Three Mile Island Nuclear Plant, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 204

1.0 Proposed Technical Specification Change Request

GPU Nuclear requests that the following pages of the TMI-1 Technical Specification be replaced as indicated below:

Replace Pages: ii, vii, 3-18b, 3-18c, 3-18d, 3-18e, 3-41a, 3-41b, 4-29, 4-30, 4-31, 4-32, 4-33, 4-39, 4-42, 4-43, 4-44, 4-47, 4-58, and 4-68.

Replace Figures: 2.1-1, 2.1-2, 2.1-3, 2.3-1, 2.3-2, 3.1-1, 3.1-2, 3.1-2a, 3.1-3, 3.5-2m, 3.5-1, 3.5-2, 3.5-3, 3.11-1, 4.17-1.

Delete Pages: viii, 4-34, 4-34a, 4-34b, and 4-72.

2.0. Reason For Change

The changes included in this submittal are as follows:

2.1 CHANGES TO INCORPORATE EXEMPTION FROM THE 10 CFR 50, APPENDIX J REQUIREMENTS WHICH LINK THE SCHEDULE FREQUENCY OF INTEGRATED LEAKAGE RATE TESTS TYPE A TESTS (ILRT), WITH THAT OF THE INSERVICE INSPECTION (ISI) PROGRAM REQUIRED BY 10 CFR 50.55A.

GPUN submitted a request for exemption from the requirements of 10 CFR Part 50 Appendix J, Section III.D.1(a) on August 30, 1990. Section III.D.1(a) requires the performance of a set of three Type A tests after the preoperational leakage rate tests, at approximately equal intervals during each 10-year service period, such that the third test of each set is conducted when the plant is shutdown for the 10-year plant inservice inspections required by Section 50.55a. The exemption would allow the third Type A Containment Integrated Leak Rate Test (ILRT) of the current 10-year service period (and subsequent periods) to be uncoupled from the ISI schedule. This change is needed in order for the TMI-1 Technical Specifications to conform with the exemption granted by the NRC on February 25, 1991.

The link between ILRT and ISI schedules is not addressed in the current Bases for Specification 4.4; however, this change adds appropriate language to the Bases, describing exemption from the Appendix J requirement which connects the two schedules. In addition, paraphrased Appendix J wording is being administratively removed from Technical Specification 4.4 to eliminate redundancy in the Technical Specifications and thus improve clarity.

2.2 CHANGES TO ALLOW USE OF AN EQUIVALENT VERIFICATION METHODOLOGY FOR ES VALVE SURVEILLANCE.

GPU Nuclear requests the addition of an alternate methodology for verification of valve travel in the performance of surveillance tests prescribed in Technical Specifications 4.5.1.1.b, 4.5.1.2.b, 4.5.2.4.b, 4.5.3.1.b.2 and 4.5.3.2.b. The alternate method, which GPU Nuclear finds is an acceptable mechanism for secondary verification of valve travel during the surveillance test, would utilize control board operating lights on a different control room panel which is energized from a separate power supply and actuated by separate limit switch contacts. The plant computer uses this method for position indication, also. By use of this alternate methodology, the intent of the specification is preserved.

2.3 EDITORIAL CHANGES AND CORRECTIONS TO IMPROVE CLARITY AND CONSISTENCY, AND TO CONFORM THE ABOVE CHANGES TO THE AMENDMENT APPROVING THIS TECHNICAL SPECIFICATIONS CHANGE REQUEST (TSCR 204).

Changes in this category include the following:

a. Page numbers are being added to all 8+1/2" x 11" figures throughout the Technical Specifications. Currently no page numbers are assigned, except for Figures 3.1-2a (p 3-9b) and 3.11-1 (p 3-56b).

b. Redesignating lettered or numbered subsections with numbers or letters respectively to provide consistency of format within a given specification.

3.0 Safety Evaluation Justifying the Change

3.1 CHANGES NEEDED IN THE TMI-1 TECHNICAL SPECIFICATIONS FOR CONFORMANCE WITH THE EXEMPTION FROM REGULATORY REQUIREMENT COUPLING THE SCHEDULE FREQUENCY OF ILRT WITH THAT OF THE ISI PROGRAM.

In a letter dated February 25, 1991 the NRC approved an exemption for TMI-1 from a portion of the requirements stated in 10 CFR Part 50 Appendix J, Section III.D.1(a). This section of Appendix J requires the performance of a set of three Type A tests, at approximately equal intervals during each 10-year service period, such that the third test of each set is conducted when the plant is shutdown for the 10-year plant inservice inspections required by Section 50.55a. This exemption allows the third Type A test of the current 10-year service period (and subsequent periods) to be uncoupled from the ISI schedule.

Performance of the third Type A test and the 10-year ISI in non-concurrent outages has no effect on safety inasmuch as the purposes of the Type A tests and the 10-year ISI are independent of each other and the performance of one does not directly affect the other. The operability of the Reactor Building containment and those components which are tested in accordance with 10 CFR 50, Appendix J will continue to be verified in accordance with acceptable schedules, methods, and acceptance criteria consistent in all other respects with Appendix J except for uncoupling the test schedule from that of the ISI Program. Therefore, uncoupling the 10-year ISI and the third Type A test of a 10-year service period is justified.

3.2 CHANGES TO ALLOW USE OF AN EQUIVALENT VERIFICATION METHODOLOGY FOR ES VALVE SURVEILLANCE.

The changes to Technical Specifications 4.5.1.1.b, 4.5.1.2.b, 4.5.2.4.b, 4.5.3.1.b.2 and 4.5.3.2.b provide an additional alternative methodology for performance of the verification of valve travel during ES valve tests. The change serves to enhance safety and reduces potential operator error. Safety is enhanced because the need to attach test instrumentation to spare contacts or to have operations personnel stationed locally is eliminated. Yet the intent of the specification is preserved through the provision for a secondary means of verification of valve travel. No additional possibility of component malfunction is created by this change, and no new accident scenario is created nor existing accident scenarios impacted. This change provides for additional safety during performance of the test and eliminates the potential risk of failing to reassemble or improperly disconnecting test instrumentation previously used. Therefore, the alternative provides an overall improvement in safe plant operations.

3.3 EDITORIAL CHANGES AND CORRECTIONS:

For ease in reviewing the administrative changes proposed by TSCR No. 204, the changes being made by this request are discussed below on a page-by-page basis:

p ii — Reflects changes to pages being renumbered.

p vii — Table of Contents, page numbering added for Figures.

p viii — Deleted, original information moved to p vii.

p 3-9b — Figure 3.1-2a title retyped for clarity

p 3-18b — Pages renumbered due to Figure pagination
p 3-18c
p 3-18d
p 3-18e

p 3-41a	T.S. 3.6.8.2 references to Section 4.4.1.7.1 changed to 4.4.1.7.a.
p 3-41b	Bases; comma added in paragraph 3.
p 4-29	Original Section 4.4.1.1.3 deleted, wording in T.S. paraphrased that contained in 10 CFR 50 Appendix J sections III.A.5.a.1 and b.1, and testing at Pt is not performed at TMI-1. Old 4.4.1.1.4 renumbered as 4.4.1.1.3 and moved from p 4-30 to 4-29. New 4.4.1.1.3 revised to delete paraphrased Appendix J wording and use of the acronym ILRT to replace the words "Integrated Leak Rate Test."
p 4-30	Original Section 4.4.1.1.5 renumbered as Section 4.4.1.1.4. New wording reflects the NRC exemption granted February 25, 1991.
p 4-30	Original Section 4.4.1.1.6 renumbered as Section 4.4.1.1.5 and Appendix J paraphrased wording deleted, as well as, acceptance criteria revised to reflect high pressure test symbols of 10 CFR 50 Appendix J III.A.5.a.2 and b.2 rather than testing at Pt which is not performed at TMI-1.
p 4-30	Original Section 4.4.1.1.7 renumbered as Section 4.4.1.1.6 and Appendix J paraphrased wording (III.A.1.a) deleted, as well as, revision to reflect testing at Pa rather than at Pt which is not performed at TMI-1.
p 4-30 p 4-31	Original Section 4.4.1.1.8 renumbered as Section 4.4.1.1.7 and revised to reflect commitment to Appendix J reporting requirements.
p 4-30	Section 4.4.1.2.1 reference to 4.4.1.2.5.f revised to 4.4.1.2.5.c.
p 4-30	Sections 4.4.1.2.2 and .3 are changed to reflect use of the acronym LLRT and revision of reference to 4.4.1.2.5.b changed to 4.4.1.2.5.a. Paraphrased Appendix J wording in paragraphs b, c, and d deleted (See Appendix J III.c.1).
p 4-31	Section 4.4.1.2.5 revised to reflect use of acronym LLRT and exceptions to Appendix J frequency deleted where applicable.
p 4-31	Section 4.4.1.4 revised to reflect use of the acronym ILRT.
p 4-31	Section 4.4.1.5 reference to 4.4.1.1.6 changed to 4.4.1.1.5, and revised to reflect the use of the acronyms LLRT and ILRT.
p 4-31 p 4-32	Section 4.4.1.6 Subparagraph numbering changed to letters for consistency within Specification 4.4.
p 4-32	Section 4.4.1.7 Subparagraph numbering changed to letters for consistency within Specification 4.4.

p 4-32 p 4-33	Bases, wording added to reflect NRC exemption granted as well as use of the acronym ILRT and LLRT, new references added.
p 4-39	Sections 4.5.1.1.b and 4.5.1.2.b revised to reflect an alternate methodology for secondary verification of valve travel during the surveillance test.
p 4-42	Section 4.5.2.2.c typographical correction: effects changed to affects.
p 4-42	Section 4.5.2.4.b revised to reflect an alternate methodology for secondary verification of valve travel during the surveillance test.
p 4-42	Bases, typographical corrections: storgage changed to storage, and lines changed to line.
p 4-43	Section 4.5.3.1.b moved from p 4-44 to 4-43 and Subparagraph 2 revised to reflect use of an alternate methodology for secondary verification of valve travel during the surveillance test.
p 4-44	Section 4.5.3.2.b revised to reflect use of an alternate methodology for secondary verification of valve travel during the surveillance test.
p 4-44	Bases, wording added to clarify the flow for testing the delivery capability of the Reactor Building Spray pumps.
p 4-47	Editorial corrections of typographical errors in paragraph 4.6.3.a (1), misplaced wording relocated; and, typographical correction in paragraph 4.6.3.a (3): Verify.
p 4-58	T.S. 4.15 changes to the Main Steam ISI specification to reflect the TMI-1 inservice inspection program, applicable code requirements, and the correct weld identification numbers. In addition, the third paragraph of section 4.15.1 was deleted and the Tech. Spec. Bases expanded to include potential for future changes in the inspection interval. An appropriate reference to the UFSAR has been added.
Figures	Currently page numbers are not assigned to most figures throughout the Technical Specifications. Assignment of page numbers will clarify the intended location of the figures. This has been done for all figures except Figures 5.1, 5.2, and 5.3 which are 11" x 17" fold-outs in Section 5.0.

