ATTACHMENT 1

This listing denotes those pages to Appendix A, Technical Specifications which are revised as a result of re-evaluation of ECCS performance in conformance with the provisions of 10CFR 50.46 and of new ejected rod worth calculations.

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3.1.7. Moderator Temperature Coefficient of Reactivity

Specification

The moderator temperature coefficient shall not be positive at power levels above 95% 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 $+0.5 \times 10^{-4}$ $\Delta k/k/F$ corrected to 95% of rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ $\Delta k/k/F$.

When the hot zero-power value is corrected to obtain the hot 95% value, the following corrections will be applied.

1. Uncertainty in isothermal measurement - The measured moderator temperature coefficient will contain uncertainty owing to (a) ± 0.2 F in the ΔT of the base and perturbed conditions, and (b) uncertainty in the reactivity measurement of $\pm 0.1 \times 10^{-4}$ $\Delta k/k$.

   Proper corrections will be added for these conditions to provide a conservative moderator coefficient.

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

3. Moderator temperature change - The hot zero-power measurement must be reduced by 0.08×10^{-4} $\Delta k/k/$ F. This corrects for the difference in water temperature from zero power (532F) and 15% power (580F). Above this power, the average moderator temperature remains 580F.

4. Fuel temperature interaction (power effect) - The moderator coefficient must be adjusted to account for the interaction of an average moderator temperature with increasing fuel temperatures (as power increases). This correction is $0.0022 \times 10^{-4} \Delta\alpha m / \Delta\%$ power. It adjusts the moderator coefficient at 15% power to the coefficient at any power level above 15% coefficient to 100% power is

$$(0.0022 \times 10^{-4})(100\% - 15\%) = 0.187 \times 10^{-4}\ \Delta\alpha m.$$

6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation above 60 percent of the thermal power allowable for the reactor coolant pump combination 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 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.5.2.3 The worth 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 tilt:

1. Except for physics tests, if quadrant tilt exceeds 4%, power shall be reduced immediately to below the power level cutoff (see Figures 3.5.2-1A, 3.5.2-1B, and 3.5.21C). Moreover, the power level cutoff value shall be reduced 2% for each 1% tilt in excess of the thermal power allowable for the reactor coolant pump combination for each 1% tilt in excess of 4%.

2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than 4%, except for physics tests, or the following adjustments in setpoints and limits shall be made:

   a. The protection system maximum allowable setpoints (Figure 2.3-2) shall be reduced 2% in power for each 1% tilt.

   b. The control rod group withdrawal limits (Figures 3.5.2-1A, 3.5.2-1B, and 3.5.2-1C) shall be reduced 2% in power for each 1% tilt in excess of 4%.

   c. The operational imbalance limits (Figure 3.5.2-3) shall be reduced 2% in power for each 1% tilt in excess of 4%.

3. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.

4. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

3.5.2.5 Control rod positions:

1. Technical Specification 3.1.3.5 (safety rod withdrawal) 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.

2. Operating rod group overlap shall be 25% ± 5 between two sequential groups, except for physics tests.

3. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1A, 3.5.2-1B, and 3.5.2-1C for four pump operation and on Figure 3.5.2-2 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 be attained within four hours.

4. Except for physics tests, power shall not be increased above the power level cutoff (see Figures 3.5.2-1) unless the xenon reactivity is within 10 percent of the equilibrium value for operation at rated power and asymptotically approaching stability.

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 Figure 3.5.2-3. If the imbalance is not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four 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 superintendent.

Bases

The power-imbalance envelope defined in Figure 3.5.2-3 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 powition, 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 percent ±5 percent 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:

*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.

Group	Function
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. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistant 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 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2) 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 0.65% $\Delta k/k$ ejected rod worth at rated power.

