METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 103

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

METROPOLITAN EDISON COMPANY

By H D Hukill
Director, TMI-1
H. D. Hukill

Sworn and subscribed to me this 18th day of May, 1981.

Pamela Joy Lubrecht
Notary Public

PAMELA JOY LUBRECHT, Notary Public
Middletown, Dauphin County, Pa.
My Commission Expires August 29, 1983

8105220057

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-289
LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No. 103 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with the U. S. Nuclear Regulatory Commission and been served on the chief executives of Londonderry Township, Dauphin County, Pennsylvania and Dauphin County, Pennsylvania by deposit in the United States mail, addressed as follows:

Mr. Donald Hoover, Chairman
Board of Supervisors of
Londonderry Township
R.D. #1, Geyers Church Road
Middletown, PA 17057

Mr. John E. Minnich, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

METROPOLITAN EDISON COMPANY

By H D Hukill
Director, TMI-1
H. D. Hukill

Dated: May 18, 1981

Three Mile Island Nuclear Station, Unit 1
Operating License No. DPR-50
Docket No. 50-289

Technical Specification Change Request No. 103

The licensee herein requests that a number of existing Technical Specification pages, contained in Appendix A to the Operating License, be amended. The proposed amended pages together with supporting safety analyses are contained herein.

Reasons for Change Request No. 103

A number of Technical Specification changes are required in order to reflect equipment, administrative, and analytic changes which must be implemented prior to restart of TMI-1. The purpose of this change request is to present all Technical Specification changes for restart of TMI-1. The proposed changes are as follows:

Change Request No. A -	Anticipatory Reactor Trips
Change Request No. B -	Operability of PORV and Block Valve, Position indications for PORV and Safety Valves, and Setpoints
Change Request No. C -	Emergency Power Supply Requirements (Pressurizer Heaters)
Change Request No. D -	Containment Isolation Modifications
Change Request No. E -	Instrumentation to Detect Inadequate Core Cooling
Change Request No. F -	Emergency Feedwater System Requirements
Change Request No. G -	TMI-1/TMI-2 Separation
Change Request No. H -	Setpoints Associated with ECCS Analyses

The above Change Requests are intended to satisfy the Technical Specification change requirements associated with the "Short Term" actions identified in the NRC's Order of August 1979. Technical Specifications associated with "Long Term" actions will be proposed at a later date consistent with the schedule for implementation of these items.

Safety Analysis Justifying Changes

The Technical Specifications associated with Change Requests A through H, together with supporting analyses, are presented in separate sections since

each request addresses a different safety issue and can be considered on its own technical merits.

Implementation

In order to provide an appropriate period for completion of equipment installation, system start-up and testing, and implementation of administrative requirements, the licensee requests an effective date of 90 days following issuance of this amendment, or criticality, whichever is later.

License Fee (10 CFR Part 70.22)

The enclosed Technical Specification Change Request is submitted without remittance since this change was requested by NRC under the August 9,1979 Order.

Implementation Time Period

It is requested that this Technical Specification Change Request be approved expeditiously in order that it may be implemented upon initial criticality for Cycle 5.

Technical Specification Change Request No.103A-Anticipatory Reactor Trips

The licensee requests that the attached revised pages replace pages 3-28, 3-30, 4-2, and 4-7 of the existing Technical Specifications, Appendix A to the Operating License. The licensee also requests that attached Page 4-2a be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

Item B-5 of IE Bulletin 79-05B requires licenses to provide for NRC approval a safety grade automatic anticipatory reactor scram using appropriate signals. Item B-7 of the same bulletin required the submittal of Technical Specifications for design changes, including those changes associated with Item B.5.

Safety Analysis Justify Change

The reactor trips on loss of feedwater or turbine trip are designed as anticipatory reactor trips which respond to equipment situations which would produce significant primary system pressure transients. By tripping the reactor on the anticipation of a pressure transient, (1) the probability of an overpressure trip is reduced and (2) the challenge rate to the pressurizer code safety valves is reduced. The challenge rate to the pressurizer power operated relief valve, a piece of equipment which is not credited in the safety analysis, is likewise reduced. The design of the reactor trips on loss of feedwater or turbine trip incorporates a 2-out-of-4 logic, is fully testable, and meets the single failure criterion of IEEE-279. A description and evaluation of these reactor trips is contained in Section 2.1.1.1 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1." The NRC accepted the design for TMI-1 in NUREG 0680 at page C2-14.