In summary, the changes described above are administrative in nature and are intended to make corrections that are justified and appropriate in accordance with NRC regulations. No unreviewed safety question is involved in the changes as reflected in this Technical Specification Change Request.

4.0 No Significant Hazards Considerations

GPUN has determined that this Technical Specification Change Request (TSCR) poses no significant hazards as defined by the NRC in 10 CFR 50.92. This change is considered to be administrative in nature and does not involve significant hazards consideration as evaluated below.

4.1 Operation of Three Mile Island Nuclear Station, Unit-1, in accordance with this TSCR would not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed Technical Specification change does not modify or create any accident initiating condition. This change provides administrative changes and corrections to the Technical Specifications; conforms the Technical Specifications to the NRC granted exemption from the ISI schedule requirements of 10 CFR 50.55a; and, provides additional flexibility in the performance of surveillance test confirmations of valve travel. The changes do not result in any condition contrary to the requirements of 10 CFR 50 Appendix J. Therefore, the changes do not impact on nuclear safety or safe plant operations.

4.2 Operation of Three Mile Island Nuclear Station, Unit-1, in accordance with this change would not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed Technical Specification change does not modify or create any accident initiating condition. The proposed changes will result in Technical Specification requirements that meet or exceed the requirements of 10 CFR 50 Appendix J.

4.3 Operation of Three Mile Island Nuclear Station, Unit-1, in accordance with this change would not involve a significant reduction in a margin of safety because the margins of safety, as described in existing Technical Specification bases, remain unaffected by this change request; further, the margins of safety defined in the SAR are not impacted by this change.

The Commission has provided guidelines pertaining to the application of the three standards by listing specific examples in the Federal Register (48 FR 14870). This proposed change is considered to be in the same category as examples (i), (ii), or (iv) of the "Amendments Not Likely to Involve Significant Hazards Consideration" from that listing. Thus, operation of the facility in accordance with the proposed amendment involves no significant hazards considerations.

5.0 Implementation

It is requested that the amendment authorizing this change become effective upon issuance and shall be implemented within thirty (30) days of receipt.

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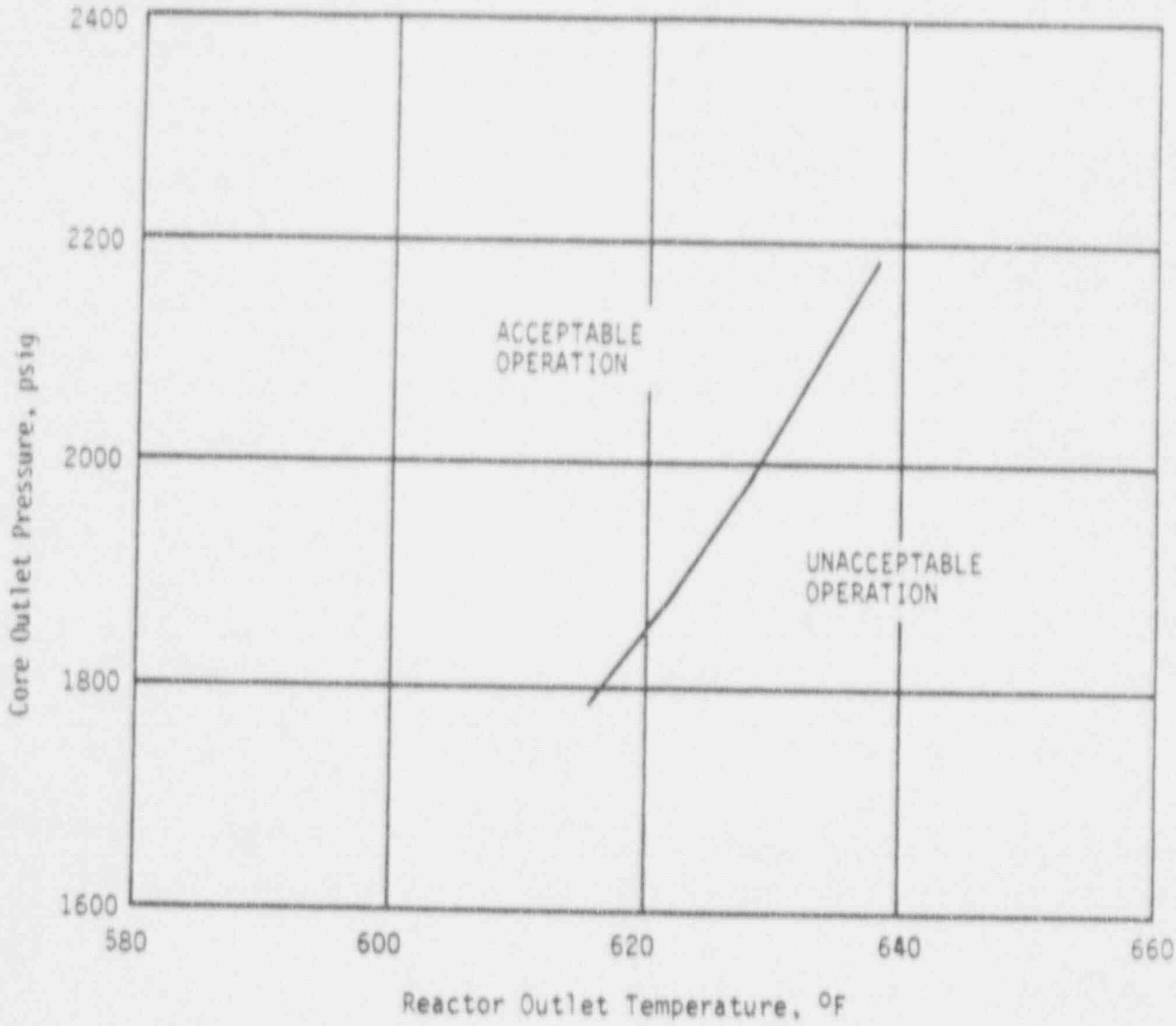
LIST OF FIGURES

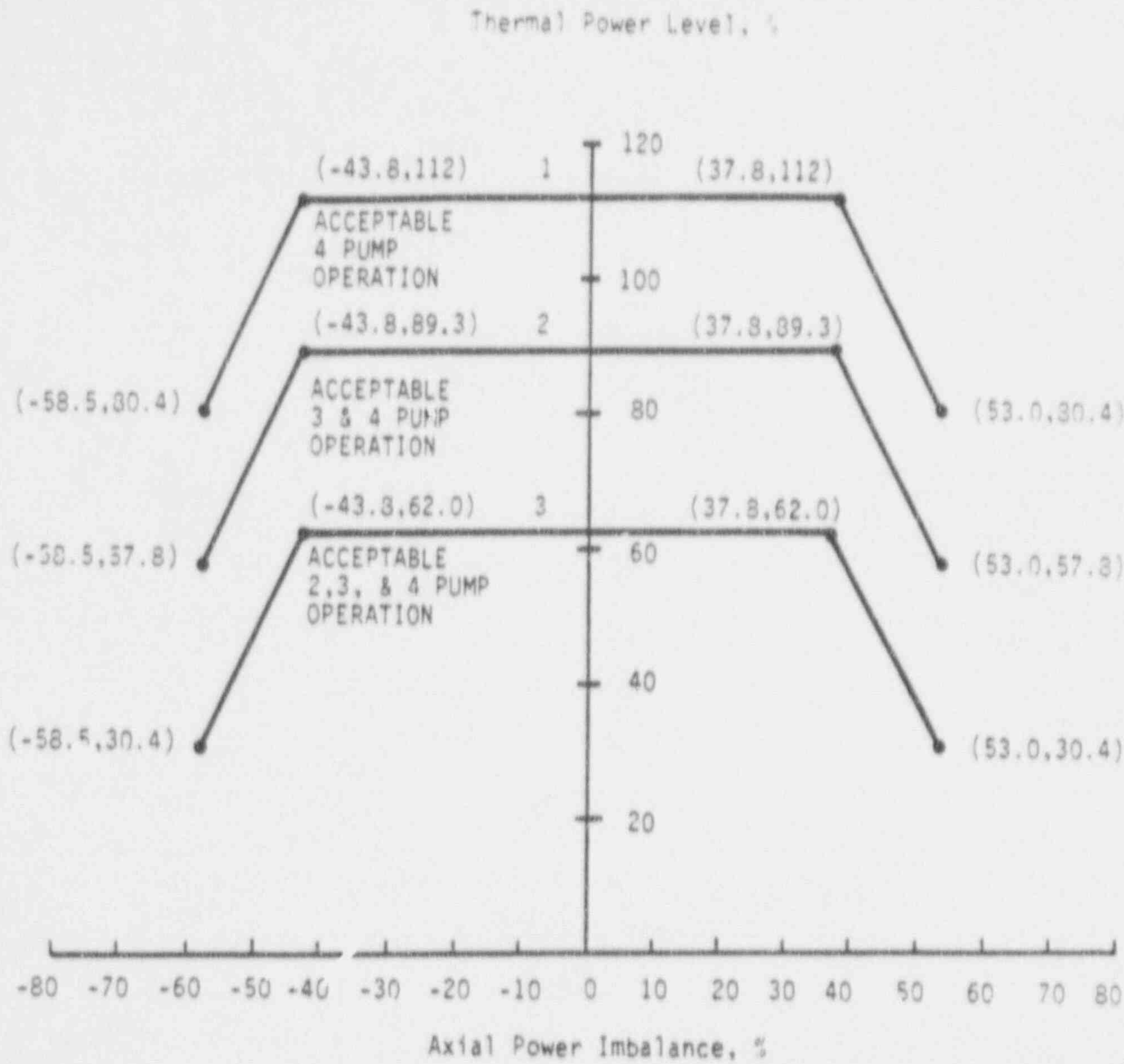
Amendment Nos. ~~11~~, ~~17~~, ~~29~~, ~~39~~, ~~45~~, ~~50~~, ~~59~~, 72, ~~106~~, ~~109~~, ~~120~~, ~~126~~, ~~134~~, ~~142~~, ~~150~~



