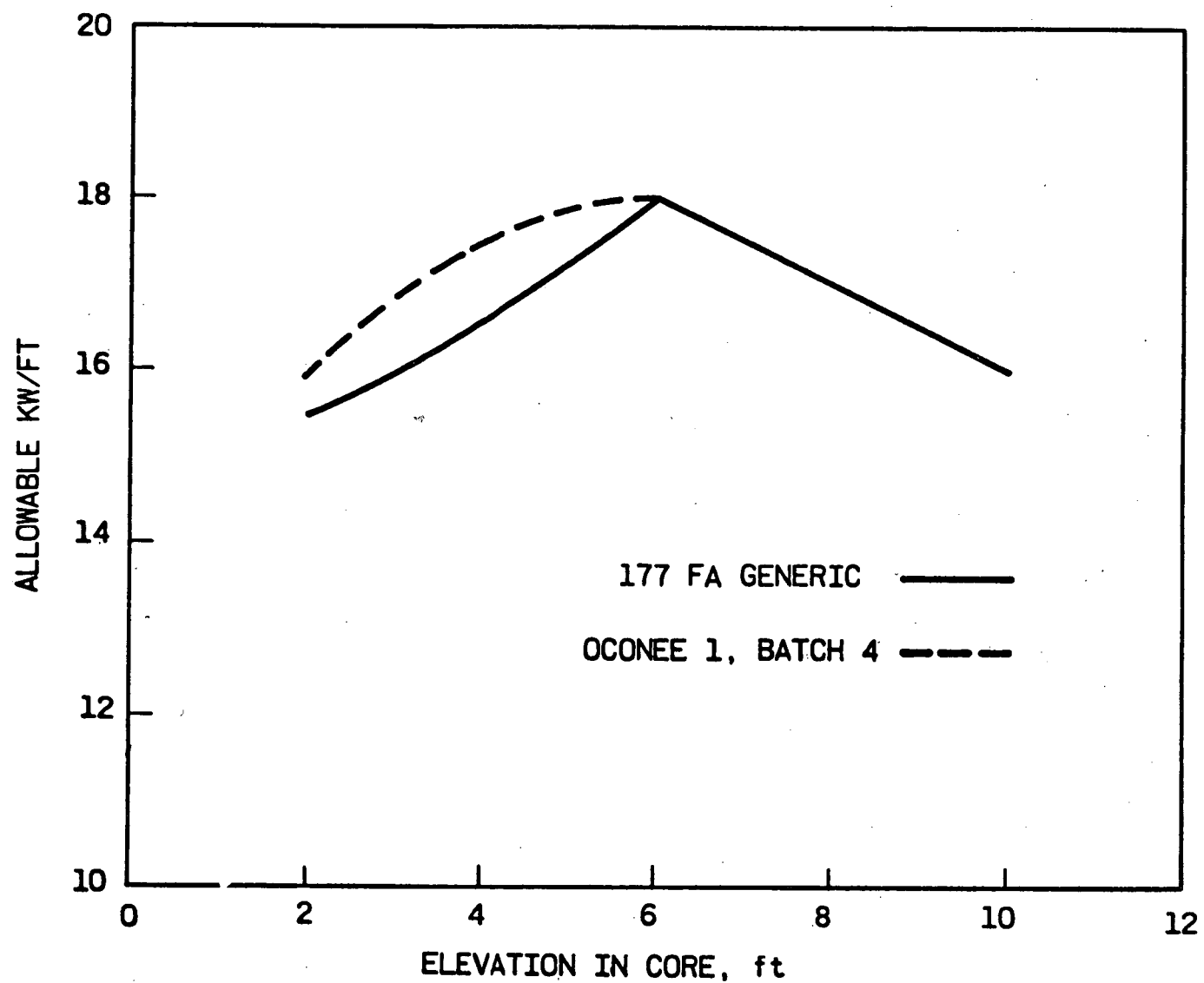
6. References

- ¹ R. C. Jones, J. R. Biller, and B. M. Dunn, ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, June, 1975.
- ² B. M. Dunn et al., B&W's ECCS Evaluation Model, BAW-10104, Babcock & Wilcox, May, 1975.
- ³ R. A. Hedrick, J. J. Cudlin, and R. C. Foltz, CRAFT 2-Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant, BAW-10092, Rev. 2, Babcock & Wilcox, April, 1975.
- ⁴ B. E. Bingham, and K. C. Shieh, REFLOOD-Description of Model for Multinode Core Reflood Analysis, BAW-10093, Babcock & Wilcox, March, 1974.
- ⁵ Y. H. Hsii, CONTEMPT-Computer Program for Predicting Containment Pressure-Temperature Response to LOCA- B&W's Revised Version of Phillips Petroleum Company Program (L. C. Richardson, et al., June, 1967), BAW-10095, Babcock & Wilcox, July, 1974.

- ⁶ R. H. Stoudt and K. C. Heck, THETA-B-Computer Code for Nuclear Reactor Core Thermal Analysis- B&W Revisions to IN-1445 (Idaho Nuclear, C. J. Hovevar and T. W. Wineinger), BAW-10094, Rev. 1, Babcock & Wilcox, April, 1975.
- ⁷ C. D. Morgan, and H. S. Kao, TAFY-Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, April, 1972.
- ⁸ B. M. Dunn, et al., B&W's ECCS Evaluation Model Report With Specific Application to 177-FA Class Plants With Lowered Loop Arrangement, BAW-10091, Babcock & Wilcox, August, 1974.
- ⁹ B. M. Dunn, et al., REFLECHT Correlation, BAW-10091P, Appendix B, Babcock & Wilcox, August, 1974, (PROPRIETARY).