The reactor trip on loss of feedwater and the reactor trip on turbine trip* are of the same safety quality as other Reactor Protection System trip functions. The Limiting Conditions for Operation in proposed Technical Specifications 3.5.1.1 (Item 1 "Other Reactor Trips," Table 3.5-1) and the Surveillance Requirements in proposed Technical Specification 4.1.1 (Items 45 and 46 of Table 4.1-1) have therefore been chosen to be consistent with other Reactor Protection System functions, a "check" being required each shift, a "test" each month, and a "calibration" during each refueling outage. Bypass of the loss of feedwater trip below 7% power, and turbine trip below 20% power is permitted to allow normal reactor startup operations.

In conclusion, we have determined that, with regard to the reactor trip on loss-of-feedwater and the reactor trip on turbine trip:

(1) The probability or consequences of pressurization accidents previously evaluated have not been increased. The proposed trips are anticipatory

* These portions of the reactor trip functions are safety quality however their location (i.e. turbine building) does not permit a full safety grade status therefore isolation devices are used.

(2) Extended periods of operation without the delta-pressure or acoustic monitors will not preclude the reactor operator detecting a leaking or stuck-open safety or relief valve. Several indications of safety or relief valve discharge flow are available including safety and relief valve tailpipe (discharge) temperature.

In addition to delta-pressure and acoustic monitor Technical Specifications, Limiting Conditions for Operations are proposed to specify operability of the PORV and the associated block valve. Technical Specification 3.1.12.4 requires the block valve to be closed within one (1) hour of determining that the PORV is inoperable or be in at least hot shutdown within 6 hours. If the block valve becomes inoperable the PORV is to be closed **within one (1)** hour and power removed from the PORV. The above actions are appropriate since the PORV has been shown to be a potential path for reactor coolant system depressurization; however, concurrent failure of both the PORV and the block valve is unlikely, and thus there is a high degree of certainty that the PORV discharge line will be isolated if required.

IE Bulletin 79-05B, item B.3, recommended a decrease in the Reactor Protection System (RPS) high pressure trip setpoint to reduce the challenge rate to the PORV. Analyses indicate that the RPS high pressure trip setpoint should be reduced to 2300 psig. The decrease in the RPS high pressure trip, from 2390 to 2300 psig is incorporated in proposed Technical Specification 2.3.1.

The Surveillance Requirements associated with the delta-pressure and acoustic safety and relief valve monitors are contained in proposed Technical Specification 4.1.1 (Items 47a and b in Table 4.1-1); these monitors are to be checked each shift and tested/calibrated each refueling period. The "check" and "test" surveillance need not be performed when the average reactor coolant system temperature (Tave) is below hot shutdown since this function would be unnecessary. Accessibility considerations, as noted previously are significant with regard to the delta-pressure and acoustic monitors and are a consideration in determining the test/calibration interval.

A Surveillance Requirement (setpoint test) for the pressurizer Electromatic relief valve is incorporated in proposed Technical Specification 4.1.2 (Item 48 of Table 4.1-1); a refueling period interval has been selected to be consistent with the pressurizer safety valve surveillance interval. Functional testing of the PORV Block Valve is specified as quarterly in Technical Specification Table 4.1-2, item 11.

Conclusion

In conclusion, with regard to the additional requirements for the delta-pressure and acoustic monitors, PORV and Block Valve operability, and the setpoint requirements for the pressurizer PORV:

(1) The probability or consequences of accidents, previously evaluated have not been increased. The requirement for PORV and Block Valve Operability and operability and surveillance of the safety and relief valve

and have not been taken credit for in the accident analyses. The reactor trip on overpressure and the pressurizer Code Safety Valves remain the principal means of mitigating pressure transients.

Loss of both feedwater pumps always resulted in a reactor trips on high pressure. Reactor trip will still occur but with reduced consequences. However, it is anticipated that an increased number of plant trips will occur as a result of the anticipatory trip on loss of turbine. This increased number of trips increases the challenge rate to plant systems needed to respond to plant trips. The NRC has reviewed this increased challenge rate and concluded that on balance the reduction in pressurization challenges offsets the negative affects of an increased number of plant trips.