CORE PROTECTION SAFETY LIMIT

TMI-1

Figure 2.1-1

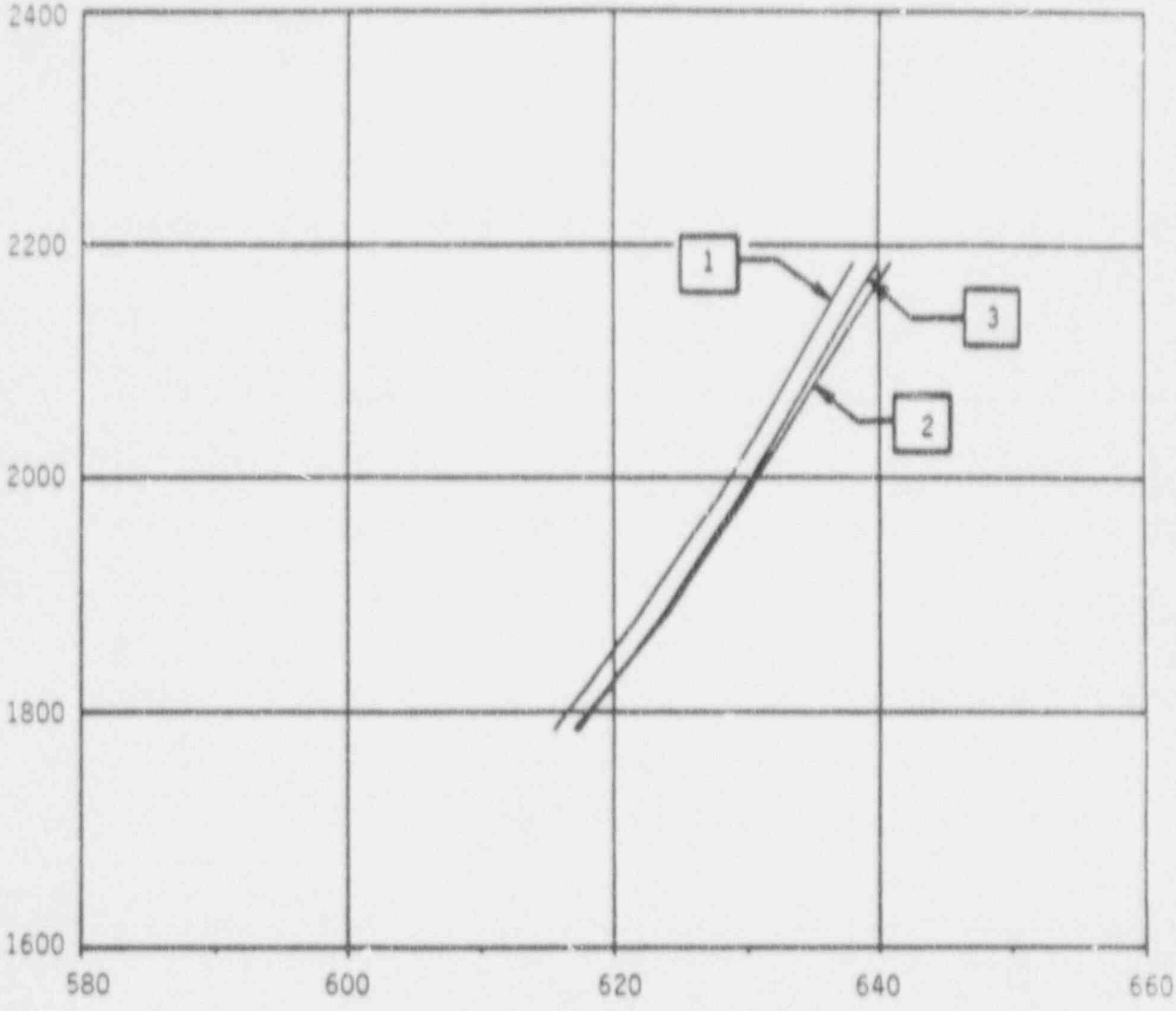


Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

CORE PROTECTION SAFETY LIMITS

TMI-1

Figure 2.1-2



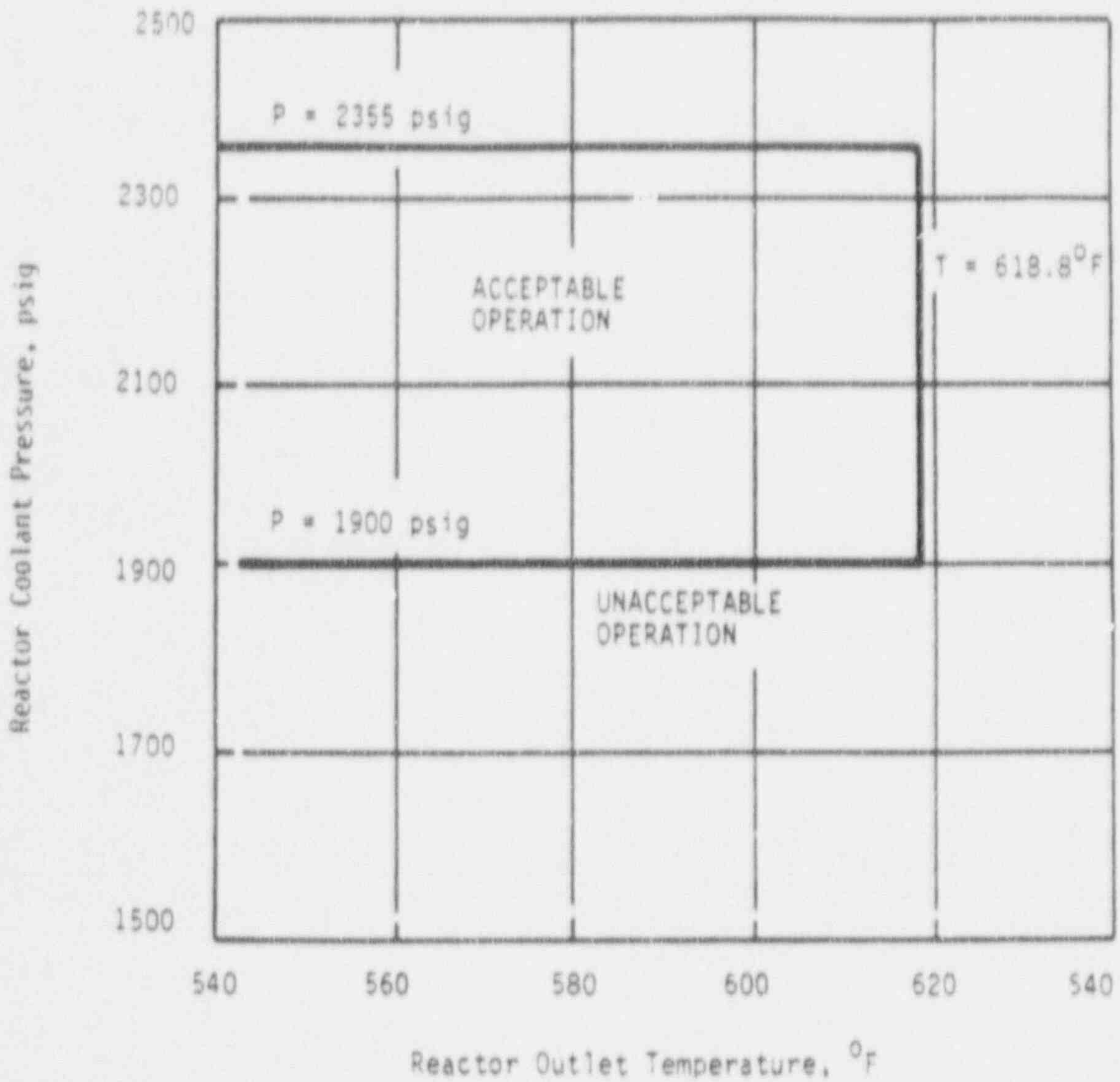
Curve	Reactor Coolant Flow (lbs/hr)	Power	Pumps Operating (Type of Limit)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	89.4%	Three Pumps (Quality Limit)
3	68.8×10^6 (49.2%)	62.0%	One Pump in Each Loop (Quality Limit)

*106.5% of Cycle 1 Design Flow

CORE PROTECTION SAFETY BASES

TMI-1

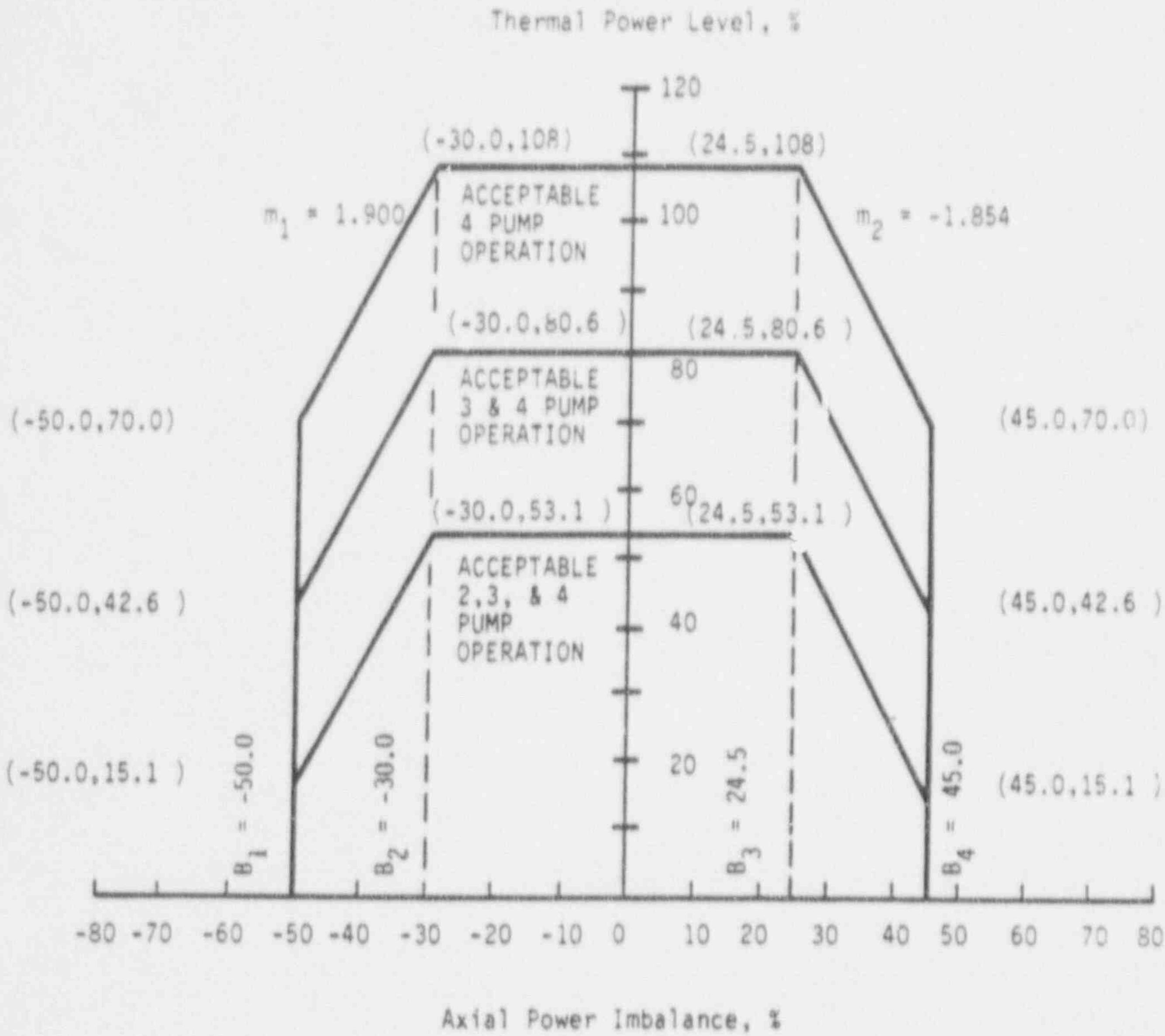
Figure 2.1-3



TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS

Amendment No. ~~13~~, ~~17~~, ~~28~~, ~~39~~, ~~48~~,
~~78~~, ~~126~~, ~~135~~, 142