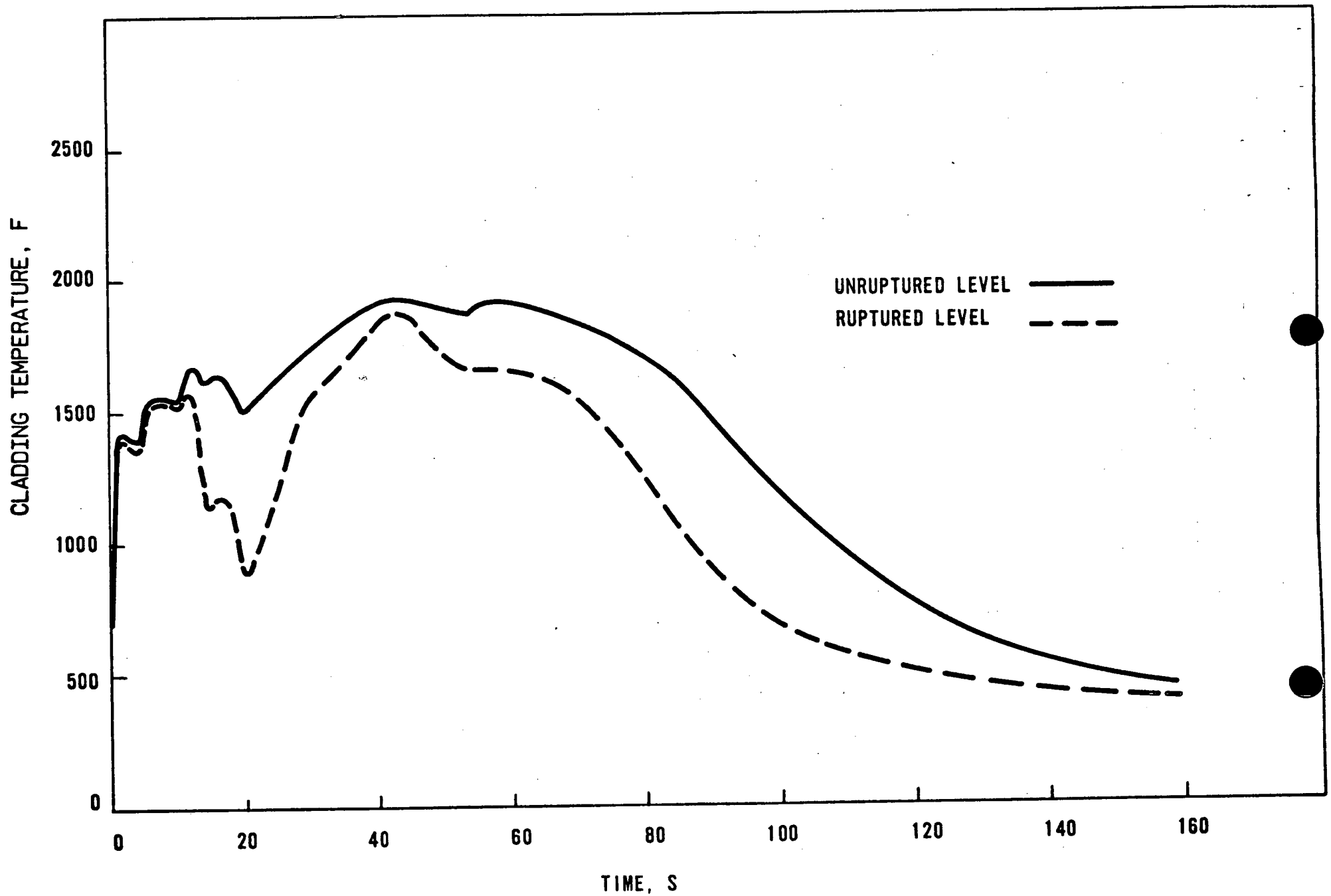
7. Computer Data on Figures

<u>Figure No.</u>	<u>Version Name</u>	<u>Version Date</u>	<u>Run Name</u>	<u>Run Date</u>
2	THETA 1B VERSION 6F	01/23/75	T1410UL	07/02/75
3	CRAFT 2 Version 5PP	04/17/75	FC141QF	06/30/75
4	REFLOOD 2 NO LOOP VERSION	12/20/74	RF1411V	07/02/75
5	THETA 1B Version 6F	01/23/75	T1410UL	07/02/75
6	THETA 1B Version 6F	01/23/75	T1430WK	07/03/75
7	CRAFT 2 Version 5PP	04/17/75	FC1431B	07/02/75
8	REFLOOD (MIMIC) Version 1	12/20/74	RF143SY	07/02/75
9	THETA 1B Version 6F	01/23/75	T1430WK	07/03/75



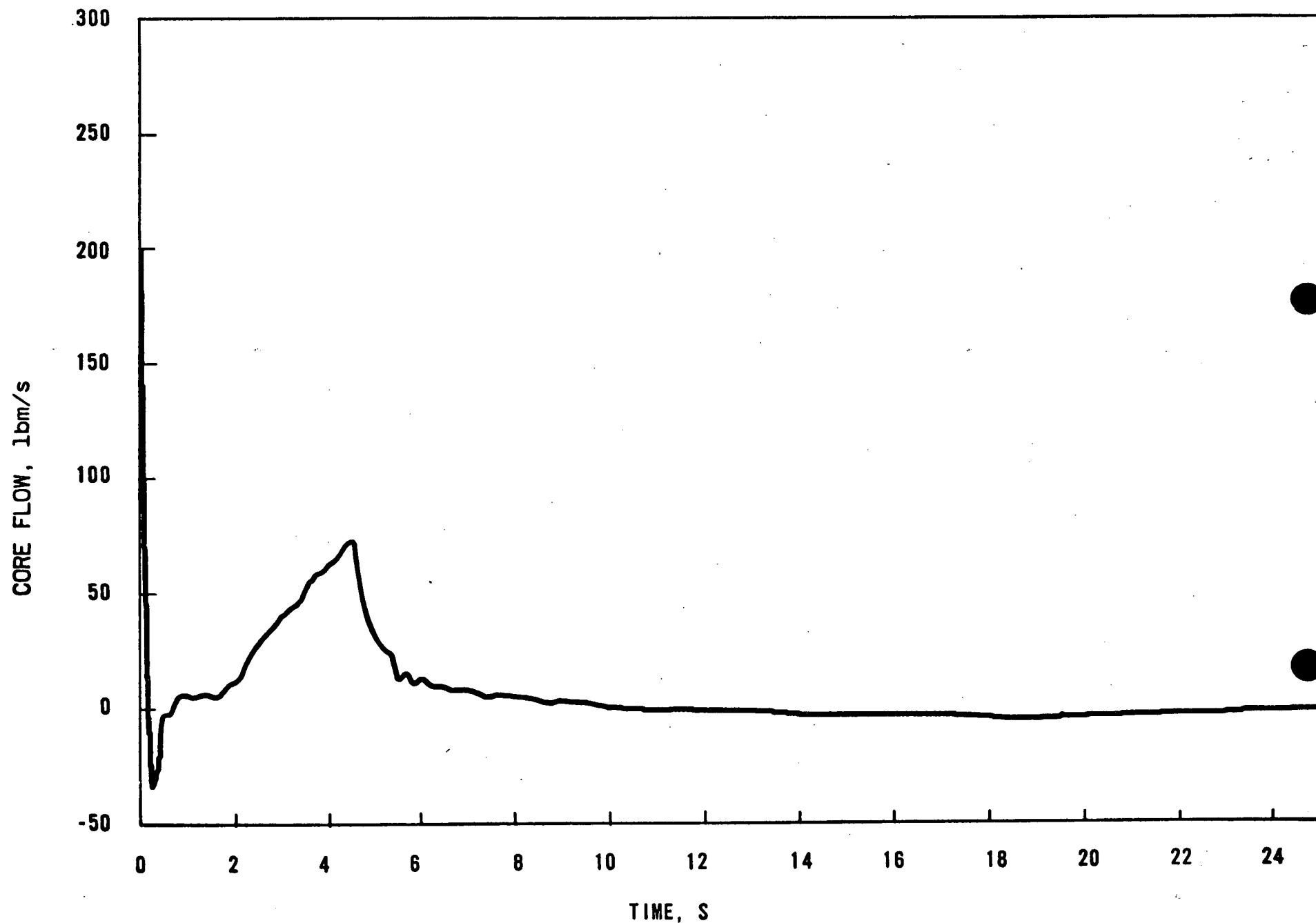
ALLOWABLE KW/FT VS ELEVATION IN CORE - 177 FA
GENERIC AND OCONEE 1, BATCH 4 RESULTS