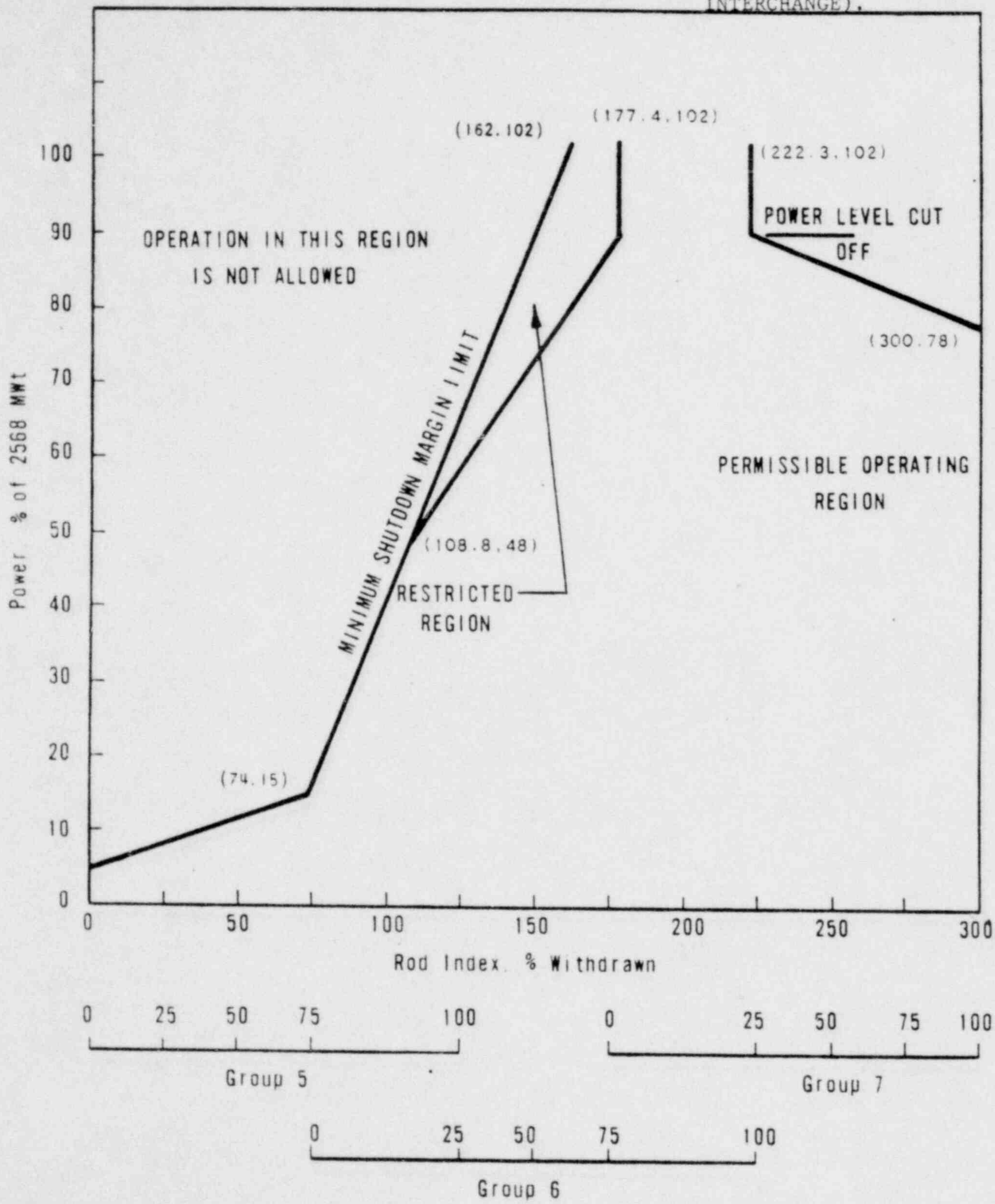
Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6, and 7 are overlapped 25%. The normal position at power is for groups 6 and 7 to be partially inserted.

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 3.5.2.5.3 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 3.5.2.4.6 and 3.5.2.5.4, respectively, will normally be performed in the plant computer. The two hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided:

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 100 EFPD TO 250±10 EFPD (PRIOR TO CONTROL ROD INTERCHANGE).