(2) No accidents, other than those previously considered, will be introduced. The additional reactor trips have been designed so as not to introduce additional failure modes into the Reactor Protection System or other safety equipment. Moreover, by anticipating significant pressure transients, the challenge rate to the overpressure trip and pressure relieving capacity has been reduced.

(3) No safety margins have been reduced. Since the additional reactor trips scram the reactor on anticipation of significant pressure transients, the peak pressure associated with these transients can be expected to decrease which would result in an increase in safety margins as a result of postulated turbine trip or loss-of-feedwater transients.

For these reasons, we conclude that implementation of the design for additional reactor trips, and adoption of associated Technical Specifications, does not involve unreviewed safety questions.

Technical Specification Change Request No.103B - Operability of the PORV and Block Valve, Position Indicators for the PORV and Safety Valves, and Setpoints

The licensee requests that the attached revised pages replace pages 3-18c, 4-1, 4-6 and 4.8 of the existing Technical Specifications, Appendix A to the Operating License. The licensee also requests that attached pages 3-18d, 3-40a, 3-40b and 3-40c be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

Section 2.1.3 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979, contains NRC recommendations on installations of valve position indications for safety and relief valves. The guidance on safety and relief valve position indications is to, "Provide in the control room either a reliable, direct position indication for the valves or a reliable flow indication device downstream of the valves."

Safety Analysis Justifying Change

In response to the recommendations in Section 2.1.3 of NUREG-0578, a system of safety and relief valve position indications has been designed. The safety and rel ef valve position indication system, described in Section 2.1.1.2 of "Repor in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1", consists of the following:

(1) delta-pressure taps, and monitors, at discharge piping elbows downstream of the safety and relief valves, and

(2) acoustic monitor (accelerometer) mounted on the pressurizer power operated relief valve (PORV).

The above sensors are in addition to the tailpipe thermocouples and reactor coolant drain tank instrumentation which are presently installed.

Technical Specifications, Limiting Conditions for Operation and Surveillance Requirements, are for the safety and relief valve position instrumentation. The Limiting Conditions for Operation, contained in Technical Specification 3.5.5, requires the three delta-pressure monitors and an acoustic monitor to be operable. The remedial action to be taken if one or more delta-pressure monitors or the acoustic monitor becomes inoperable is to require operability of an alternate indication of safety or relief valve positions (tail pipe thermocouples) and repair the monitors prior to startup following the next cold shutdown. This requirement is based upon the following considerations:

(1) The sensors for the delta-pressure and acoustic monitors are located inside containment and would most likely require containment entry for repair or replacement.

monitors increases the probability that malfunction of the relief or safety valves will be detected, remedial action taken, and thus reduces the consequences associated with malfunctioning.

(2) No accidents, other than those previously considered, will be introduced. The delta-pressure and acoustic instrumentation have no automatic functions and therefore cannot by themselves change the course of any accident or transient; sufficient confirmatory information is available in the control room to detect improper functioning of these monitors. The requirements for periodic testing of the setpoint will enhance the availability of the PORV.

(3) No safety margins have been reduced. Although the setpoint of the pressurizer PORV has been increased, no credit was taken for this equipment in the safety analyses. The decrease in the RPS high pressure trip setpoint will cause the reactor to trip earlier in the course of significant presure transients and thus reduce the peak presure during the transient.

For reasons presented above, implementation of the design changes associated with the delta-pressure and acoustic monitors and associated Technical Specifications, including those addressing the setpoint of the pressurizer PORV, do not involve any unreviewed safety considerations.

Technical Specification Change Request No.103C - Emergency Power Supply Requirements (Pressurizer Heaters)

The licensee requests that the attached revised pages replace pages 3-6. 3-7, 4-46, and 4-47 of the existing Technical Specifications, Appendix A to the operating license. The licensee also requests that attached pages 3-6a be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

IE Bulletin 79-05B requires licensees of operating reactors to develop procedures and train personnel to "...assure availability of adequate capacity of pressurizer heaters, for pressure control and maintain primary system pressure to satisfy the subcooling criterion for natural circulation." Section 2.1.1 of NUREG-0578, "TMI-1 Lessons Learned Task Force Status Report and Short-Term Recommendations," goes further in that it recommends that reactors, "Provide redundant emergency power for the minimum number of pressurizer heaters required to maintain natural circulation conditions in the event of loss of offsite power."