Figure 1

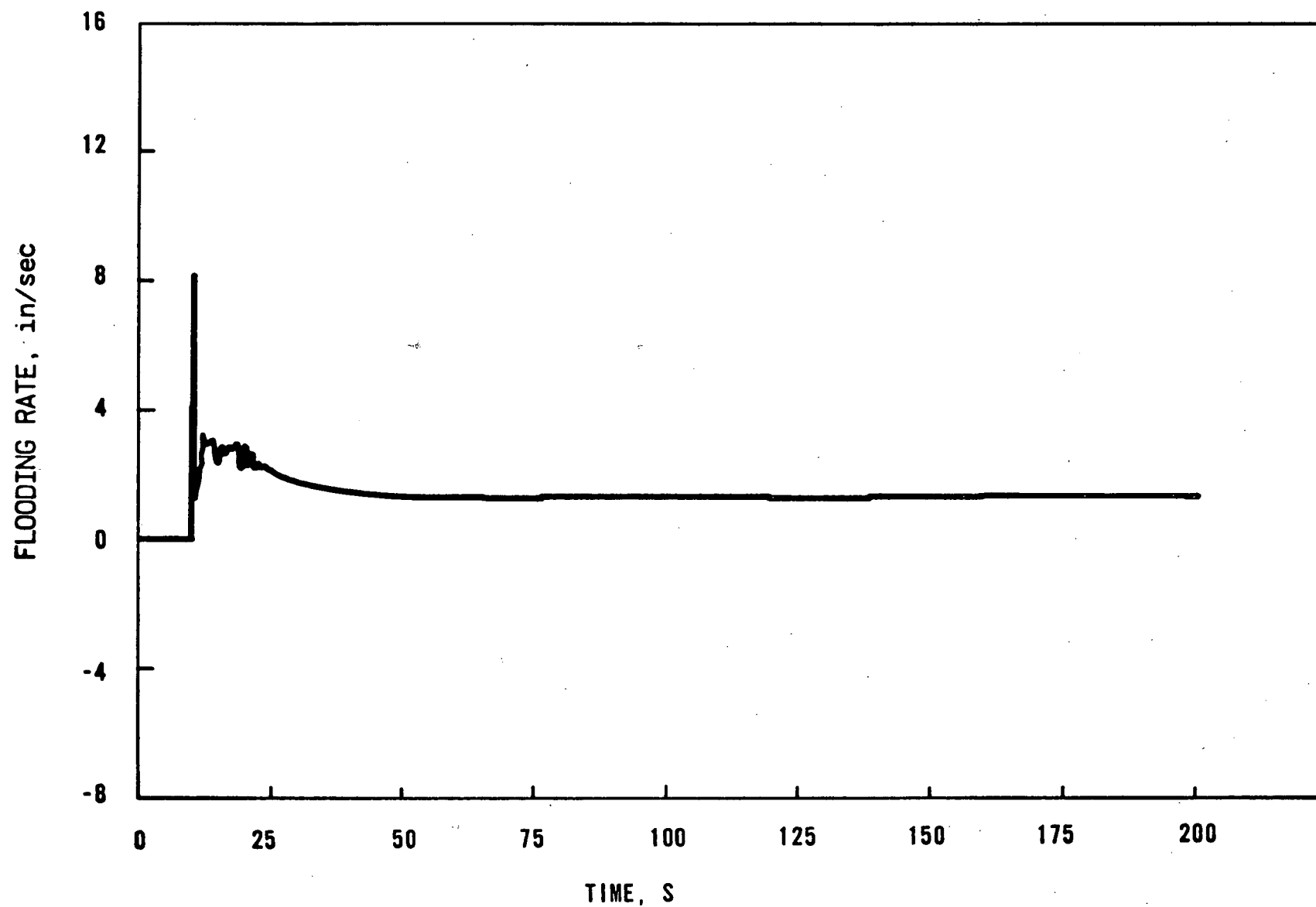


RUPTURED AND UNRUPTURED LEVEL CLADDING TEMPERATURE-LOCA LIMITS- OCONEE I,
BATCH 4- 16 KW/FT AT 2 FT ELEVATION

FIGURE 2