Figure 2.3-1

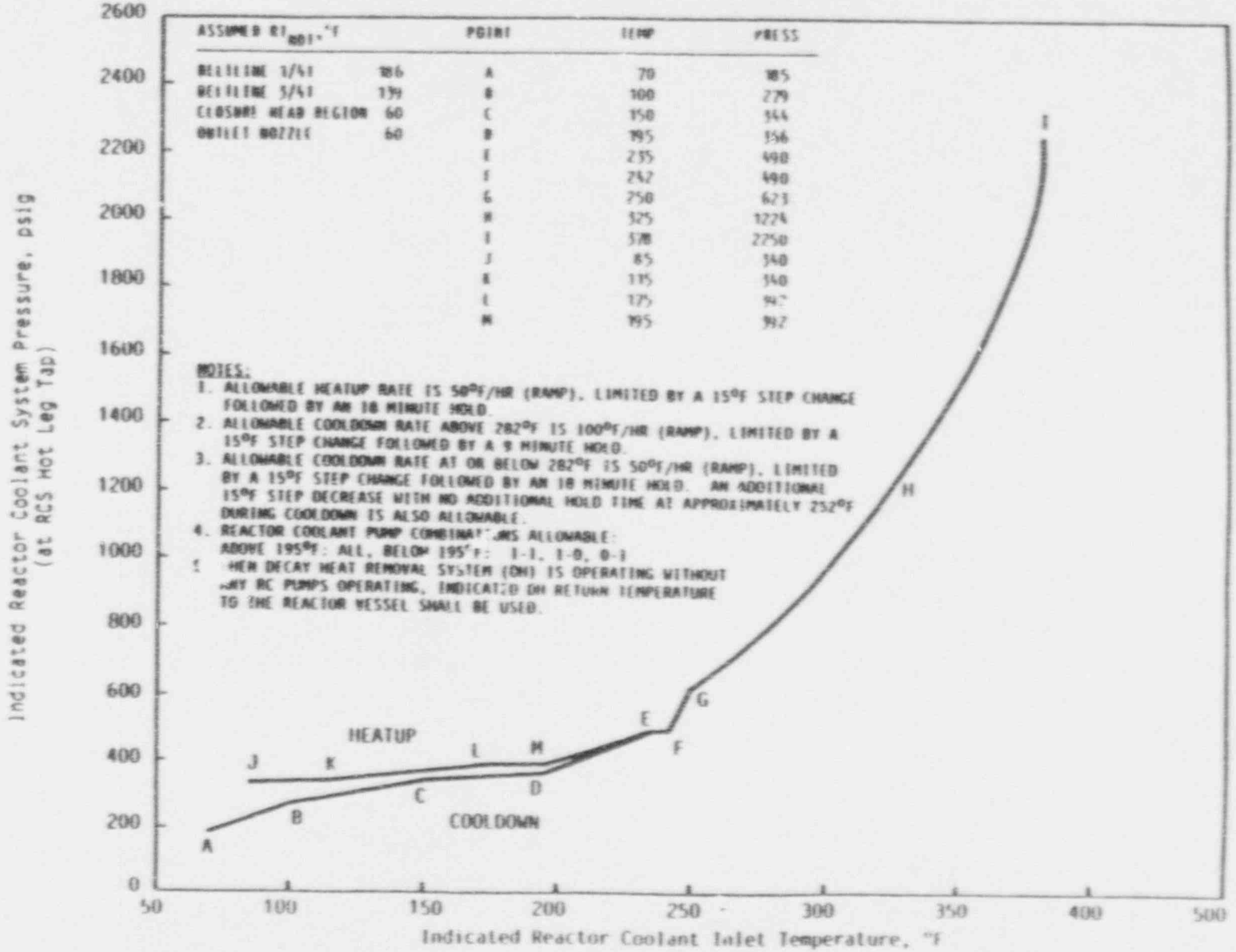


PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS FOR
AXIAL POWER IMBALANCE

TMI-1

Figure 2.3-2

Figure 3.1-1 Reactor Coolant System Heatup/Cooldown Limitations
(Applicable thru 10EFPY)

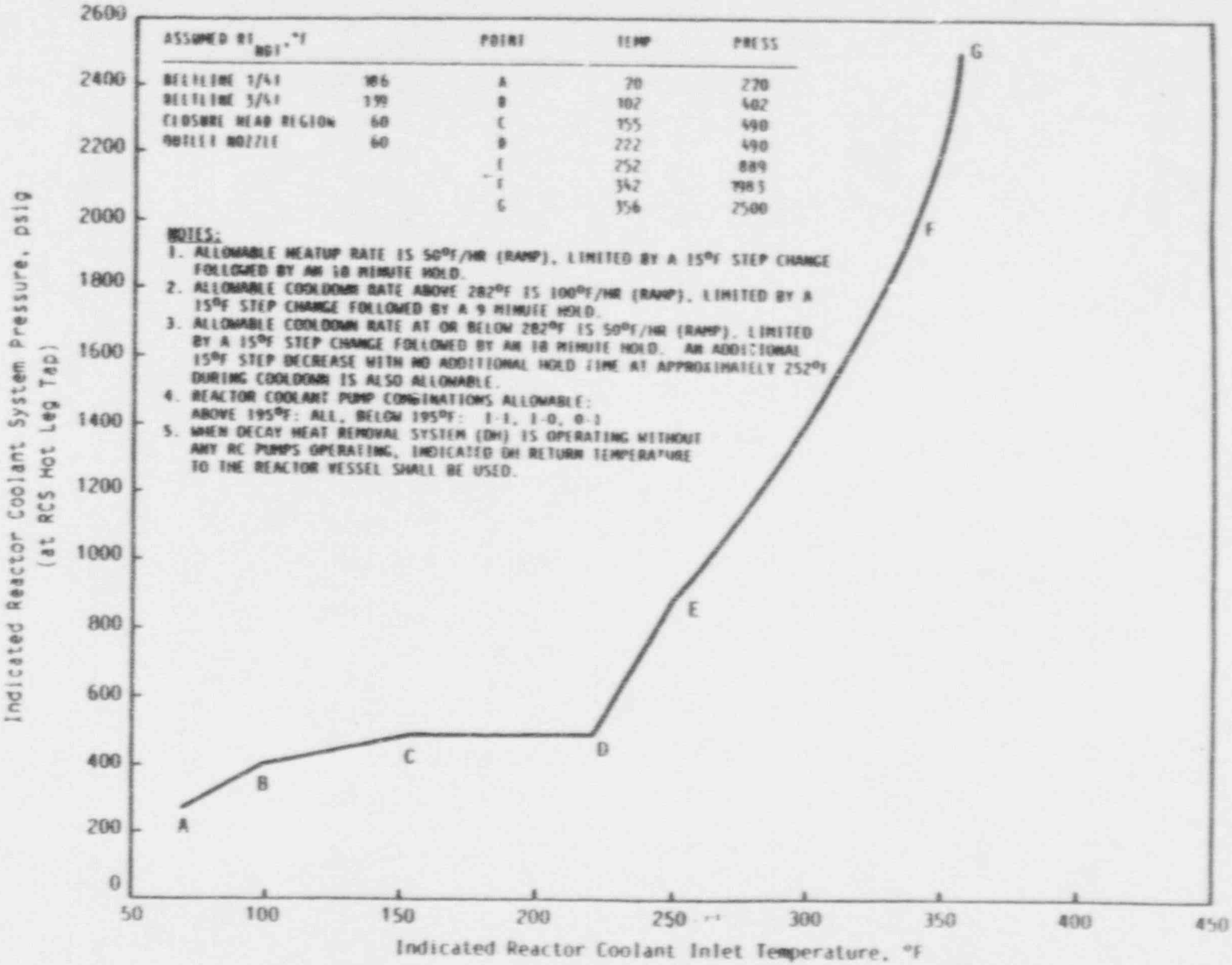


ASSUMED RT_{NDT}, °F		POINT	TEMP	PRESS
BELTLINE 1/4T	186	A	70	185
BELTLINE 3/4T	139	B	100	279
CLOSURE HEAD REGION	60	C	150	344
OUTLET NOZZLE	60	D	195	356
		E	235	490
		F	242	490
		G	250	623
		H	325	1224
		I	378	2250
		J	85	340
		K	115	340
		L	175	392
		M	195	392

NOTES:

1. ALLOWABLE HEATUP RATE IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD.
2. ALLOWABLE COOLDOWN RATE ABOVE 282°F IS 100°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY A 9 MINUTE HOLD.
3. ALLOWABLE COOLDOWN RATE AT OR BELOW 282°F IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD. AN ADDITIONAL 15°F STEP DECREASE WITH NO ADDITIONAL HOLD TIME AT APPROXIMATELY 252°F DURING COOLDOWN IS ALSO ALLOWABLE.
4. REACTOR COOLANT PUMP COMBINATIONS ALLOWABLE: ABOVE 195°F: ALL, BELOW 195°F: 1-1, 1-0, 0-1
5. WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.

Figure 3.1-2 Reactcr Coolant Inservice Leak and Hydrostatic Test
(Applicable thru 10EFPY)



ASSUMED RT_{NDT}, °F		POINT	TEMP	PRESS
BELTLINE 1/4t	186	A	70	270
BELTLINE 3/4t	159	B	102	402
CLOSURE HEAD REGION	60	C	155	490
OUTLET NOZZLE	60	D	222	490
		E	252	889
		F	342	1983
		G	356	2500

NOTES:

1. ALLOWABLE HEATUP RATE IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD.
2. ALLOWABLE COOLDOWN RATE ABOVE 282°F IS 100°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY A 9 MINUTE HOLD.
3. ALLOWABLE COOLDOWN RATE AT OR BELOW 282°F IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD. AN ADDITIONAL 15°F STEP DECREASE WITH NO ADDITIONAL HOLD TIME AT APPROXIMATELY 252°F DURING COOLDOWN IS ALSO ALLOWABLE.
4. REACTOR COOLANT PUMP COMBINATIONS ALLOWABLE: ABOVE 195°F: ALL, BELOW 195°F: 1-1, 1-0, 0-1
5. WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.

