AmerGen

An Exelon/British Energy Company

AmerGen Energy Company, LLC
Three Mile Island Unit 1
Route 441 South, P.O. Box 480
Middletown, PA 17057

Telephone: 717-944-7621

10 CFR 50.90

August 20, 2001
5928-01-20211

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
LICENSE AMENDMENT REQUEST NO. 308, TRANSMITTAL OF CAMERA-READY TECHNICAL SPECIFICATION PAGES

This letter transmits the camera-ready Technical Specification pages to support NRC issuance of an amendment approving TMI Unit 1 License Amendment Request No. 308.

Please contact David J. Distel at (610) 765-5517, if you have any questions regarding this submittal.

Very truly yours,

M. E. Warner
Vice President, TMI Unit 1

MEW/djd

Enclosure: TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 308

cc: H. J. Miller, USNRC Regional Administrator, Region I
J. D. Orr, USNRC TMI Unit 1 Resident Inspector
T. G. Colburn, USNRC TMI Unit 1 Senior Project Manager
File No. 01021

A001

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 308
Transmittal of Camera-Ready Technical Specification Pages

COMMONWEALTH OF PENNSYLVANIA)

COUNTY OF DAUPHIN) SS:

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These camera-ready Technical Specification pages are submitted in support of Licensee's request to change the Technical Specifications for Three Mile Island, Unit 1. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

AmerGen Energy Company, LLC

BY: Mark E W
Vice President, TMI Unit 1

Sworn and Subscribed to before me

this 30th day of August 2001.

Linda C. Witter
Notary Public

Notarial Seal
Linda C. Witter, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Sept. 25, 2004
Member, Pennsylvania Association of Notaries

ENCLOSURE 1

TMI Unit 1 Technical Specification Revised Pages for

License Amendment Request No. 308

(Pages ii, vii, 3-3, 3-4, 3-5, 3-5a, 3-5b, 3-18d, 3-18e, 3-18f, and 4-41)

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LIST OF FIGURES

Amendment Nos. ~~11, 17, 29, 39, 45, 50, 59, 72, 106, 109, 120, 126, 134, 142, 150, 164, 167, 168, 184, 211, 227~~

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objectives

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

To assure that reactor vessel integrity by maintaining the stress intensity as a result of operational plant heatup and cooldown conditions and inservice leak and hydro test conditions below values which may result in non-ductile failure.

Specification

3.1.2.1 For operations until 29 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding 29 effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specification submitted in accordance with 10 CFR 50, Appendix G.

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional temperature step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 240°F indicated RCS temperature.

The unirradiated reference nil ductility temperature (RT_{NDT}) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated as described in Reference No. 5.

The predicted RT_{NDT} was calculated using the respective predicted neutron fluence at 29 effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2, Section C.1.1 for the plate metals and for the limiting weld metals (SA-1526 & WF-25).

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 4. Extrapolation of reactor vessel fluxes and corresponding fluence accumulations, based on predicted fuel cycle design conditions through 29 effective full power years of operation are described in Reference 6.

Based on the predicted RT_{NDT} after 29 effective full power years of operation, the pressure/temperature limits of Figure 3.1-1 and 3.1-2 have been established by FTI calculation, Reference No. 7, in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2 and ASME Code Section XI, Appendix G, as modified by ASME Code Case N-640 and N-588. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figure 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in RCS "A" loop hot leg. RCS "A" is most conservative and bounds use of "B".

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

<u>REFERENCES</u>

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits.

(5) FTI Calculation No. 32-5011059-00, "TMI-1 Reactor Vessel Adjusted RTNDT Values for 23 and 29 EFPY."

(6) FTI Calculation No. 86-5010023-00, "TMI Cycle 5-11 Final Report."

(7) FTI Calculation No. 32-5011638-02, "TMI-1 29 EFPY P/T Limits."

Figure 3.1-1
Reactor Coolant System Heatup/Cooldown Limitations
[Applicable through 29 EFPY]