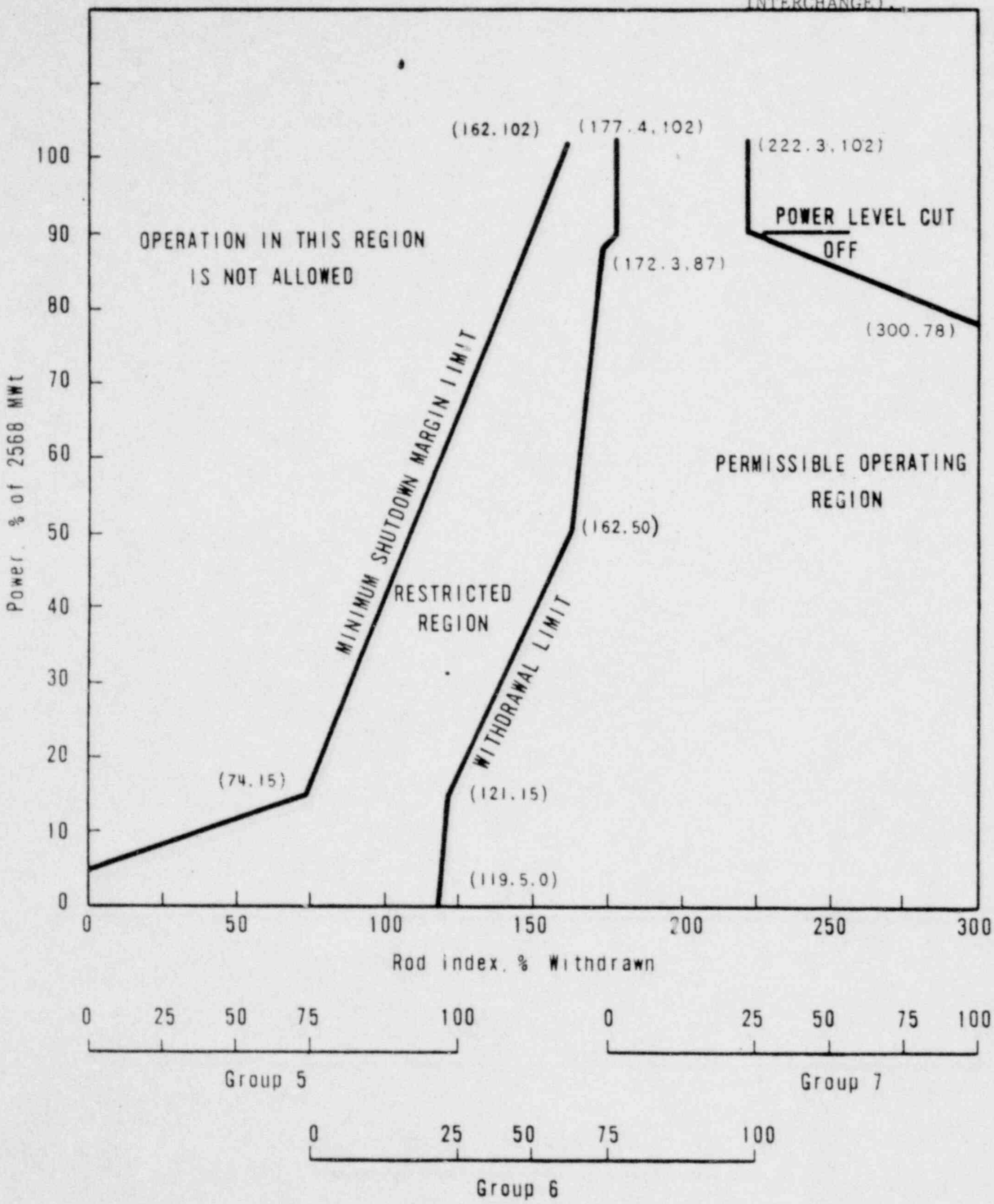
Rod index is the percentage sum of the withdrawl of Groups 5.6 and 7

UNIT 1

ROD POSITION LIMITS

Figure 3.5.2-1A

ROD POSIT.ON LIMITS FOR 4 PUMP OPERA.ION APPLICABLE DURING THE PERIOD FROM 250±10 EFPD TP 435±10 EFPD (AFTER CONTROL ROD INTERCHANGE).