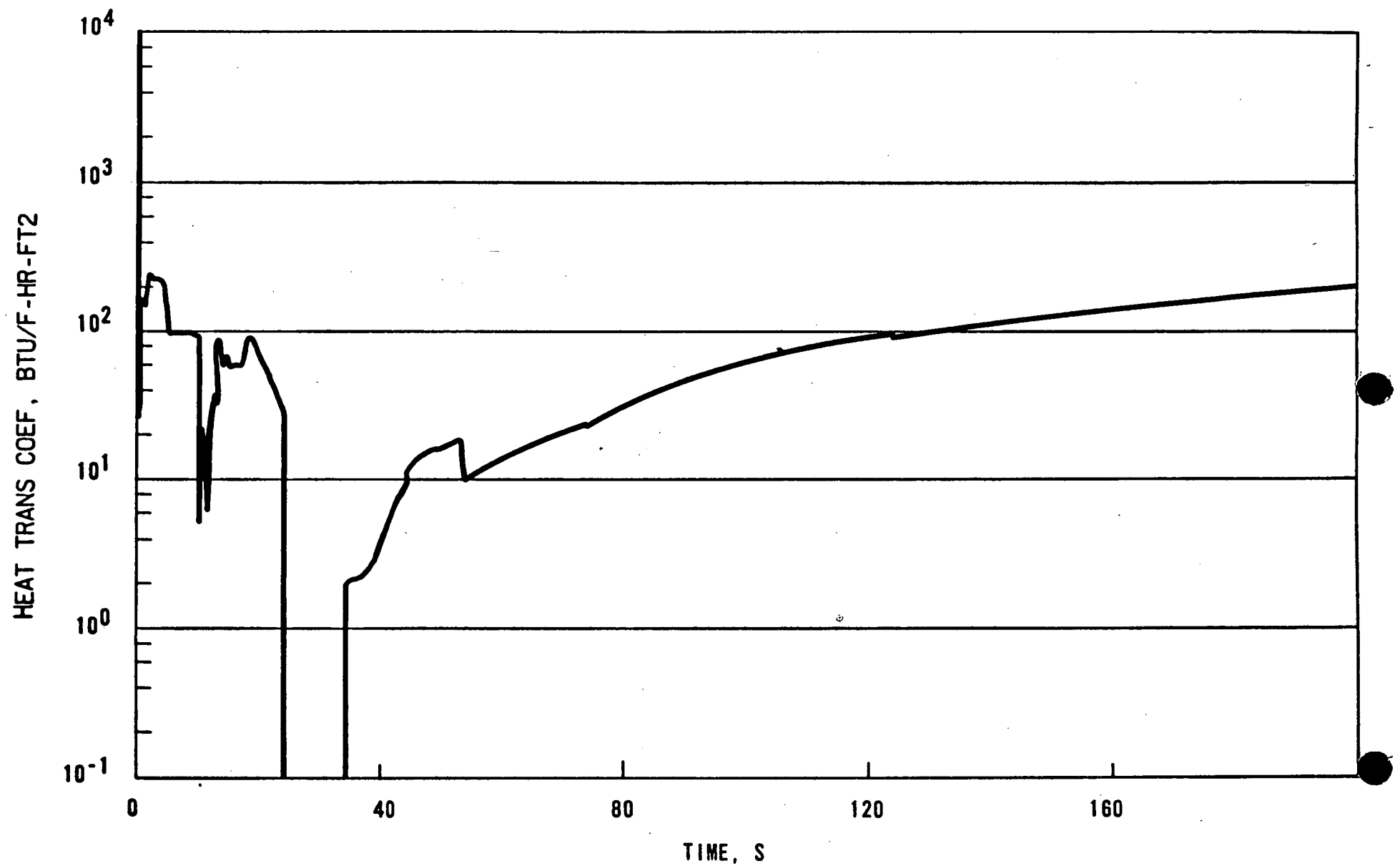


CORE FLOW-LOCA LIMITS-OCONEE I, BATCH 4-16.0 KW/FT AT 2 FT ELEVATION
FIGURE 3



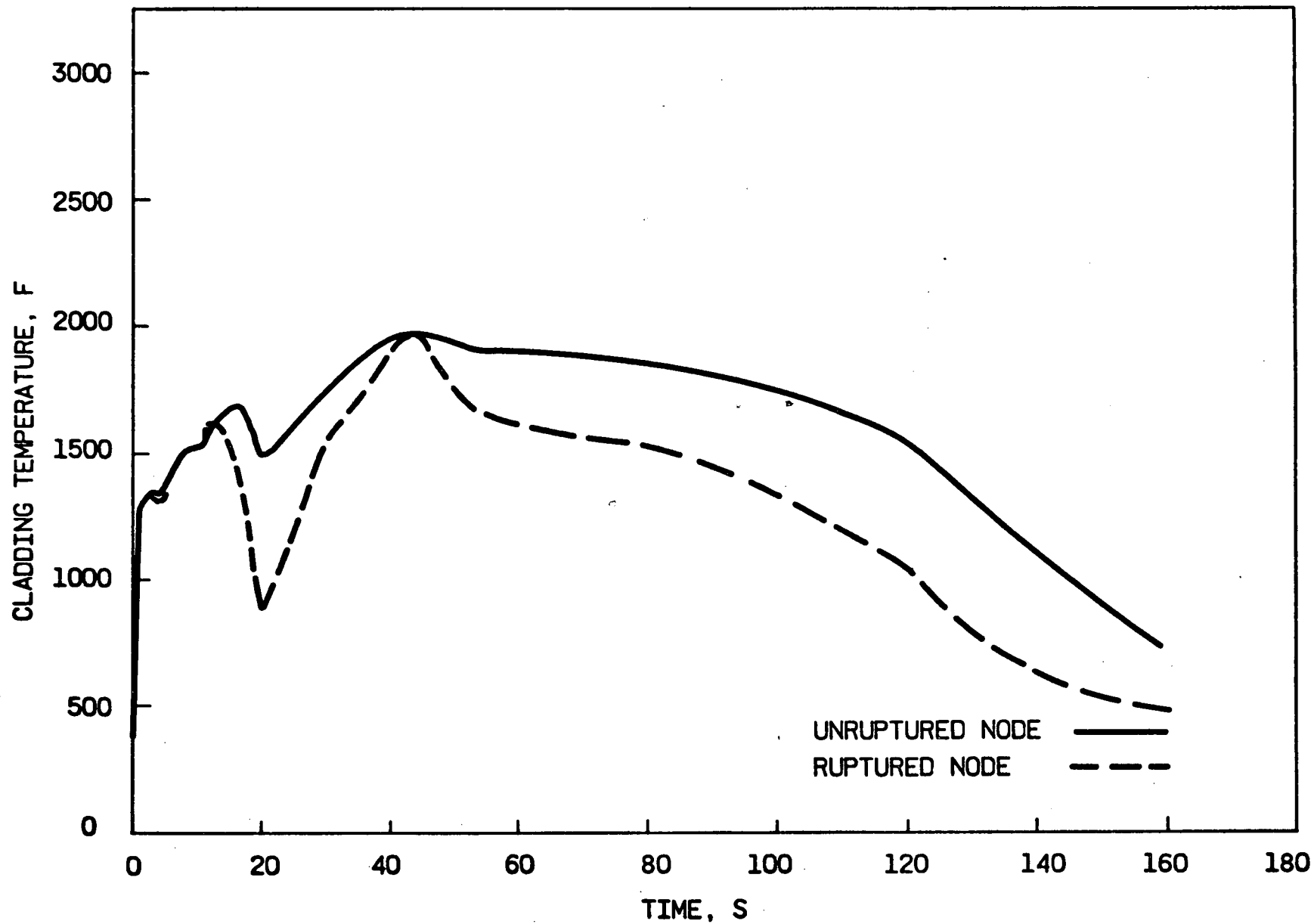
FLOODING RATES-LOCA LIMITS-OCONEE I, BATCH 4-