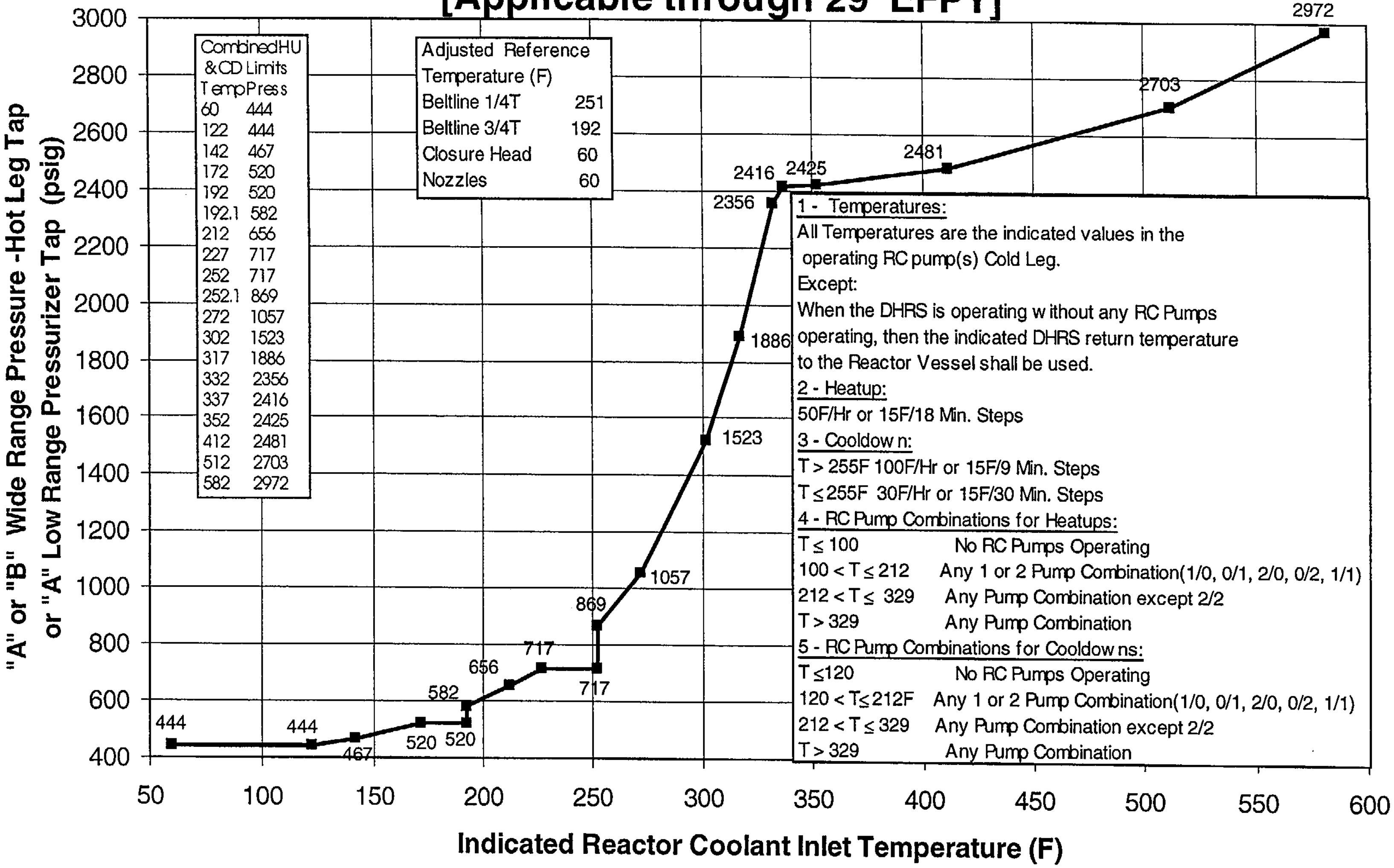
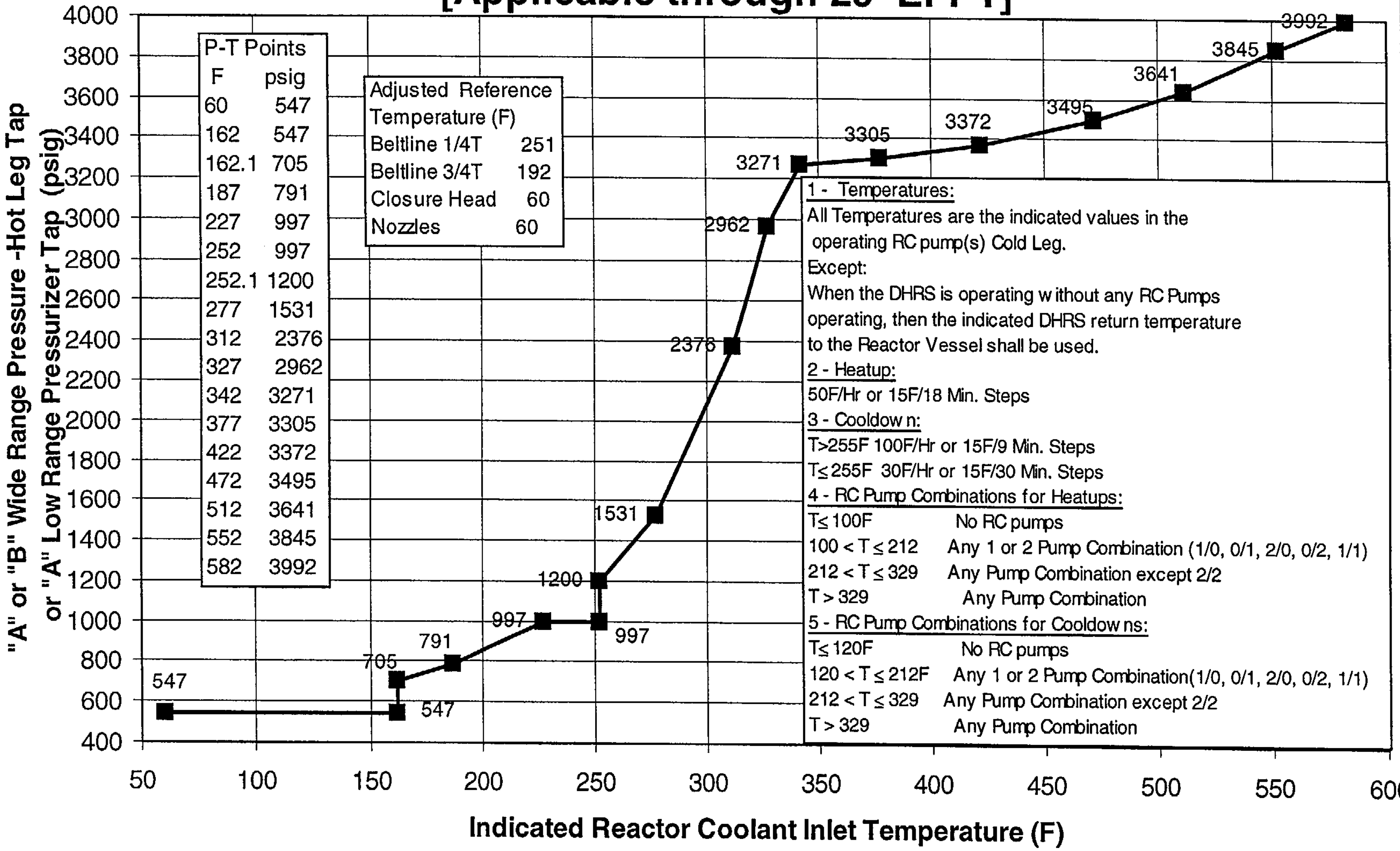