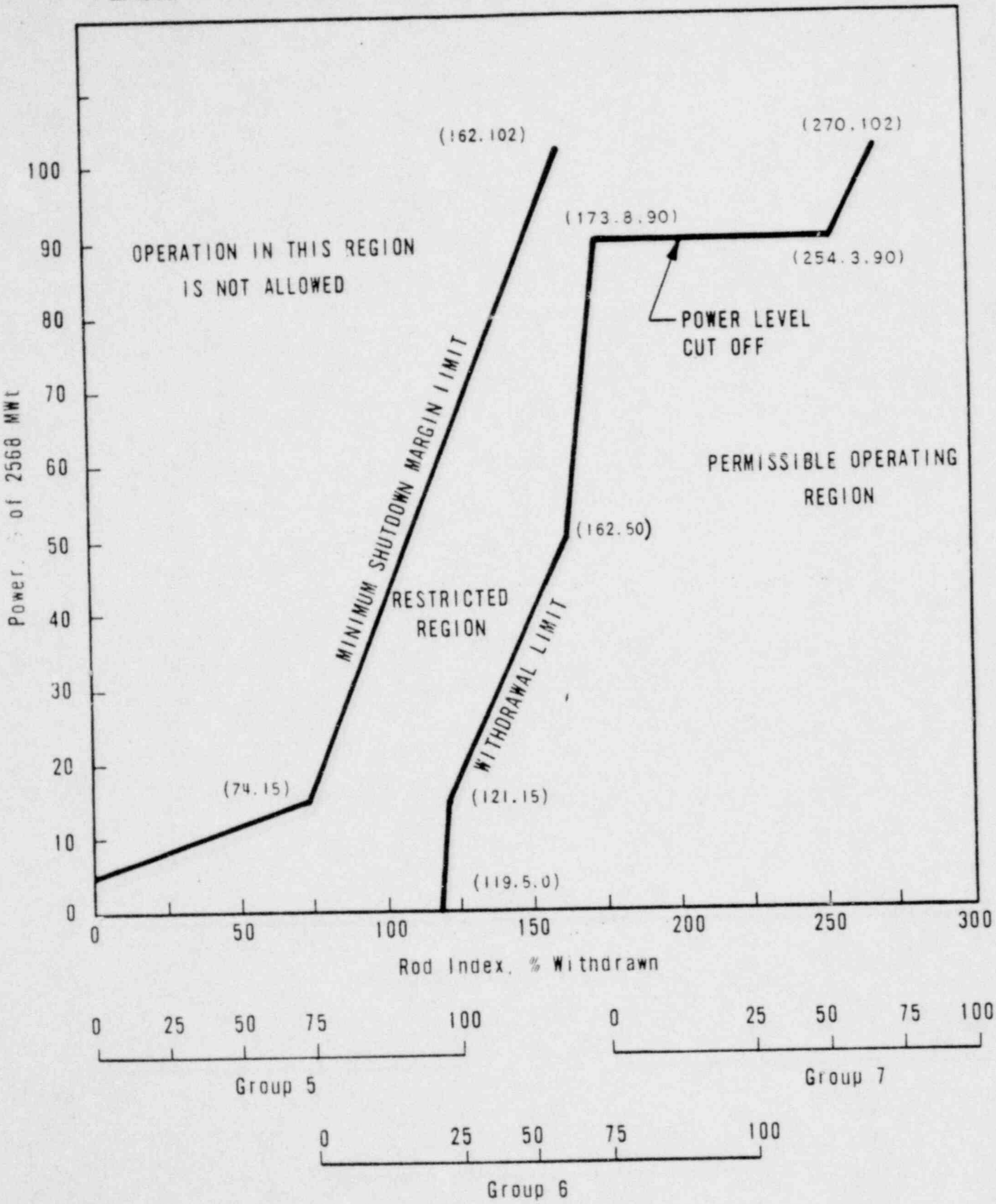
Rod index is the percentage sum of the withdrawl of Groups 5,6 and 7.

UNIT 1

ROD POSITION LIMITS

Figure 3.5.2-1B

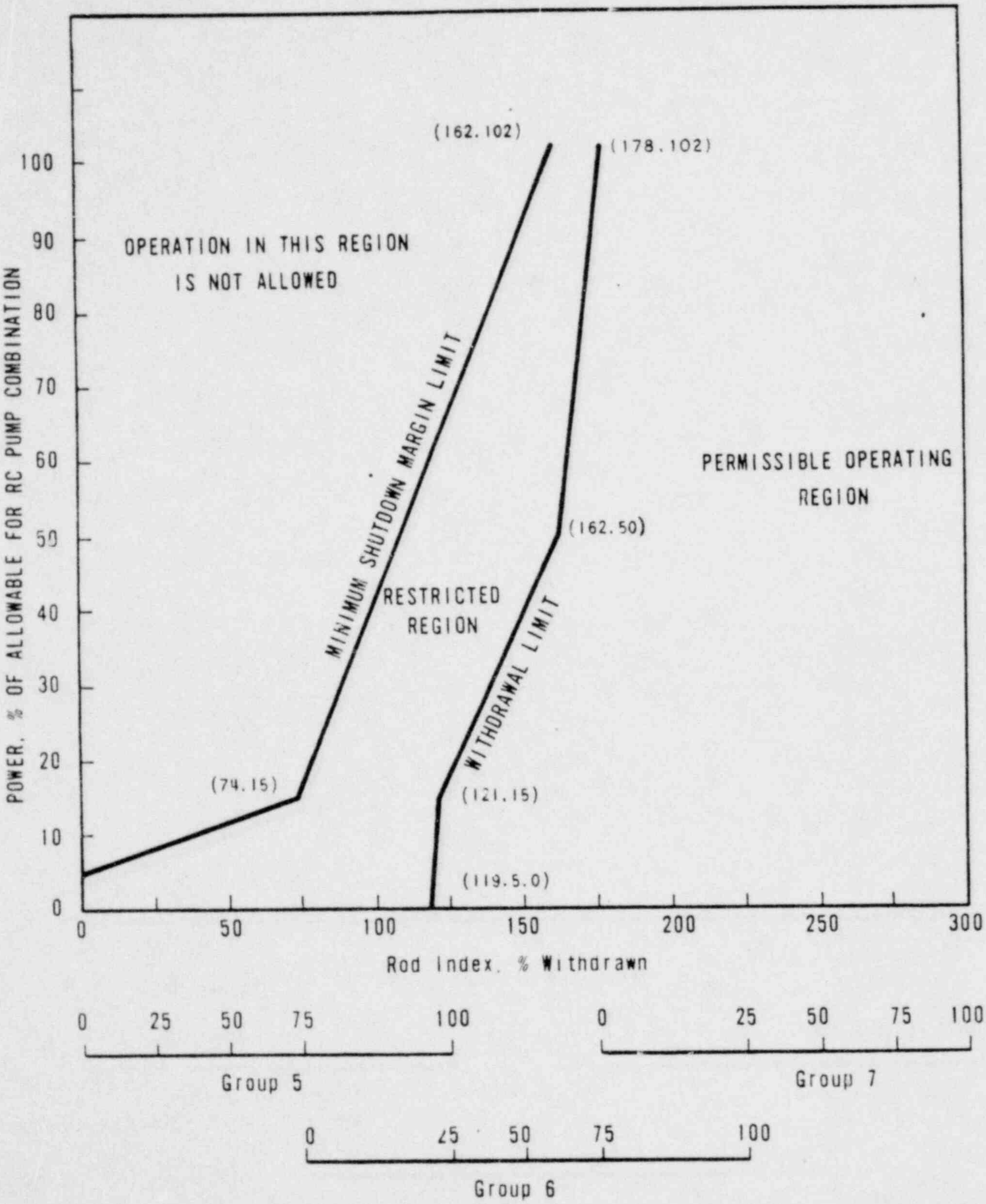
ROD POSITI LIMIT FOR 4 PUMP OPERATIC' APPLICABLE DURING THE PERIOD AFTER 435±10 EFPD.