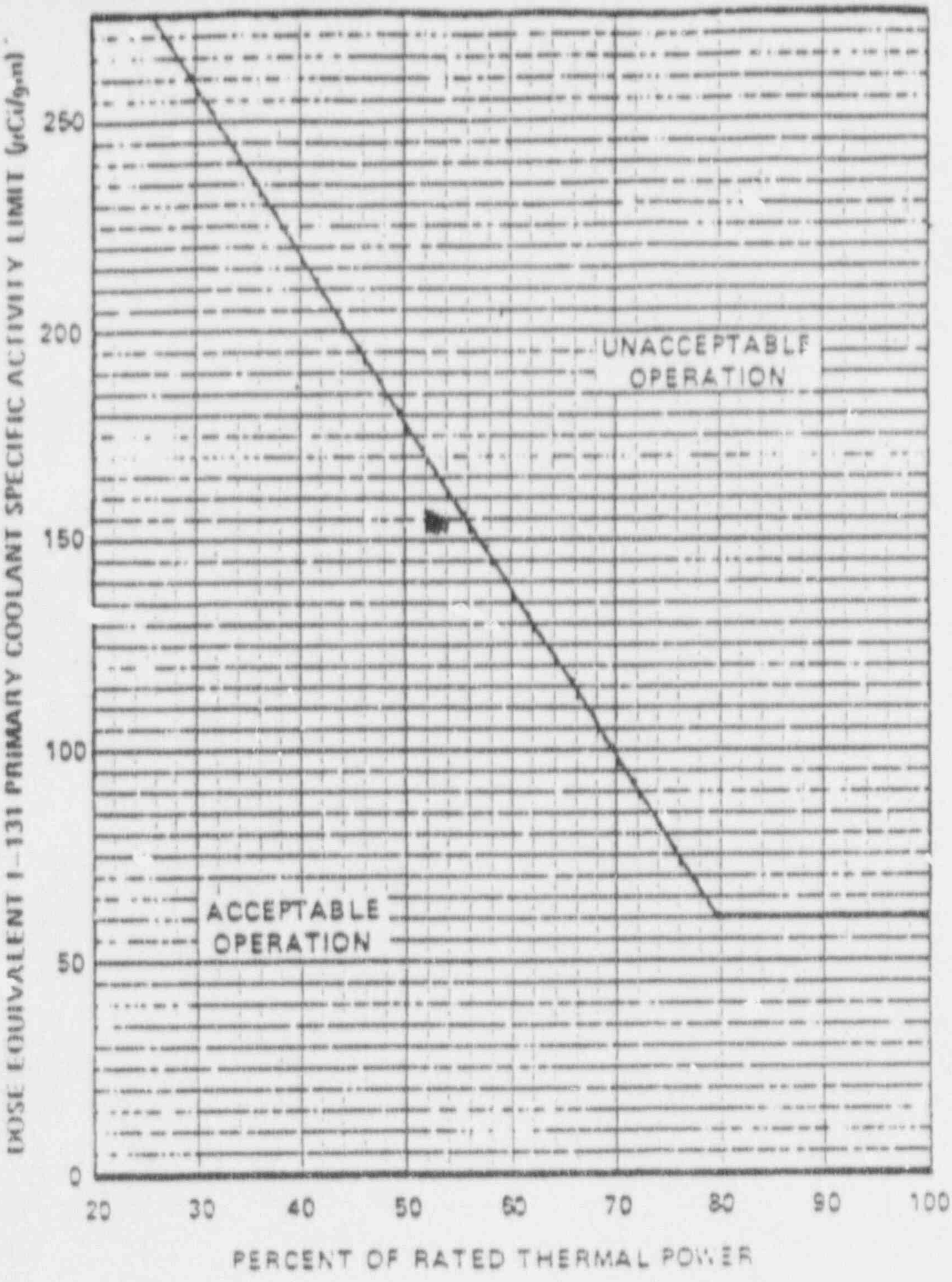
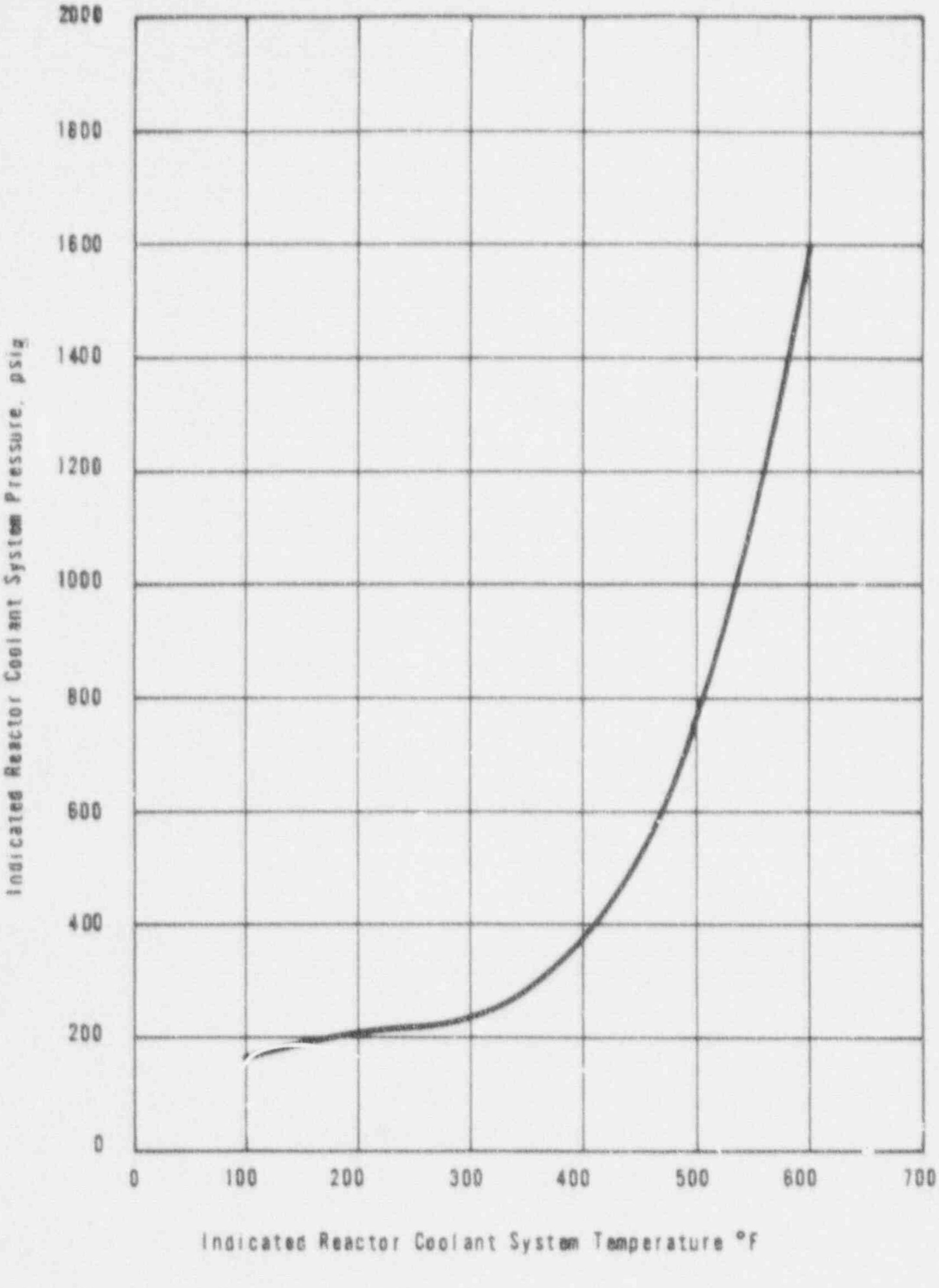


FIGURE 3.1-2a

Dose equivalent 1-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER (with the Primary Coolant Specific Activity >1.0 μCi/gram Dose Equivalent 1-131).



LIMITING PRESSURE VS TEMPERATURE
CURVE FOR 100 STD CC/LITER H_2O

THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.1-3

3.1.11 REACTOR INTERNALS VENT VALVES

Applicability

Applies to Reactor Internals Vent Valves

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Specifications

3.1.11.1 The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.16.

3.1.12 Pressurizer Power Operated Relief Valve (PORV) and Block Valve

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.1 The PORV shall not be taken out of service, nor shall it be isolated from the system (except that the PORV may be isolated to limit leakage to within the limits of Specification 3.1.6) unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out or MU-V16A/B/C/D and MU-V217 are closed.
b. Head of the Reactor Vessel is removed.
c. Tavg is above 332°F.

3.1.12.2 The PORV settings shall be as follows, within the tolerances of ± 25 psi and ± 12°F:

Above 275°F - 2450 psig
Below 275°F - 485 psig

3.1.12.3 If the reactor vessel head is installed and Tavg is ≤332°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D and MU-V217 are closed, and
b. Pressurizer level is ≤ 220 inches. If pressurizer level is > 220 inches, restore level to ≤ 220 inches within 1 hour.

3.1.12.4 PORV and Block Valve

The PORV and the associated block valve shall be OPERABLE during HOT STANDBY, STARTUP, AND POWER OPERATION:

a. With the PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the PORV block valve inoperable, within 1 hour either restore the PORV block valve to OPERABLE status or close the PORV (verify closed) and remove power from the PORV; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With either the PORV or block valve inoperable, restore the inoperable valve to operable status prior to startup from the next cold shutdown unless the cold shutdown occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a cold shutdown occurs within this 90-day period, restore the inoperable valve to operable status prior to the startup for the next fuel cycle.

Bases

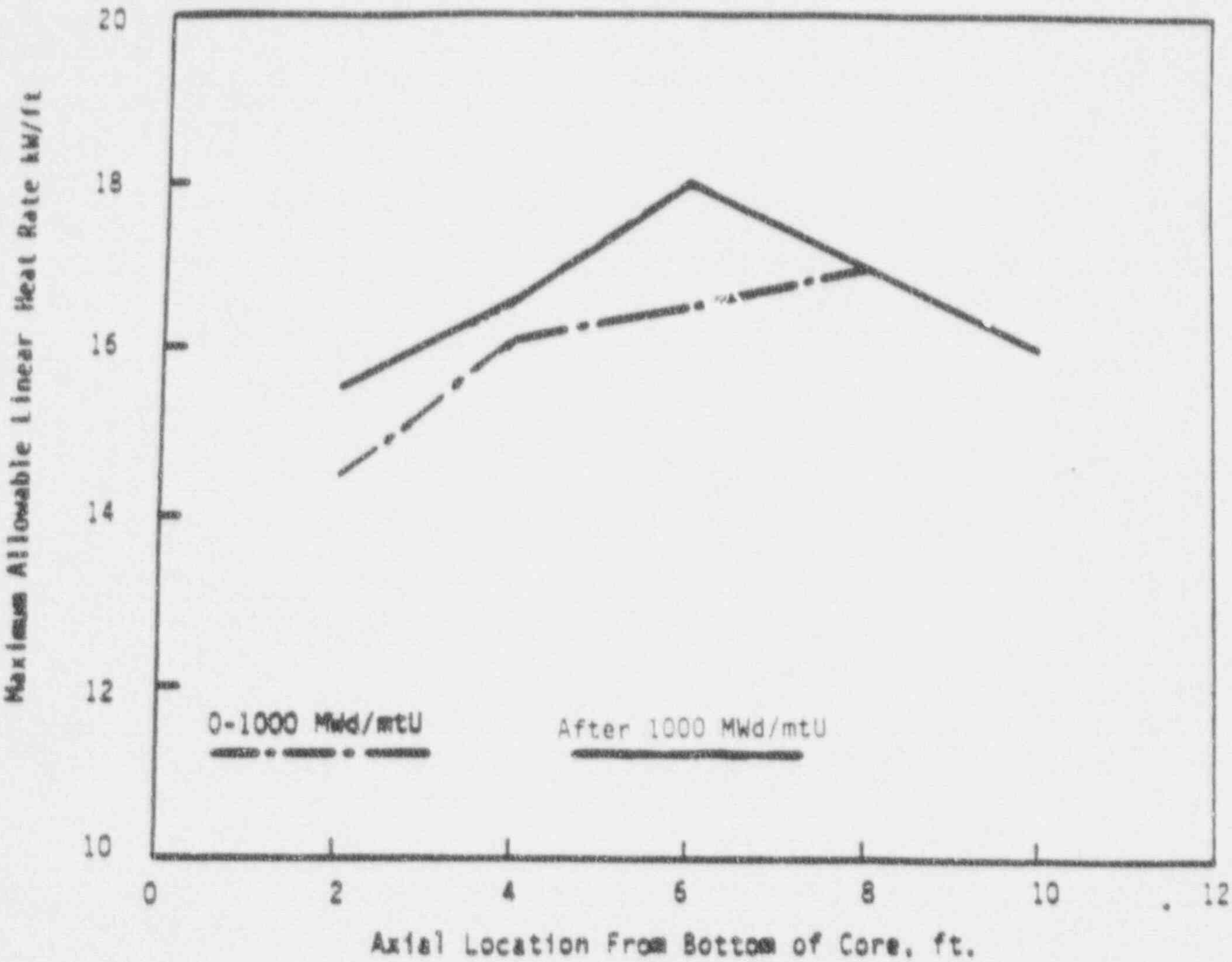
If the PORV is removed from service, sufficient measures are incorporated to prevent severe overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere. In order to prevent exceeding leakage rates specified in T.S. 3.1.6., the PORV may be isolated.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2.