Figure 3.1-2
Reactor Coolant Inservice Leak Hydrostatic Test
[Applicable through 29 EFPY]



Amendment 25, 134, 167, 176, 208

3.1.12 Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)

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 LTOP Protection

If the reactor vessel head is installed and indicated RCS temperature is ≤ 329°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed, and

b. Pressurizer level is maintained ≤ 100 inches. If pressurizer level is > 100 inches, restore level to ≤ 100 inches within 1 hour.

3.1.12.2 The PORV settings shall be as follows:

a. Low Temperature Overpressure Protection Setpoint

1. When indicated RCS temperature is ≤ 329°F, the LTOP system shall be operable as defined in Specification 3.1.12.1 and

2. The PORV will have a maximum lift setpoint of 552 psig.

With the PORV setpoint above the maximum value, within 8 hours either:

1. restore the setpoint below the maximum value, or

2. verify pressurizer level is ≤ 100 inches indicated and satisfy the requirements of Technical Specification 3.1.12.3 allowing the PORV to be taken out of service.

b. Unless the Low Temperature Overpressure Protection Setpoint is in effect, the PORV lift setpoint will be a minimum of 2425 psig.

With the PORV setpoint below the minimum value, within 8 hours either:

1. restore the setpoint above the minimum value, or

2. close the associated block valve, or

3. close the PORV, and remove power from PORV

4. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.1.12.3 When the indicated RCS temperature is below 329°F the PORV shall not be taken out of service, nor shall it be isolated from the system unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out.
b. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed.
c. Head of the Reactor Vessel is removed.

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

a. With the PORV Block Valve inoperable, within 1 hour either:

1. restore the PORV Block Valve to OPERABLE status or

2. close the PORV (verify closed) and remove power from the PORV

3. 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, 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 startup for the next fuel cycle.

Bases

If the PORV is removed from service while the RCS is below 329°F, 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.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2. Above 329°F, the PORV setpoint has been chosen to limit the potential for inadvertent discharge or cycling of the PORV. Other action such as removing the power to the PORV has the same effect as raising the setpoint which also satisfies this requirement. There is no upper limit on this setpoint as the Pressurizer Safety Valves (T.S. 3.1.1.3) provide the required overpressure relief.

Below 329°F, the PORV setpoint is reduced to provide the required low temperature overpressure relief when high pressure sources and flowpaths are in service. There is no lower limit on the pressure actuation specified as lower setpoints also provide this same protection.

In both cases, the setting is specified to reflect the nominal value which allows for normal variations in the temperature setpoint while maintaining the tolerances assumed in the bases for T.S. 3.1.2. Either pressure actuation setpoint is acceptable within the temperature range between 313°F and 329°F.

With RCS temperatures less than 329°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than 100 inches to allow time for action to prevent severe overpressurization in the event of any single failure.

The PORV block valve is required to be OPERABLE during the HOT STANDBY, STARTUP, and POWER OPERATION in order to provide isolation of the PORV discharge line to positively control potential RCS depressurization.

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

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification

4.5.2.1 High Pressure Injection

a. During each refueling interval and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

b. The test will be considered satisfactory if the valves (MU-V-14A/B & 16A/B/C/D) have completed their travel and the make-up pumps are running as evidenced by system flow. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A/B/C/D shall be conducted only under either of the following conditions:

1) Indicated RCS temperature shall be greater than 329°F.
2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. During each refueling period and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps listed in 4.5.1.1b have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.