Rod index is the percentage sum of the withdrawl of Groups 5.6 and 7

UNIT 1
ROD POSITION LIMITS
Figure 3.5.2-1C

ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION AFTER 250±10 EFPD

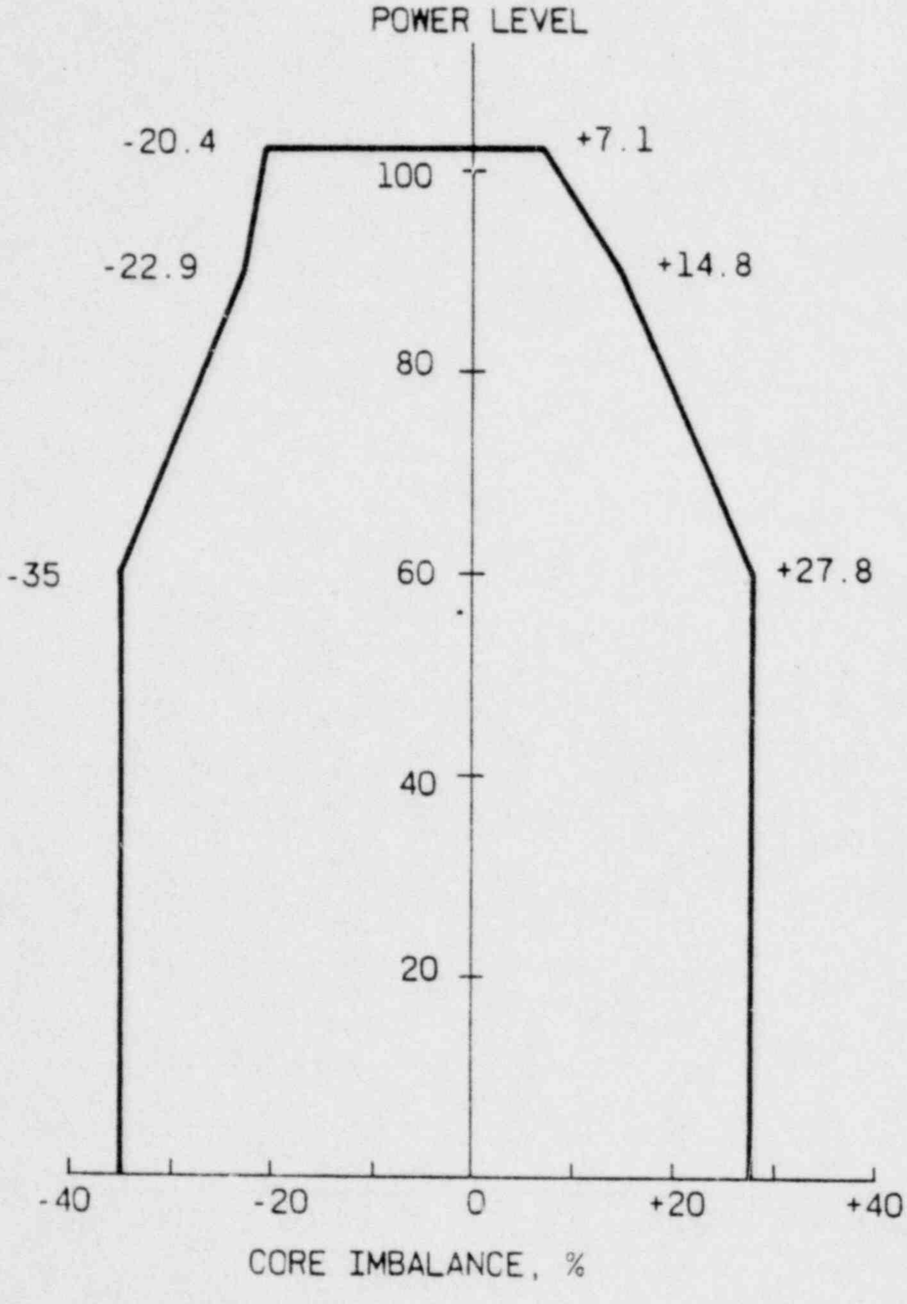


Rod index is the percentage sum of the withdrawl of Groups 5.6 and 7

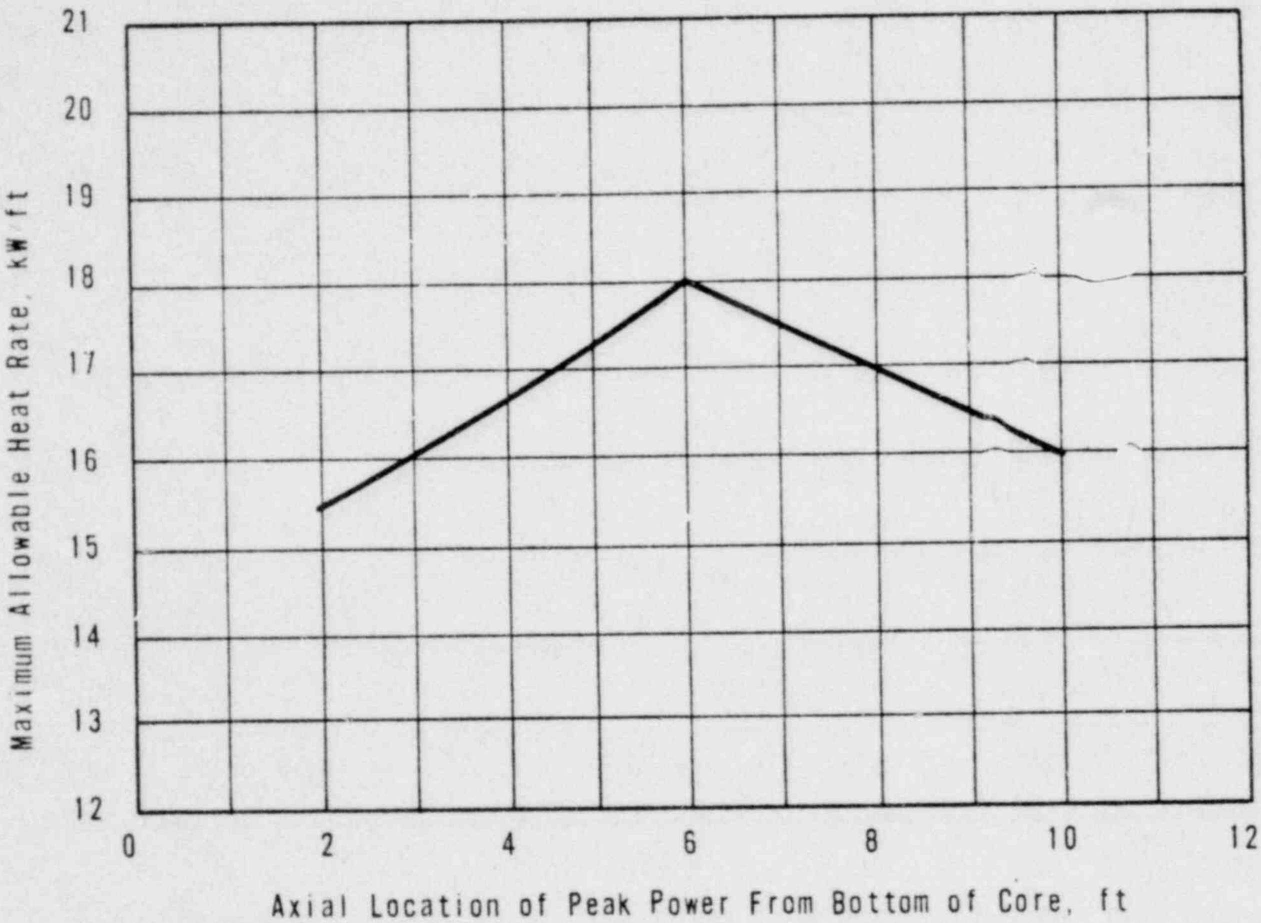
UNIT 1

ROD POSITION LIMITS

Figure 3.5.2-2



OPERATIONAL POWER
IMBALANCE ENVELOPE
Figure 3.5 2-3



LOCA LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT RATE
Figure 3.5.2-4

Regulatory Docket File

ATTACHMENT 2

7-9-75

Single Failure Analysis

Per Branch Technical Position EICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," safety systems were analysed to determine if a single failure could result in loss of capability to perform a safety function. Those systems which were reviewed for the applicability of the single failure criteria are as follows:

1) Service Water System

2) Reactor Coolant System

3) Decay Heat Removal System (Low Pressure Injection)

4) Makeup and Purification System (High Pressure Injection)

5) Reactor Building Spray System

6) Core Flooding System

These systems were first reviewed to determine those manually-controlled electrically-operated valves which were present. Upon determining these valves, analysis was performed individually to evaluate the potential consequences of these valves failing in an unsafe position. The determination yielded three valves which could potentially have serious adverse effects to safety system operation if failure occurred. These valves are as follows:

1 & 2) Core Flood Tank vent valves (CF-3A and CF-3B)

3) Service Water System common return valve to circulating water discharge flume (CV-3824)

The normal position of the Core Flood Tank (CFT) vent valves is the closed position. If single failure were to occur, either through operator error or electrical fault, the valve would fail open. If these vent valves were to open before or during CFT discharge, the tank discharge flow rate could be less than that used in the ECCS analysis. However, it has been determined also that the time it would take for the tank to vent below Technical Specification limits (575 psig) is more than enough time for the failure to be recognized and corrected. The valve is equipped with a pressure reducing orifice so that pressure would bleed off at a very slow rate. An alarm situation would occur when tank pressure reached 585 psig. This would occur within approximately 15 minutes following valve failure. It has been estimated approximately 30 total minutes would elapse before the tank pressure would reach 575 psig, allowing more than enough time for the valve to be closed manually. For this reason it is felt no modification is needed.

The closure of the common return valve to the circulating water discharge flume (CV-3824)(normally open) will result in the loss of cooling by both Low Pressure Injection (LPI) strings and to all ECCS components and provides no discharge for service water. For this reason, the breaker to this valve will be locked open and tagged during normal operation.

Based on the above analyses and the proposed corrective action, it is concluded that a single failure or operator error will not result in significantly adverse consequences to ECCS performance.

ATTACHMENT 3

7-9-75

Submerged Valves

Conservative analysis has shown that the depth of water which may accumulate in the reactor building following a LOCA will be 8'9". This yields an upper level of water in the reactor building at Elev. 345'3" (bottom of the reactor building is Elev. 336'6"). However, no motor operated valves are present between Elev. 345'3" and Elev. 357'0", thus a very conservative depth of water of 20.5 ft. may be considered.

Those motor operated valves which may become submerged f- owing a LOCA, including that corrective action or modification needed i .ne event of submergence, are as follows:

a) Core Flood Tank (T2A) Outlet Block Valves, CV-2415, Elevation 338'6". This valve is normally open and requires a key to unlock the switch which closes the valve. Following a LOCA, the core flood tank would be empty. Therefore, this valve would no longer be functionally useful for short or long term cooling or containment isolation. Thus, no modification is deemed necessary.

b) Reactor Coolant Quench Tank (T42) Discharge Isolation Valve, CV-1053, Elevation 337'3 3/4". This valve is normally closed. Following a LOCA, this valve would serve no function in short or long term cooling. This valve does receive an ES-5 signal to close and is needed for containment isolation. No modification is necessary.

c) Steam Generator Letdown Cooler Inlet Valves, CV-1213 & CV-1215, Elevation 340'2". CV-1213 is normally open and CV-1215 is normally closed. Following a LOCA, these valves would not be used for short or long term cooling. The system is isolated by the outlet valves discussed in d) below. No modification is needed.

d) Steam Generator Letdown Cooler Outlet Valves, CV-1214 & CV-1216, Elevation 338'7". These valves are normally open. Following a LOCA, these valves would not be used for short or long term cooling. These valves receive an ES-1 signal to close for system isolation. No modification is deemed necessary.

e) Intermediate Cooling Water Supply Inlet Valve to Steam Generator Letdown Coolers, CV-2216 & CV-2217, Elevation 342'4". These valves control cooling water to the Letdown Coolers and are not needed for short or long term ECCS functions or containment isolation. No modification is needed.

f) Reactor Building Sump to Decay Heat Removal System Block Valves, CV-1414 & CV-1415, Elevation 331'1 7/8" and 330'8", respectively. These valves are normally open. They are required for long term ECCS functions. The corrective action proposed is locking open the breakers to prevent closing. During shutdown, the breakers

ATTACHMENT 3

Submerged Valves

Conservative analysis has shown that the depth of water which may accumulate in the reactor building following a LOCA will be 8'9". This yields an upper level of water in the reactor building at Elev. 345'3" (bottom of the reactor building is Elev. 336'6"). However, no motor operated valves are present between Elev. 345'3" and Elev. 357'0", thus a very conservative depth of water of 20.5 ft. may be considered.