16.0 KW/FT AT 2 FT ELEVATION

FIGURE 4



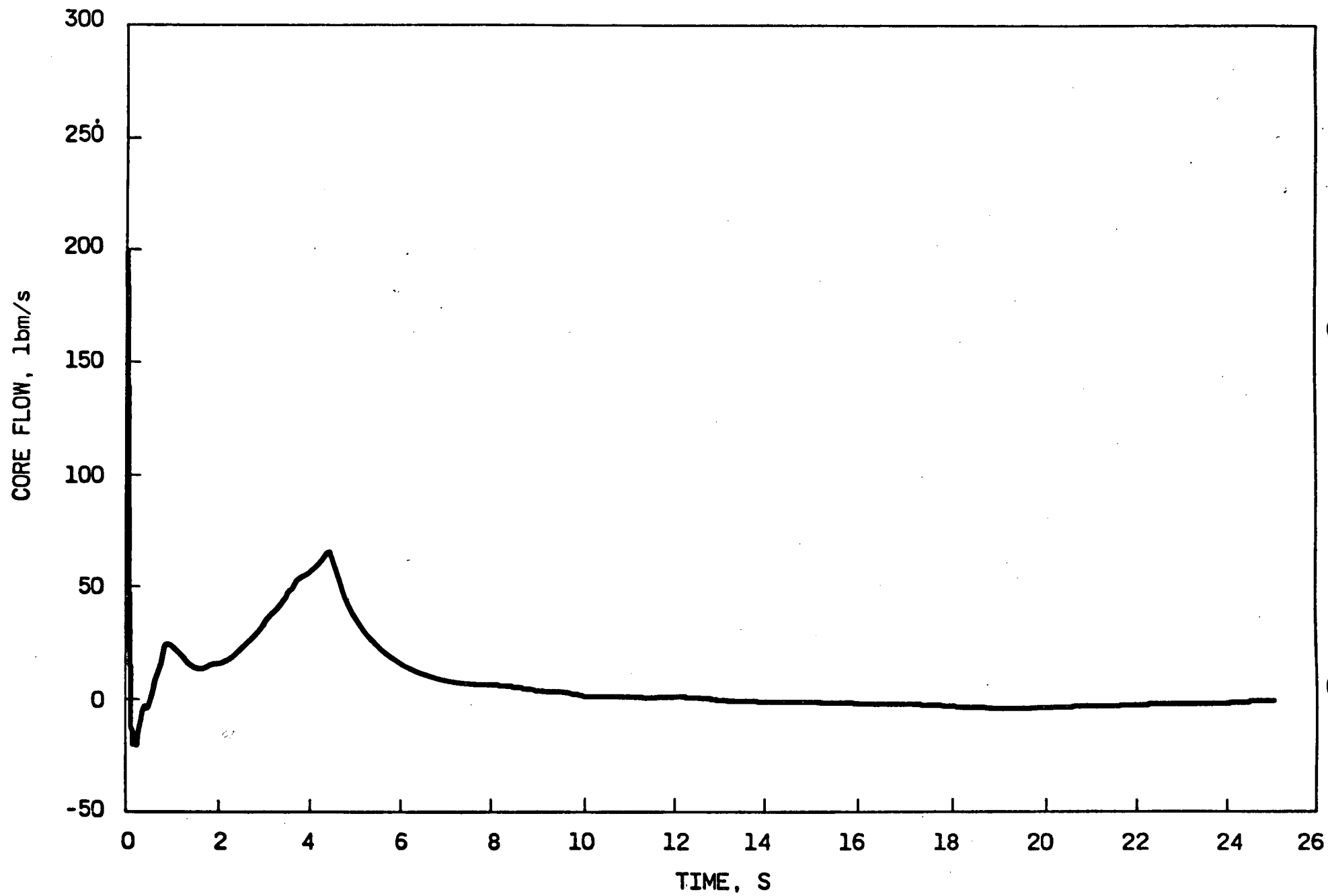
HOT SPOT HEAT TRANSFER COEFFICIENT-LOCA LIMITS-OCONEE I, BATCH 4
16.0 KW/FT AT 2 FT ELEVATION