With RCS temperatures less than 332°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than 220 inches to prevent severe overpressurization in the event of any single failure.

Both the PORV and the PORV block valve should be operable during the HOT STANDBY, STARTUP, AND POWER OPERATION. If either the PORV or the PORV block valve are inoperable, the PORV discharge line should be isolated to prevent potential uncontrolled RCS depressurization.

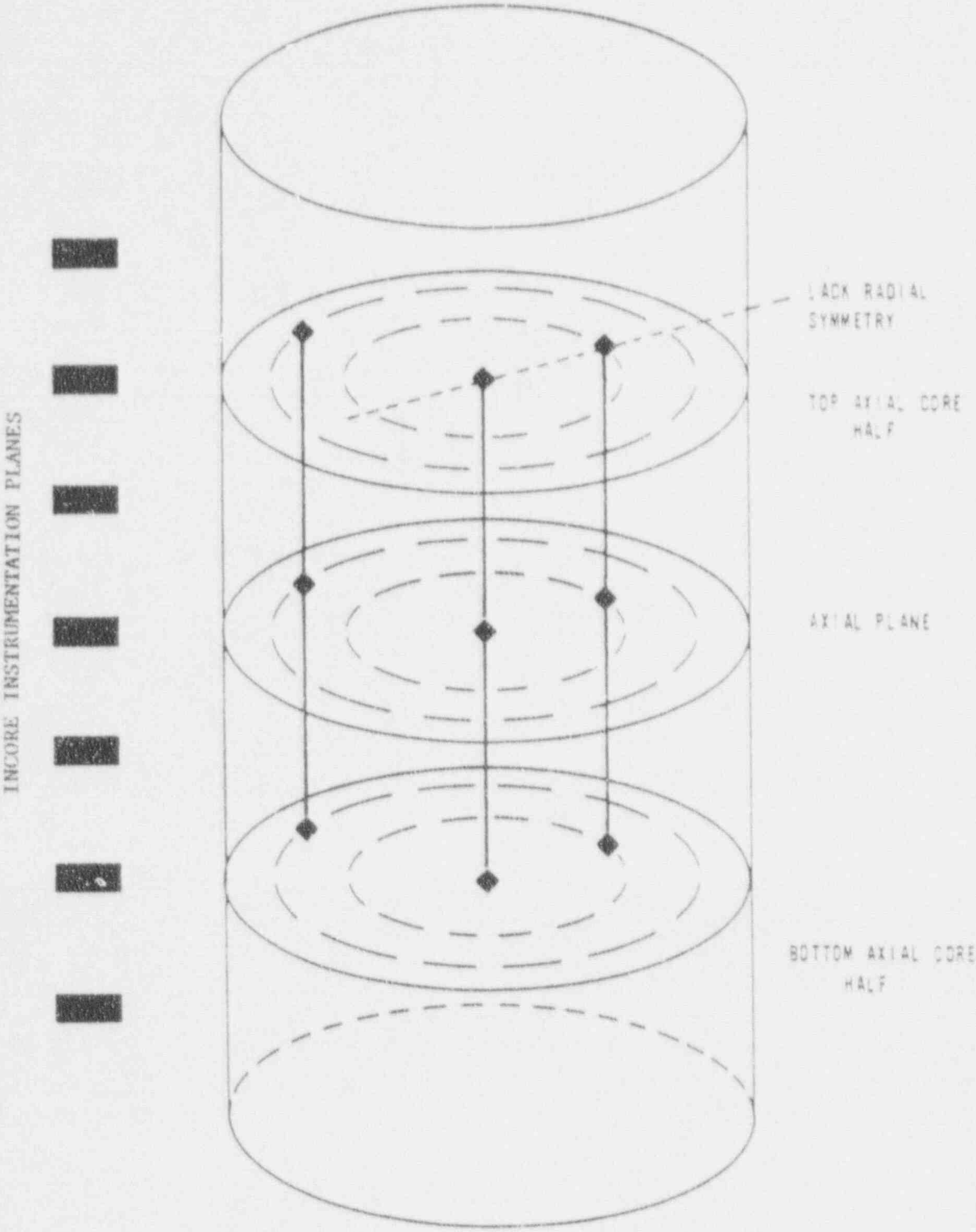
For protection from severe overpressurization during HPI testing, refer to Section 4.5.2.1.c.