Those motor operated valves which may become submerged following a LOCA, including that corrective action or modification needed in the event of submergence, are as follows:

a) Core Flood Tank (T2A) Outlet Block Valves, CV-2415, Elevation 338'6". This valve is normally open and requires a key to unlock the switch which closes the valve. Following a LOCA, the core flood tank would be empty. Therefore, this valve would no longer be functionally useful for short or long term cooling or containment isolation. Thus, no modification is deemed necessary.

b) Reactor Coolant Quench Tank (T42) Discharge Isolation Valve, CV-1053, Elevation 337'3^3/4". This valve is normally closed. Following a LOCA, this valve would serve no function in short or long term cooling. This valve does receive an ES-5 signal to close and is needed for containment isolation. No modification is necessary.

c) Steam Generator Letdown Cooler Inlet Valves, CV-1213 & CV-1215, Elevation 340'2". CV-1213 is normally open and CV-1215 is normally closed. Following a LOCA, these valves would not be used for short or long term cooling. The system is isolated by the outlet valves discussed in d) below. No modification is needed.

d) Steam Generator Letdown Cooler Outlet Valves, CV-1214 & CV-1216, Elevation 338'7". These valves are normally open. Following a LOCA, these valves would not be used for short or long term cooling. These valves receive an ES-1 signal to close for system isolation. No modification is deemed necessary.

e) Intermediate Cooling Water Supply Inlet Valve to Steam Generator Letdown Coolers, CV-2216 & CV-2217, Elevation 342'4". These valves control cooling water to the Letdown Coolers and are not needed for short or long term ECCS functions or containment isolation. No modification is needed.

f) Reactor Building Sump to Decay Heat Removal System Block Valves, CV-1414 & CV-1415, Elevation 331'1 7/8" and 330'8", respectively. These valves are normally open. They are required for long term ECCS functions. The corrective action proposed is locking open the breakers to prevent closing. During shutdown, the breakers

ATTACHMENT 4

7-9-75

Containment Pressure

The following is a presentation of as-built passive heat sink data for Arkansas Nuclear One-Unit 1. The overriding aspect of this data which makes the Babcock & Wilcox model a very conservative model as compared to ANO-1 is the value of net free volumes used. It can be seen the generic model has nearly 400,000 ft^3 of net free volume more than ANO-1. Also, ANO-1's small heat sink area as compared to the generic model contributes to a higher containment back pressure during LOCA.

Attachment 4 (cont'd)

As-built heat sink data
for Arkansas Nuclear One-Unit 1

Net free volume $1,865,590\ ft^3$

a) Reactor Building walls including concrete wall, steel liner and anchors:

exposed area	$65,000\ ft^2$
paint thickness	0.000564 ft
steel thickness	0.0208 ft
concrete thickness	3.75 ft

b) Reactor Building dome including concrete, steel liner and anchors:

exposed area	$15,500\ ft^2$
paint thickness	0.0005 ft
steel thickness	0.0208 ft
concrete thickness	3.25 ft

c) Painted internal steel

exposed area	$92,024.00\ ft^2$
paint thickness	0.000717 ft
steel thickness	0.0312 ft

d) Unpainted internal steel (stainless and carbon)

exposed area	$84,099\ ft^2$
steel thickness	0.0111 ft

e) Internal concrete

exposed area	$94,110\ ft^2$
paint thickness	0.00169 ft
concrete thickness	2.264 ft

f) Thermodynamic Properties

Thermal Conductivity (BTU/hr-ft-F)

Material	Value
Concrete	0.9
Carbon Steel	26
Stainless Steel	10
Paints	0.083 to 1.5

Heat Capacity (BTU/ft^3-F)

Material	Value
Concrete	30
Carbon steel	56
Stainless steel	55.7
Paints	39.6 to 76.8

g) Delay times, sec.

Reactor Building Coolers (no loss of off-site power)	22 sec.
Reactor Building Sprays (no loss of off-site power)	56 sec.

h) Building initial conditions

Temperature, F	Basement 80 Dome 130
pressure, psia	16.1
relative humidity, %	basement 40 ceiling 15

i) Outside ambient temperature (summer), F

Minimum	75
Maximum	95