FIGURE 5



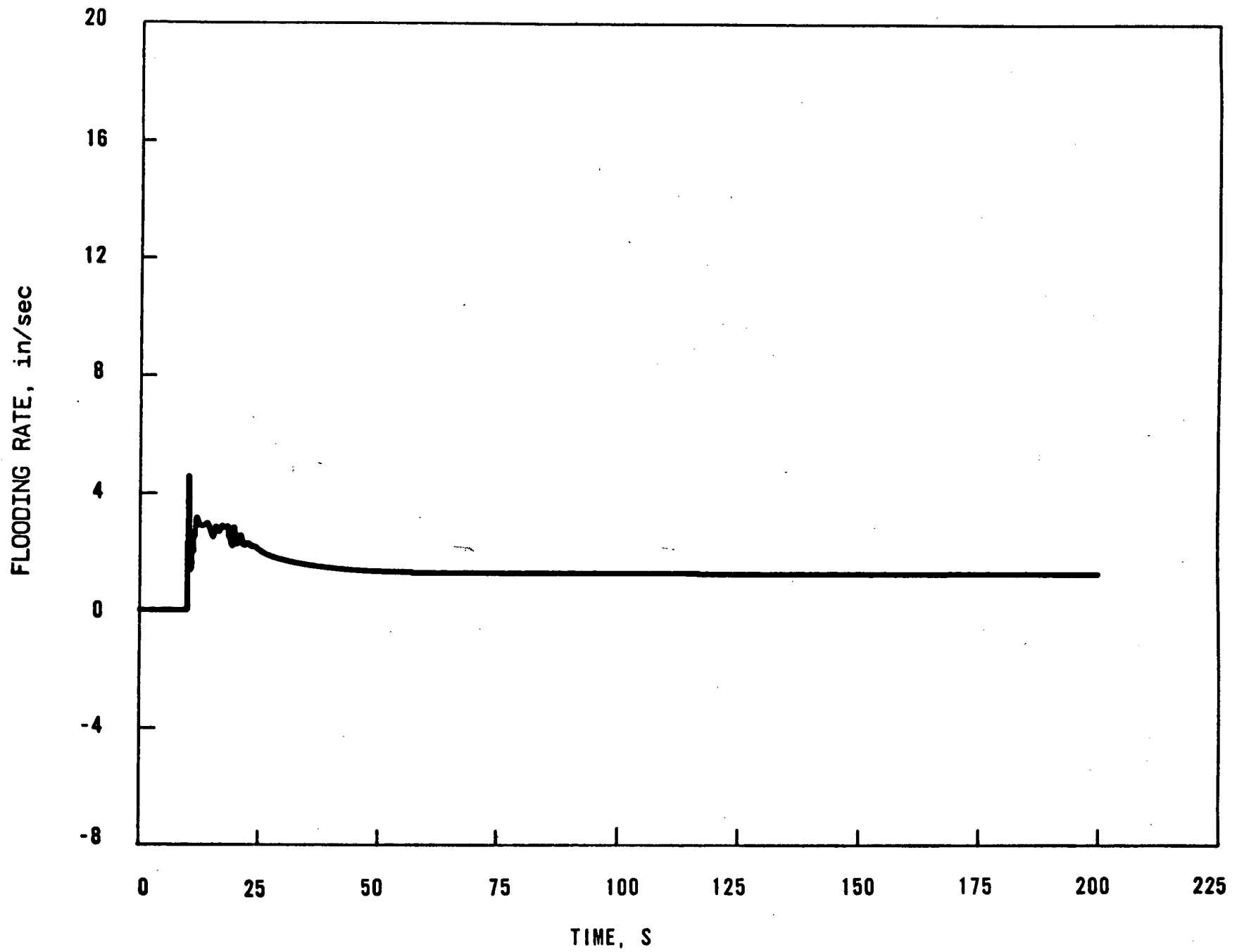
RUPTURED AND UNRUPTURED NODE
CLADDING TEMPERATURE - LOCA
LIMITS - OCONEE 1, BATCH 4-17.5
KW/FT AT 4 FT ELEVATION