Safety Analysis Justifying Change

Section 2.1.3 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" describes design changes, and operator actions, that are required to supply an adequate number of pressurizer heaters from either of two independent engineered safeguard power sources in the event offsite power is lost. The manual transfer of power from the normal (balance of plant) to the onsite (engineered safeguards) power source involves the use of a "Kirk Key" system that assures proper transfer of power from the normal to the onsite power source. With regard to operability of pressurizer heaters and emergency power supplies; proposed Technical Specification 3.1.3.4 requires two pressurizer heater groups of at least 107 kw (each) to be operable. In addition, proposed Technical Specification 4.6.3 requires that the manual transfer of power from normal to backup supply, be demonstrated during each refueling outage.

In the event that an engineered safeguards actuation signal is received while the pressurizer heaters are powered from the diesel generators, the pressurizer heater loads are automatically shed from the diesel generators to prevent overloading of the diesels. To assure proper load shedding (breaker operation) of the pressurizer heaters, from the diesel generators upon an engineered safety feature actuation signal, a test of the engineered safety features pressurizer heater supply breaker will be undertaken on a periodic basis. Technical Specification 4.6.1.b requires a test of the diesel generators, during each refueling shutdown, to determine proper automatic response under loss of normal AC power conditions concurrent with an engineered safety features actuation signal. Technical Specification 4.6.1.b also includes a requirement to confirm proper trip of the engineered safety features pressurizer heater supply breakers upon receipt of an engineered safety features actuation signal.

In conclusion, with regard to the provisions for transfer of pressurizer heater loads from normal to back-up power supplies:

(1) The probability or consequences of accidents, previously evaluated, have not increased. Periodic testing of the engineered safety features pressurizer heater supply breaker will assure that, in the event that the pressurizer heaters are powered by the diesel-generators when an engineered safety features actuation signal is received, the pressurizer heaters will be shed from the diesel generators supply busses.

(2) No accidents, other than those previously considered, will be introduced. The manual transfer of the pressurizer heater supply loads, in the correct manner, is assured by the "Kirk Key" system. Periodic testing of the engineered safety features pressurizer heater supply breaker will prevent diesel overloading in the event that the pressurizer heaters are powered by the diesel generators when an engineered safety features actuation signal is received.

(3) No safety margins have been reduced. The availability of the pressurizer heaters on loss of offsite power provides additional assurance that primary system subcooling margin can be maintained such that natural circulation will be enhanced.

Based upon the above, we conclude that plant modifications necessary to allow manual transfer of selected pressurizer heater loads, from normal to backup power sources, and adoption of associated Technical Specifications, do not involve an unreviewed safety question.

NOTE: Technical Specification Change Request No. 60A (November 7, 1980) information is included with that of Technical Specification Change Request No. 103C.

Technical Specification Change Request No. 103D - Containment Isolation Modifications

The licensee requests that the attached revised pages replace pages 3-32 and 4-5 of the existing Technical Specifications, Appendix A to the operating license. The licensee also requests that page 3-32a be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

IE Bulletin 79-05A requires licensees of operating B&W facilities to, "Review the containment isolation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection." Section 2.1.4 of "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979, expanded the requirements of I&E Bulletin 79-05A, as follows:

"Provide containment isolation on diverse signals in conformance with Section 6.2.4 of the Standard Review Plan, review isolation provisions for non-essential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening upon reset of the isolation signal."