LOCA LIMITED MAXIMUM
ALLOWABLE LINEAR HEAT RATE

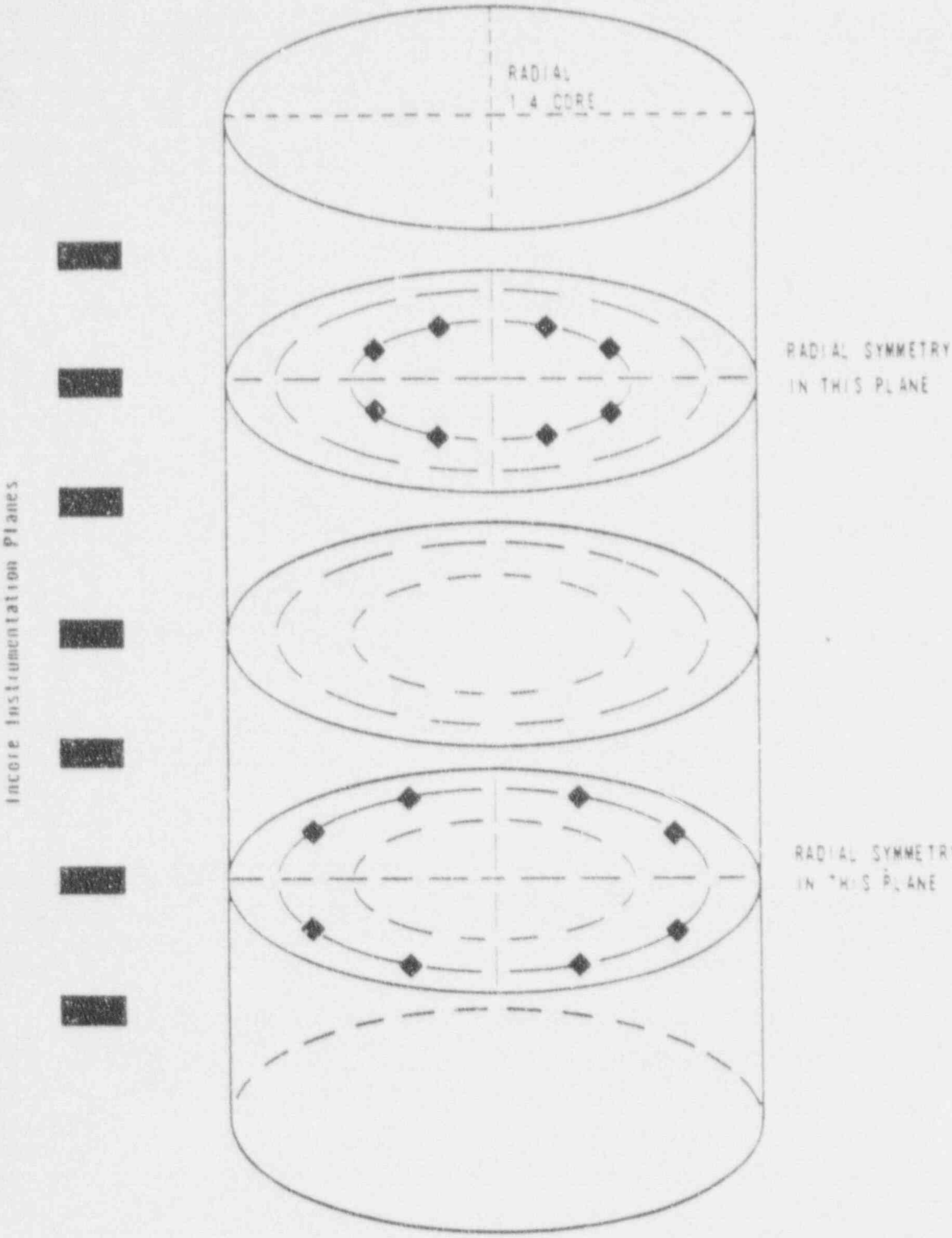
TMI-1

Figure 3.5-2M



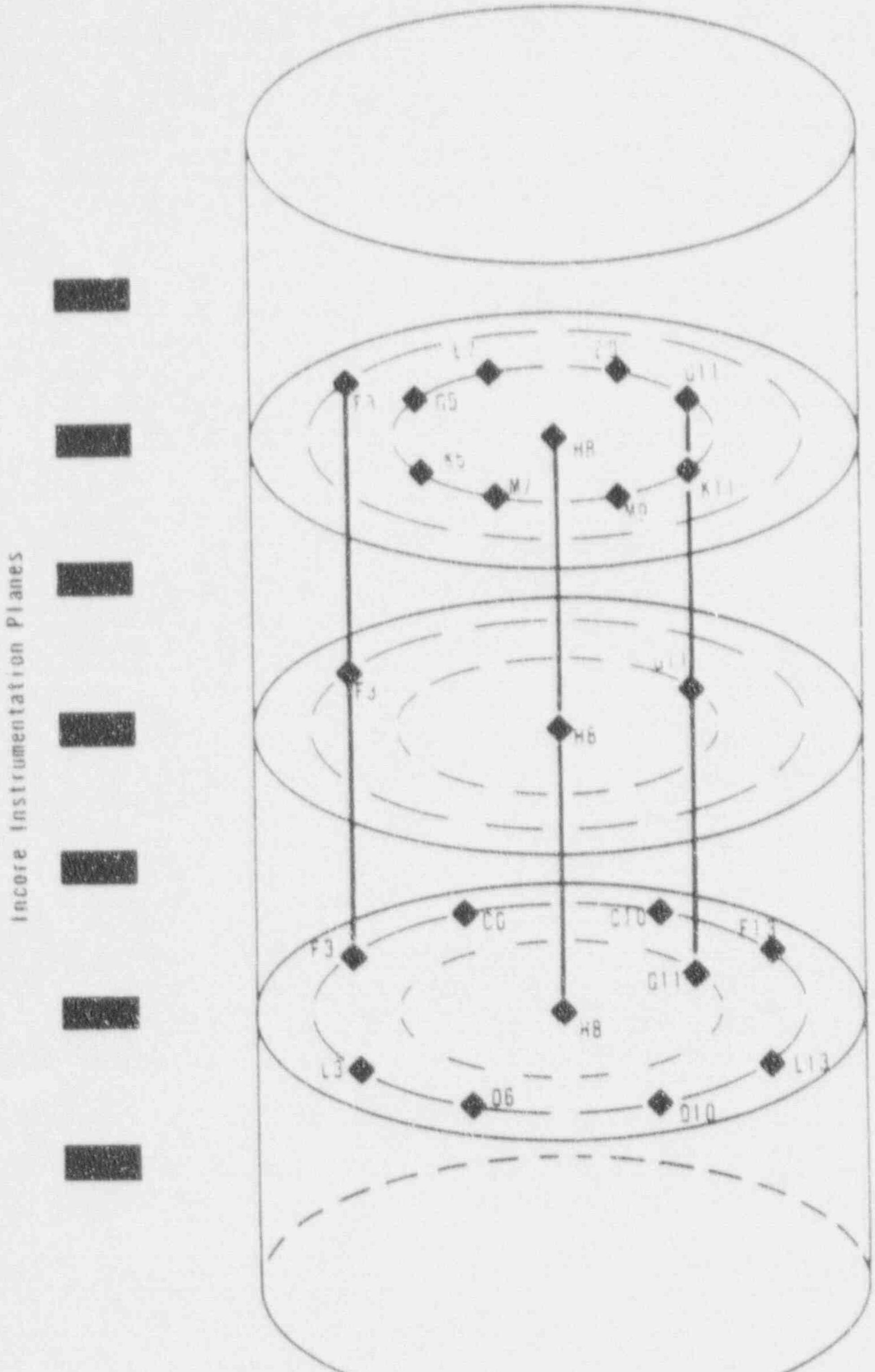
INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-1



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX INDICATION
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-2



INCORE INSTRUMENTATION SPECIFICATION

THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-3

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage, close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared OPERABLE. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (See Specification 4.4.1.7.a), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below.

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of Specification 4.4.1.7.a and perform either (1) or (2) below.

(1) Restore the leaking valve to OPERABILITY within the following 72 hours.

(2) Maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test of Specification 4.4.1.7.a every 3 months. In order to accomplish repairs, one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V1-A&D) shall be limited to less than 31° and (AH-V1-B&C) shall be limited to less than 33° open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY, and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons:

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

(3) Performance of Technical Specification required surveillance.

(4) Radiation Surveys.

(5) Engineering support of safety-related modifications for pre-outage planning.

(6) Purging prior to shutdown to prevent delaying of outage commencement (24 hours prior to shutdown).

c. Containment purging is permitted for Reactor Building pressure control.

d. To the extent practicable, the above containment entries shall be scheduled to coincide, in order to minimize instances of purging.

3.6.11 When the reactor is in COLD SHUTDOWN or REFUELING SHUTDOWN, continuous purging is permitted with the Reactor Building purge isolation valves opened fully.

Bases

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed an hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based in the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure of 2.5 psig greater than the internal pressure.

Due to industry reports of elastomer degradation in containment purge valve seats, unique action requirements are now designated to help preclude common mode failure of both valves in series. An increased frequency of leak rate testing is also incorporated to help assure timely discovery and resolution of any seat degradation.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30° open position.

Allowing purge operations during STARTUP, HOT STANDBY, and POWER OPERATION (TS 3.6.10) is more beneficial than requiring a cooldown to COLD SHUTDOWN from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions 2) radiolytic decomposition of water and 3) corrosion of metals within containment. The recombiner is designed in accordance with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971, the acceptance criteria of the Standard Review Plan (SRP) 6.2.5, and NUREG 0578, July 1979. In addition to the installed hydrogen recombiner, a second recombiner including all piping, electrical, and structural provisions is available on site.

4.4 Reactor Building

4.4.1 Containment Leakage Tests

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Test (ILRT)

4.4.1.1.1 Design Pressure Leakage Rate

The design integrated leakage rate, (L_d), from the Reactor Building at the 55 psig design pressure, P_d, is 0.1 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.2 Allowable Integrated Leakage Rate

The maximum allowable integrated leakage rate, (L_a), from the Reactor Building at the calculated peak Reactor Building internal pressure of 50.6 psig (P_a) associated with the design basis accident, shall not exceed 0.1 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.3 Conduct of Tests

a. During the period between the initiation of the containment inspection and the performance of a periodic ILRT, no repairs or adjustments shall be made unless the inspection reveals structural deterioration which could affect the containment structural integrity or leak-tightness. Such structural deterioration shall be corrected before performance of the test and a description of the deterioration and the corrective action taken shall be reported as part of the test report submitted in accordance with 10 CFR 50 Appendix J.V.B.

b. The test duration shall be at least 24 hours unless experience from at least two prior tests provides evidence of the adequacy of a shorter test duration.

c. Missile shielded lines outside the secondary shield will not be vented.

d. All containment components normally pressurized by the penetration pressurization system shall be at atmospheric pressure during the ILRT.

4.4.1.1.4 Frequency of test

The containment leakage rates shall be demonstrated at the following test schedule:

a. Three Type A tests shall be conducted in accordance with 10 CFR 50 Appendix J, Section III.D.1.(a), except as noted in 4.4.1.1.4.b and as stated in the Bases.

b. Where an exemption from the frequency specified by 10 CFR 50 Appendix J has been granted by the NRC, the schedule specified by the exemption shall apply.

4.4.1.1.5 Acceptance Criteria for Periodic ILRT

For initial and periodic ILRT at P_a, L_{am} shall be less than 0.75 L_a.

4.4.1.1.6 Corrective Action and Retest

If the test data from a completed leakage rate test does not meet the acceptance criteria, the ILRT need not be repeated provided local leakage rate measurements are made at pressure P_a before and after repair to demonstrate that the leakage rate reduction achieved by the repairs reduces the overall measured integrated leakage rate to an acceptable value.

4.4.1.1.7 Report of Test Results

Reporting of test results shall be in accordance with 10 CFR 50 Appendix J.V.B. requirements.

4.4.1.2 Local Leakage Rate Tests (LLRT)

4.4.1.2.1 Scope of Testing

LLRT of penetrations and valves identified in the FSAR shall be performed in accordance with 10 CFR 50 Appendix J except as provided in 4.4.1.2.5.c.

4.4.1.2.2 Conduct of Tests

LLRT shall be performed pneumatically at a pressure of not less than P_a, with the exception that the access hatch door seal test shall normally be performed at 10 psig and the test every six months specified in 4.4.1.2.5.a shall be performed at a pressure not less than P_a.

4.4.1.2.3 Acceptance Criteria

The combined leakage from all penetrations and valves subject to LLRT shall not exceed 0.60 L_a (the maximum allowable leakage rate at P_a).

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency

LLRT shall be performed at a frequency as required by 10 CFR 50 Appendix J, except that:

a. The entire personnel and emergency airlocks shall be tested once every six months. When the airlocks are opened during the interim between six month tests, the airlock door resilient seals shall be tested within 72 hours of the first of each of a series of openings. This requirement exists whenever containment integrity is required.

b. An interspace pressurization test (See T.S. 4.4.1.7.a) shall be performed for reactor building purge isolation valves every 3 months. This requirement is not in effect during cold shutdown.

c. Where an exemption from the frequency specified by 10 CFR 50 Appendix J has been granted by the NRC, the frequency specified by the exemption shall apply.

4.4.1.3 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The valves not stroked every three months shall be stroked during each refueling period.

4.4.1.4 Annual Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any ILRT to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accordance with acceptable procedures, nondestructive tests, and inspections, and local testing where practical, prior to the conduct of any ILRT. Such repairs shall be reported as part of the test results.

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the reactor building integrity shall be followed by either an ILRT or an LLRT, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.5 and 4.4.1.2.3, respectively.

4.4.1.6 Operability of Access Hatch Interlocks

a. At least once per six months the operability of the personnel and emergency hatch door interlocks and the associated control room annunciator circuits shall be determined. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable.

b. During periods when containment integrity is required and an interlock is inoperable, each entry and exit via that airlock shall be locally supervised by a member of the unit operating, maintenance, or technical staffs, to assure that only one door is open at any time and that both doors are properly closed following use. A record of supervision and verification of closure shall be maintained during periods of interlock inoperability in an appropriate station log.

c. If an interlock is inoperable for more than 14 days following determination of inoperability, use of the airlock, except for emergency purposes, shall be suspended until the interlock is returned to operable status.

4.4.1.7 Operability of Purge Valves

a. A periodic pressurization of the purge valve interspace to 50.6 psig per Specification 4.4.1.2.5.b shall be performed to help assure timely detection and resolution of valve and/or actuator degradation. The acceptance criteria is that total local leakage, when updated for the new purge valve leakage, shall be less than 0.60 L_a. See Specification 3.6.8 for further action.

b. The rubber seats on purge valves shall be visually examined and durometer tested each refueling interval to detect degradation (e.g. cracking, brittleness, etc.) and to assure timely cleaning, lubrication, and seat replacement.

Bases (1)

The performance of periodic ILRT and LLRT during the plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, "as found" local leakage results must be documented for correction of the ILRT results. Containment isolation valves are to be closed in the normal manner prior to LLRT or ILRT. Containment Isolation Valves are addressed in the UFSAR (Reference 2).

The minimum of 24 hours was specified for the ILRT to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic ILRT is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

Surveillance tests for measuring leakage rates are consistent with the requirements of 10 CFR 50, Appendix J with the following exemption. The third test of each Type A testing set need not be conducted when the plant is shut down for the 10-year plant inservice inspections (Reference 3). The operational readiness of the containment is proven by the ILRT, and in accordance with license requirements, when completed pursuant to the frequency stated in Technical Specification 4.4.1.1.4.

The specified frequency of periodic ILRT is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.1 percent leakage rate at 55 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at P_a, of those portions of the containment envelope that are most likely to develop leaks during reactor operation and the low value of leakage that is specified as acceptable from penetrations and isolation valves (0.60 L_a). Third is the tendon stress surveillance integrity program which provides assurance that an important part of the structural integrity of the containment is maintained.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the reactor building liner due to the mechanical closure involved. The basis for specifying a total leakage rate of 0.60 L_a from those penetrations and isolation valves is that more than one-half of the allowable integrated leakage rate will be from these sources.