Figure 6



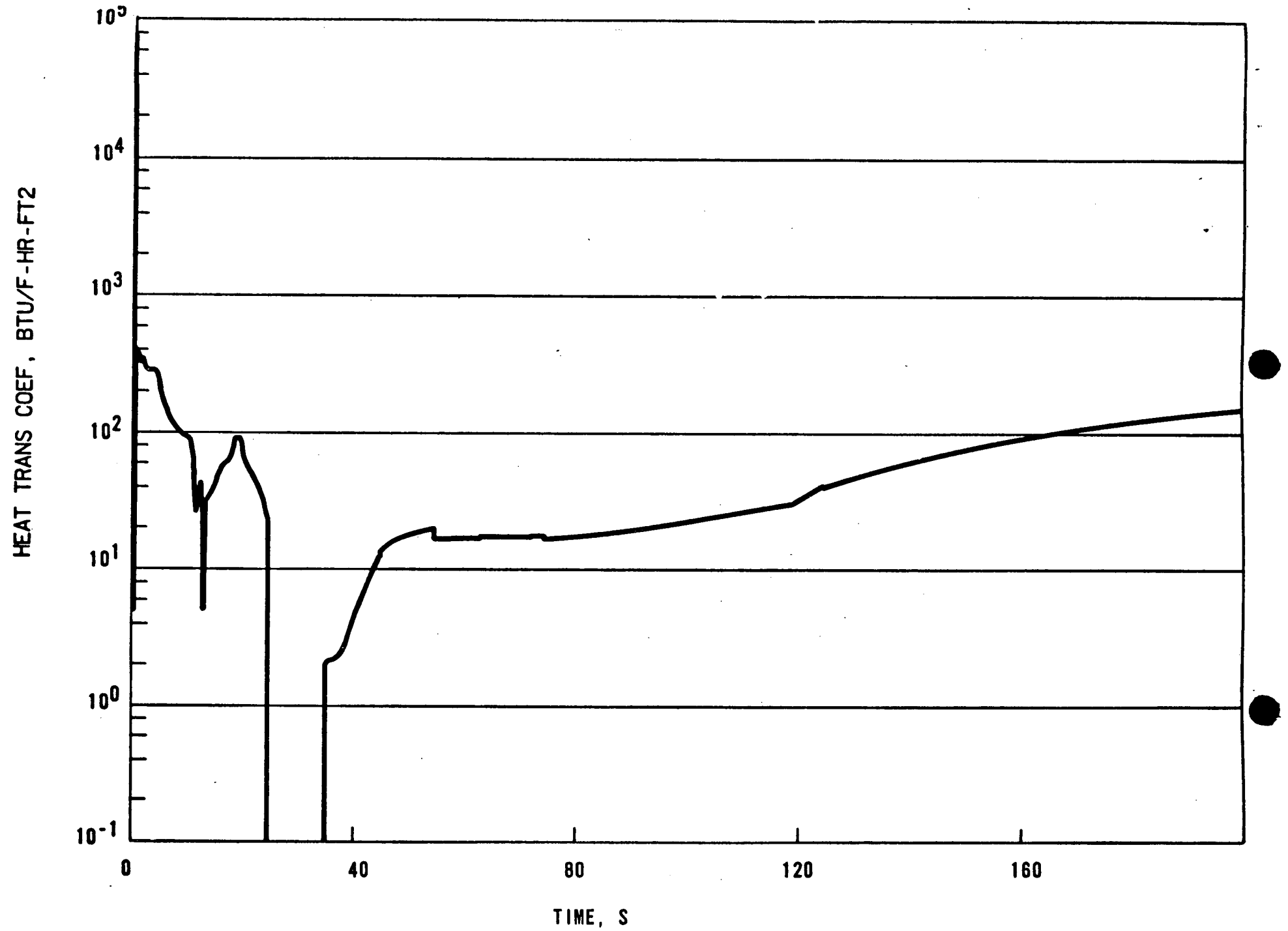
CORE FLOW - LOCA LIMITS - OCONEE 1
BATCH 4 - 17.5 KW/FT AT 4 FT ELEVATION

Figure 7



FLOODING RATES-LOCA LIMITS-OCONEE I, BATCH 4-17.5 KW/FT
AT 4 FT ELEVATION

FIGURE 8



HOT SPOT HEAT TRANSFER COEFFICIENT-LOCA LIMITS-OCONEE I, BATCH 4-
17.5 KW/FT AT 4 FT ELEVATION

FIGURE 9

Attachment 3

Manually-Controlled, Electrically-Operated ECCS Valves
Single Failure Analysis

OCONEE NUCLEAR STATION
SINGLE FAILURE ANALYSIS OF MANUALLY-CONTROLLED, ELECTRICALLY-OPERATED VALVES IN EMERGENCY CORE COOLING SYSTEM

Valve Identification		Valve Position		Evaluation
No.	Description	Normal Position	Single Failure	
A. High Pressure Injection System				
1HP-23 2HP-23 3HP-23	HP Pump Normal Suction (from LDST) " "	Open	Closes	No effect on ECCS capability. HP pump suction from LDST is not required for ECCS performance.
1HP-98 2HP-98 3HP-98	HP Pump "A" & "B" Suction Headers isolation	Open	Closes	Would prevent suction to HP pump "B." But suction to the other two HP pumps would not be affected.
1HP-115 2HP-115 3HP-115	HP Pump "A" & "B" Discharge Header isolation	Open	Closes	Would prevent HP injection by Pump "B." But HP injection by "A" and "C" would not be affected.
1HP-120 2HP-120 3HP-120	Normal RC makeup control valve in parallel with HP Injection Valve HP-26	Throttled	Will Not Close	Would prevent throttling of HP injection flow rate through "A" string. But throttling capability is not a requirement for ECCS performance.
1HP-31 2HP-31 3HP-31	RC pump seal flow control	Throttled	Will Not Close	Operability of this valve is not a requirement for ECCS performance. Also, the flow through this line is negligible compared to the available ECCS flow.
B. Low Pressure Injection System				
1LP-19 2LP-19 3LP-19 or 1LP-20 2LP-20 3LP-20	RB emergency sump isolation (Line A) RB emergency sump isolation (Line B)	Closed	1.Opens	If the valve opened immediately subsequent to a LOCA, flow through one LPI string could possibly be reduced since part of the BWST water entering that LP pump suction would be diverted to the RB emergency sump. This would have no effect on the HPI strings or on the other LPI string.

Valve Identification		Valve Position		Evaluation
No.	Description	Normal Position	Single Failure	
1LP-6 2LP-6 3LP-6 or 1LP-7 2LP-7 3LP-7	LP Pump "C" Suction "A" Header LP Pump "C" Suction "B" Header	Closed	2.fails to open after a LOCA Opens Opens	Other line admits necessary flow. No effect on LPI capability.
1LP-5 2LP-5 3LP-5 or 1LP-8 2LP-8 3LP-8	LP Pump "A" Suction LP Pump "B" Suction	Open	Closes Closes	Causes loss of suction to one LP pump. But suction to the other LP pump is not affected.
1LP-9 2LP-9 3LP-9	LP "C" pump discharge to "A" header	Closed	Opens	No effect on LPI capability.
1LP-10 2LP-10 3LP-10	LP "C" pump discharge to "B" header	Closed	Opens	No effect on LPI capability.
1LP-11 2LP-11 or 1LP-13 2LP-13	Cooler "A" inlet Cooler "B" inlet	Open	Closes Closes	Stops flow in that LPI string; but other LPI string is not affected.
1LP-12 2LP-12 or 1LP-14 2LP-14	Cooler "A" outlet Cooler "B" outlet	Open	Closes Closes	Stops flow in that LPI string; but other LPI string is not affected.

Valve Identification		Valve Position		Evaluation
No.	Description	Normal Position	Single Failure	
3LP-12 or 3LP-14	Cooler "A" outlet Cooler "B" outlet	Open Open	Closes Closes }	Would cause loss of cooling in one string. (The bypass line would provide the LPI flow rate in that string.) No effect on the other LPI string.
1LP-69 2LP-69	LPI Cooler "B" bypass	Closed	Opens	
3LP-92 or 3LP-93	LPI Cooler "A" bypass LPI Cooler "B" bypass	Closed Closed	Opens Opens }	No effect on LPI capability. If "C" pump was being used as an LP pump, would cause partial loss of cooling in "B" string.
1LP-3 2LP-3	RC return (DH line) isolation valve	Closed	Opens	
3LP-3	RC return (DH line) isolation valve	Closed	Opens	No effect on LPI capability because of the normally closed manual valve LP-4
1LP-15 2LP-15 3LP-15 or 1LP-16 2LP-16 3LP-16	LP discharge to RB spray and HP (A loop) LP discharge to RB spray and HP (B loop)	Closed Closed	Opens Opens	No effect on LPI capability.
C. Core Flooding System				
1CF-1 2CF-1 3CF-1 or 1CF-2 2CF-2 3CF-2	Tank "A" Outlet Tank "B" Outlet	Open Open	Not Credible Not Credible	Technical Specification 3.3.3(c) requires breaker for valve operator to be locked open.