Safety Analysis Justifying Change

Section 2.1.1.5 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" provides the details and evaluation of a redesigned containment isolation system with the following new features:

(1) Partial containment isolation on reactor trip,

(2) Partial containment isolation on 30 psig building pressure

(3) Specific line isolation on high radiation

(4) Specific line isolation on rupture detection concident with 1600 psig ESAS (to be added after restart due to equipment delivery)

(5) Specific line isolation on 1600 psig RCS pressure (ESAS actuation)

With regard to the revised containment isolation design, this design meets the NRC's requirements in that:

(1) The system initiates automatically on safety injection (IE Bulletin 79-05A) - The reactor trip signal is utilized to obtain a diverse isolation signal. Since the RPS trips the reactor on low pressure (1800

psig)* prior to the safety injection signal (1600 psig), an RPS trip signal on low pressure will always preceed a safety injection signal. The reactor trip signal, therefore, isolates the containment more quickly than a safety injection signal.

(2) The system is diverse (NUREG-0578) - The redesigned containment isolation system provides containment isolation on the following signals:

(a) Reactor trip

(b) High radiation (individual line isolation)

(c) Rupture detection coincident with 1600 psig ESAS (individual line isolation)**

(d) The 1600 psig Engineered Safeguards Actuation Signal (ESAS)

(e) The 30 psig containment signal

(f) The 4 psig containment signal (to be removed on certain lines when (c) is implemented.

(g) Manual closure

(3) Following isolation, lines should not be vulnerable to inadvertent reopening (NUREG-0578). Overriding the containment isolation signal does not open the containment isolation valves, deliberate operator action is required to reopen selected individual valves.

Technical Specifications, described herein, provide Limiting Conditions for Operation and Surveillance Requirements for the additional containment isolation functions. Limiting Conditions for Operation for containment isolation on the RPS Trip and the 30 psig containment pressure have been incorporated into Technical Specification 3.5.1.1 (Items 3.c and 3.d of Table 3.5-1). The operability of the Reactor Building Purge Isolation (on high radiation) is required by Item 1, Reactor Building Isolation, in Technical Specification Table 3.5-1. The minimum channel operability for containment isolation on RPS trip, and on Reactor Building 30 psig, have been chosen to be the same as the existing containment isolation functions; this would require a minimum of two channels to be operable or place the reactor in hot shutdown.

With regard to Surveillance Requirements, surveillance for containment isolation on RPS trip, and on Reactor Building 30 psig, have been incorporated into Technical Specification 4.1.1 (Items 19.c and 19.d of Table 4.1-1) and

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* Technical Specification Change Request No.103 proposes an increase in the low pressure trip setpoint from 1800 psig to 1900 psig.

** To be added after restart due to equipment delivery problems.

chosen to be the same as the existing containment isolation system surveillance requirements for the Reactor Building 4 psig signal; this requires a channel check each shift, testing each month, and calibration each refueling period.* Surveillance Requirements for containment and individual line isolation on high radiation are provided for in Technical Specification 4.1.1 (Item 28, "Radiation Monitoring Systems," Table 4.1-1). The "check" and "test" surveillances are required to be performed only when containment integrity is required. This provision deletes surveillance requirements during extended outages when containment isolation is not needed.

Conclusion

In conclusion, with regard to the revised containment isolation design and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The increased diversity of the containment isolation signals increases the probability and timeliness of containment isolation.

(2) No accidents, other than those previously considered, will be introduced. The revised containment isolation design does not in any way hamper the function of systems designed to mitigate the consequences of postulated accidents. Spurious initiation of any of the additional containment isolation signals would not isolate any components that would not also be isolated by a spurious initiation of the containment isolation system before modification.

(3) No safety margins have been reduced. The plant safety features required to mitigate the consequences of postulated transients and accidents are not hampered by the revised containment isolation design.

Based upon the above, we conclude that the modifications associated with the revised containment isolation design, and associated Technical Specifications, do not involve any unreviewed safety questions.

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* The reactor trip containment isolation function is not calibrated since no analog function is involved.

Technical Specification Change Request No. 103E - Instrumentation to Detect Inadequate Core Cooling

The licensee requests that the attached revised page replace page 4-7 of the existing Technical Specifications, Appendix A of the operating license. The licensee also requests that the attached pages 3-40a, 3-40b and 3-40c be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

Section 2.1.3b of Appendix A to "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979, requires that:

> "...each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is (SIC) not to be used exclusive of other related plant parameters."

Section 2.1.1.6 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" contains a description of a saturation margin meter which is proposed for installation at TMI-1.