Valve operability tests are specified to assure proper closure or opening of the reactor building isolation valves to provide for isolation or functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is established that such testing is not practical during operation. Valves that cannot be full-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

Periodic surveillance of the airlock interlock systems (Reference 4) is specified to assure continued operability and preclude instances where one or both doors are inadvertently left open. When an airlock is inoperable and containment integrity is required, local supervision of airlock operation is specified.

Purge valve interspace pressurization test operability requirements, inspections, and durometer testing provide a high degree of assurance of purge valve performance as containment isolation barriers.

<u>Reference</u>

(1) UFSAR, Chapter 5.7.4 - "Post Operational Leakage Rate Tests"
(2) UFSAR, Tables 5.7-1 and 5.7-3
(3) NRC Letter dated February 25, 1991 (C311-91-3033)
(4) UFSAR, Table 5.7-2

4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM & REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Loading Sequence

Applicability: Applies to periodic testing requirements for safety actuation systems.

Objective: To verify that the emergency loading sequence and automatic power transfer is operable.

Specifications:

4.5.1.1 Sequence and Power Transfer Test

a. During each refueling interval, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable.

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, verification of pressure/flow, or control board indicating lights initiated by separate limit switch contacts.

-M. U. Pump
-D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
-R. B. Cooling Pump
-R. B. Ventilators
-D. H. Closed Cycle Cooling Pump
-N. S. Closed Cycle Cooling Pump
-D. H. River Cooling Pump
-N. S. River Cooling Pump
-D. H. and N. S. Pump Area Cooling Fan
-Screen House Area Cooling Fan
-Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psig Pressure Test Signal.)
-Motor Driven Emergency Feedwater Pump

c. Following successful transfer to the emergency diesel, the diesel generator breaker will be opened to simulate trip of the generator then reclosed to verify block load on the reclosure.

4.5.1.2 Sequence Test

a. At intervals not to exceed 3 months, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power.

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, verification of pressure/flow, or control board indicating lights initiated by separate limit switch contacts.

c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance, which affects the position indicator.

4.5.2.3 Core Flooding

a. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. During depressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flooding tank level verifies that all valves have opened.

4.5.2.4 Component Tests

a. At intervals not to exceed 3 months, the components required for emergency core cooling will be tested.

b. The tes will be considered satisfactory if the pumps and fans have been success.ully started and the valves have completed their travel as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, verification of pressure/flow, or control board indicating lights initiated by separate limit switch contacts.

Bases

The emergency core cooling systems (Reference 1) are the principal reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

The minimum acceptable HPI/LPI flow assures proper flow and flow split between injection legs.

With the reactor shutdown, the valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

Reference

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

4.5.3 REACTOR BUILDING COOLING AND ISOLATION SYSTEM

Applicability

Applies to testing of the reactor building cooling and isolation systems.

Objective

To verify that the reactor building cooling systems are operable.

Specification

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At each refueling interval and simultaneously with the test of the emergency loading sequence, a Reactor Building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.

   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during the component test of the Reactor Building Cooling and Isolation System.

   The test will be considered satisfactory if the spray pumps have been successfully started as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

2. Compressed air will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every five years.

b. Reactor Building Cooling and Isolation Systems

1. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. A test signal will actuate the Reactor Building Emergency Cooling System valves to demonstrate operability of the coolers.

2. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights and a second means of verification, such as: the station computer, local verification, verification of pressure/flow, or control board component operating lights initiated by separate limit switch contacts.

4.5.3.2 Component Tests

a. At intervals not to exceed three months, the components required for Reactor Building Cooling and Isolation will be tested.

b. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights and a second means of verification, such as: the station computer, local verification, verification of pressure/flow, or control board component operating lights initiated by separate limit switch contacts.

Bases

The Reactor Building Cooling and Isolation Systems and Reactor Building Spray System are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure (References 1 and 2).

The delivery capability of one Reactor Building Spray Pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump.

With the pumps shut down and the Borated Water Storage Tank outlet valve closed, the Reactor Building spray injection valves can each be opened and closed by operator action. With the Reactor Building spray inlet valves closed, low pressure air can be blown through the test connections of the Reactor Building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves and instrumentation of the Reactor Building Cooling System are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment.

The Reactor Building fans are normally operating periodically, constituting the test that these fans are operable.

Reference

(1) UFSAR, Section 6.2 - "Reactor Building Spray System"
(2) UFSAR, Section 6.3 - "Reactor Building Emergency Cooling System"

d. The battery will be subjected to a load test at a frequency not to exceed refueling periods. The battery voltage as a function of time will be monitored to establish that the battery performs as expected during this load test.

4.6.3 Pressurizer Heaters

a. The following tests shall be conducted at least once each refueling:

(1) Pressurizer heater groups 8 and 9 shall be transferred from the normal power bus to the emergency power bus and energized. Upon completion of this test, the heaters shall be returned to their normal power bus.

(2) Demonstrate that the pressurizer heaters breaker on the emergency bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

(3) Verify that following input of the Engineered Safeguards Signal, the circuit breakers, supplying power to the manually transferred loads for pressurizer heater groups 8 and 9, have been tripped.

Bases

The tests specified are designed to demonstrate that one diesel generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an engineered safeguards Actuation Signal. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential overload condition. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the station battery is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

The PORV has a remotely operated block valve to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valve and the block valve is supplied from an ESF power source to ensure the ability to seal this possible RCS leakage path.

The requirement that a minimum of 107 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation.

4.15 MAIN STEAM SYSTEM INSERVICE INSPECTION

Applicability

This technical specification applies to the inservice inspection of four welds in the Main Steam System identified as MS-0001, MS-0002, MS-0003, and MS-0004L of the TMI-1 Inservice Inspection Program.

Objective

The objective of the Inservice Inspection Program is to provide assurance of the continuing integrity of that portion of the Main Steam System in which a postulated failure would produce pressures in excess of the compartment wall and/or slab capacities.

Specification

4.15.1. The four weld joints identified above shall be 100 percent inspected in accordance with the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant components, defined in the TMI-1 Inservice Inspection Program. Inspections are to be performed at a frequency of once every 3-1/2 years (or during the nearest refueling outage).

Prior to initial plant operation, a preoperational inspection of the identified weld joints will be performed and any data acquired will be recorded to form a baseline on which to compare results of subsequent inspections.

Bases

Calculations (Reference 1) postulated that breaks in the main steam lines at the containment penetrations in small compartments No. 2 and No. 5 could produce pressures in excess of wall and/or slab capacities.

Inspections are conducted at an inspection frequency of 3 1/2 year intervals following initial plant startup. These inspections have revealed that no degradation of the welds has occurred during the inspection cycles up to and including the 9R outage inspection. Consequently, as further degradation is not expected to occur, justification to extend the inspection frequency to once every ten (10) years is being developed. The conclusions of the technical benefit review will be submitted to the NRC for evaluation in a Technical Specification change request.

Reference

(1) UFSAR, Appendix 14A, Section 7.2.1

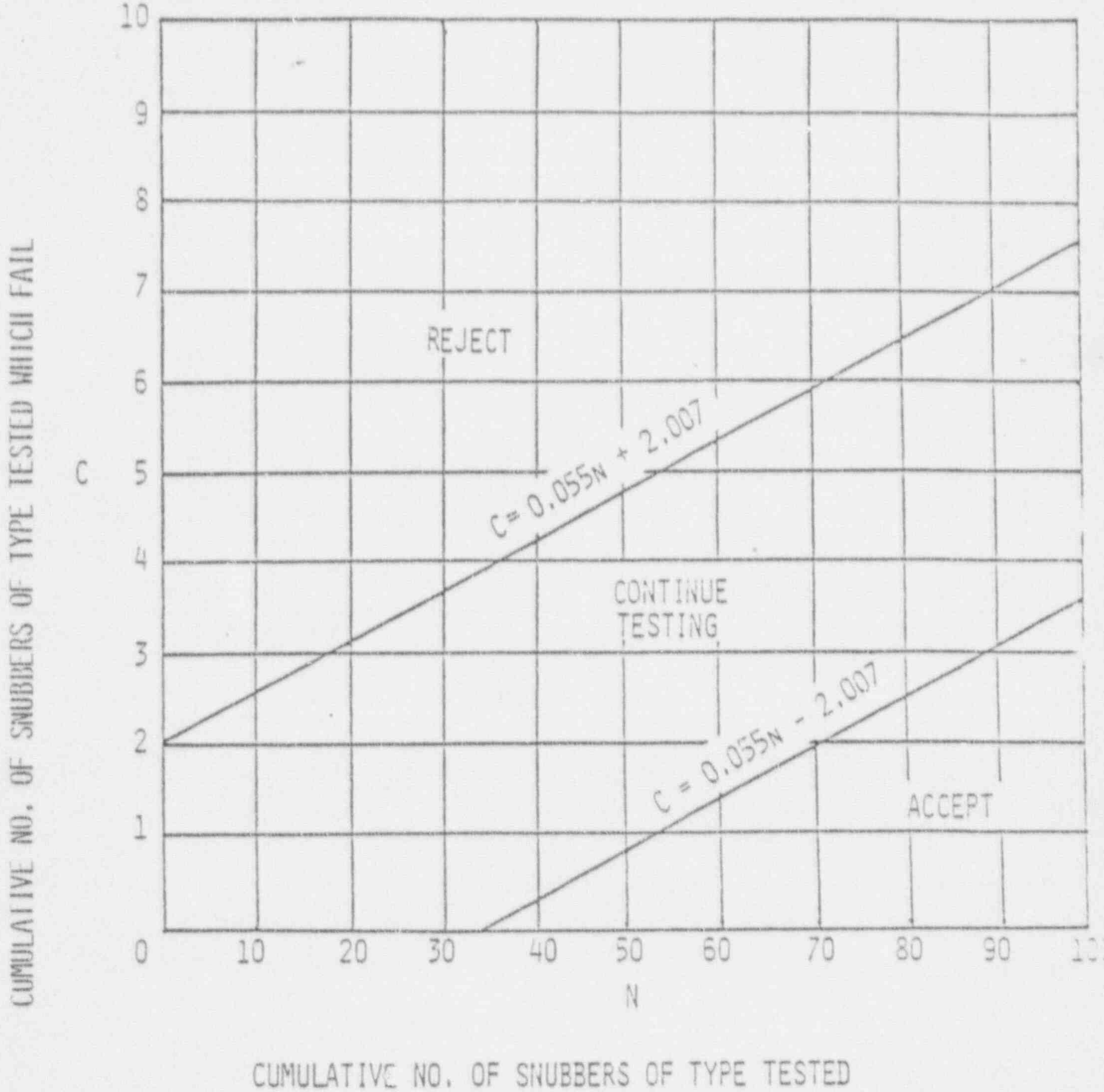


FIGURE 4.17-1

SNUBBER FUNCTIONAL TEST - SAMPLE PLAN 2

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