Valve Identification		Valve Position		Evaluation
No.	Description	Normal Position	Single Failure	
1CF-3 2CF-3 3CF-3 or 1CF-4 2CF-4 3CF-4	Tank "A" sample and drain Tank "B" sample and drain	Closed Closed	Opens Opens	No effect on ECCS performance since there is a normally closed manual valve (CF-19) in series with the power operated valves.
1CF-5 2CF-5 3CF-5 or 1CF-6 2CF-6 3CF-6	Tank "A" vent Tank "B" vent	Closed Closed	Opens Opens	
				This line is utilized during operation for CF tank venting to the quench tank. If the valve opened during CF tank discharge, the tank discharge flow rate could be affected. However, the probability of a valve single-failure during CF tank discharge is very low due to the short time interval necessary for CF tank discharge and due to the functional use of these valves during operation.

Attachment 4

Post-LOCA Submerged Valve Motor Analysis

OCONEE NUCLEAR STATION
ANALYSIS OF VALVE MOTORS WHICH MAY BECOME SUBMERGED FOLLOWING A LOCA

<u>No.</u>	<u>Valve Identification Description</u>	<u>Evaluation</u>
1CF-1 2CF-1 3CF-1	"A" Core Flood Tank Discharge Valve	No effect on ECCS capability or containment integrity. Valve is locked open during operation and is not operated subsequent to a LOCA. Valve is not a containment isolation valve.
1CS-5* 2CS-5* 3CS-5*	Reactor Building Isolation Valve for Quench Tank Drain	No effect on ECCS capability or containment integrity. Valve is normally closed. Redundant isolation valve on outside of containment is not affected.
1HP-1 2HP-1 3HP-1	Reactor Coolant Inlet to "A" Letdown Cooler	No effect on ECCS capability or containment integrity. Letdown coolers are not used following a LOCA. Valve is not a containment isolation valve.
1HP-2 2HP-2 3HP-2	Reactor Coolant Inlet to "B" Letdown Cooler	No effect on ECCS capability or containment integrity. Letdown coolers are not used following a LOCA. Valve is not a containment isolation valve.
1HP-3* 2HP-3* 3HP-3*	Reactor Coolant Outlet from "A" Letdown Cooler and Reactor Building Isolation Valve	No effect on ECCS capability or containment integrity. Letdown coolers are not used following a LOCA. Redundant isolation valve on outside of containment is not affected.
1HP-4* 2HP-4* 3HP-4*	Reactor Coolant Outlet from "B" Letdown Cooler and Reactor Building Isolation Valve	No effect on ECCS capability or containment integrity. Letdown coolers are not used following a LOCA. Redundant isolation valve on outside of containment is not affected.
1CC-1 2CC-1 3CC-1	Component Cooling Water Inlet to "A" Letdown Coolers	No effect on ECCS capability or containment integrity. Letdown coolers are not used following a LOCA. Valve is not a containment isolation valve.
1CC-2 2CC-2 3CC-2	Component Cooling Water Inlet to "B" Letdown Coolers	No effect on ECCS capability or containment integrity. Letdown coolers are not used following a LOCA. Valve is not a containment isolation valve.

*Valve is an ES valve and would shut upon ES actuation before becoming submerged.

Attachment 5

As-Built Passive Containment Heat-Sink Data

OCONEE NUCLEAR STATION
CONTAINMENT VOLUME AND HEAT SINK DATA

<u>Parameter</u>	<u>As-Built</u>
Reactor Building Free Volume (ft ³)	1.836 x 10 ⁶
Building Cylinder	
Surface Area (ft ²)	61,353
Thickness	
Concrete (ft)	3.75 (nominal)/3.8 (equivalent)
Steel (ft)	0.0208
Paint (mils)	7-10
Mass	
Concrete (lb)	35.2 x 10 ⁶
Steel (lb)	0.59 x 10 ⁶
Building Dome	
Surface Area (ft ²)	16,230
Thickness	
Concrete (ft)	3.25
Steel (ft)	0.0208
Paint (mils)	7
Mass	
Concrete (lb)	7.91 x 10 ⁶
Steel (lb)	0.166 x 10 ⁶
Building Base	
Surface Area (ft ²)	8890
Thickness	
Concrete (ft)	8.5
Steel (ft)	0.0208
Paint (mils)	7
Mass	
Concrete (lb)	11.3 x 10 ⁶
Steel (lb)	0.091 x 10 ⁶
Internal Concrete	
Surface Area (ft ²)	66,318
Thickness	
Concrete (ft)	1.76
Paint (mils)	7-10
Mass	
Concrete (lb)	17.5 x 10 ⁶
Internal Steel (Painted)	
Surface Area (ft ²)	159,746
Thickness	
Steel (ft)	0.0316
Paint (mils)	7
Mass	
Steel (lb)	2.47 x 10 ⁶

<u>Parameter</u>	<u>As-Built</u>
Internal Steel (Unpainted)	
Surface Area (ft ²)	54,153
Thickness (ft)	0.0078
Mass (lb)	0.21 x 10 ⁶
Stainless Steel	
Surface Area (ft ²)	17,553
Thickness (ft)	0.026
Mass (lb)	0.229 x 10 ⁶

AS-BUILT HEAT SINK THERMOPHYSICAL PROPERTIES

<u>Material</u>	<u>Density</u> <u>(lb/ft³)</u>	<u>Specific</u> <u>Heat</u> <u>(BTU/lb-°F)</u>	<u>Thermal</u> <u>Conductivity</u> <u>(BTU/hr-ft²-°F)</u>
Concrete	150	0.213	1.05
Carbon Steel	490	0.12	32.0
Stainless Steel	501	0.11	9.4