Safety Analysis Justifying Changes

The saturation margin monitor will display, in the control room, the margin between the actual primary plant temperature (T_H) and the saturation temperature (T_{sat}) for the existing plant pressure. The T_{sat} will be computed using primary system pressure measurements and will be compared to the wide range T_H instrument reading. The temperature margin will be displayed in the control room. An alarm will be initiated if the margin falls below a pre-set value. Redundancy will be provided by computing T_{sat} margin independently for each reactor coolant loop. The higher temperature for each loop will automatically be selected for the computations.

Limiting Conditions for Operation and Surveillance Requirements are presented herein for the saturation margin monitor. Technical Specification Table 3.5-2, "Accident Monitoring Instrumentation" requires the saturation margin monitor to be operable during start-up and power operation. If the saturation margin monitor is not operable, the reactor operator determines saturation margin manually. This remedial action is appropriate since (1) no automatic actuations of safety features are associated with the saturation monitor and (2) saturation temperature is easily calculated using reactor coolant system measurements and "steam tables." Surveillance Requirements for the saturation margin monitor to be checked each shift, tested monthly, and calibrated each refueling period are given in Item 49,

Technical Specification Table 4.1-1. The "check" surveillance is only required when above hot shutdown. The surveillance schedule is consistent with surveillance schedules for safety grade instrumentation at TMI-1 and is sufficient to assure reliable performance from the monitor.

Conclusion

In conclusion, with regard to the saturation margin monitor and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The saturation margin monitor is not required for the prevention or mitigation of accidents, or transients, previously considered.

(2) No accidents, other than those previously considered, will be introduced. No automatic actuations of safety features are associated with the saturation margin monitor.

(3) No safety margins have been reduced. The saturation margin monitor is not associated with any safety margins; both low pressure and high temperature RPS trips protect the reactor's thermal margins.

Based upon the above, we conclude that the installation and use of the saturation margin monitor, and the associated Technical Specifications, do not involve any unreviewed safety questions.

Technical Specification Change Request No. 103F - Emergency Feedwater System Requirements

The licensee requests that the attached revised pages replace pages 3-25, 3-26, 4-7, 4-8, 4-39, 4-40 and 4-52 of the existing Technical Specifications, Appendix A of the Operating License. The licensee also requests that the attached pages 3-26a, 3-32a, 3-40a, 3-40b, 3-40c, 4-52a and 4-52b be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

By letter dated June 28, 1979, Met-Ed presented NRC with modifications to TMI-1 which would be completed prior to restart of TMI-1. The June 28, 1979 letter recommended the following changes to the emergency feedwater (EFW) system and associated procedures:

1. Automatic initiation of the motor driven EFW pumps upon loss of both feedwater pumps or loss of four (4) Reactor Coolant Pumps.

2. Modification of the EFW control valves such that they fail open on loss of control air.

3. Automatic block loading of the motor driven EFW pumps on the emergency Diesel Generators.

4. Incorporation of EFW in the TMI-1 Technical Specifications as specified in IE Bulletin 79-05A, item 8. Verification that Technical Specification requirements of EFW capacity are in accordance with the accident analysis will be conducted.

5. Provide indication in the control room of EFW flow to each Steam Generator.

6. Provide procedures and training to assure that EFW is available and properly applied when required. Procedures will identify the need to verify proper operation when EFW is initiated.

7. To assure that EFW will be aligned in a timely manner to inject on all EFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary location in communications with the control room during the surveillance mode to carry out alignment changes necessary upon EFW demand events.

8. Design review and modifications, as necessary, will be conducted to provide control room annunciation for all auto start conditions of the EFW system.

On August 9, 1979 the NRC issued an "Order and Notice of Hearing" which addressed modifications to the TMI-1 facility. With regard to those changes proposed for the emergency feedwater system in the June 28, 1979 letter, the

August 9, 1979 Order directed that these changes should be made. A description and evaluation of changes to the emergency feedwater system, involving equipment modifications (items 1,2,3,5 and 8, as described above) are contained in Section 2.1.1.7 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1." Technical Specifications for the modified emergency feedwater system are discussed herein.

TMI-1 Technical Specification 3.4 provides Limiting Conditions for Operation for the emergency feedwater system. Guidance on operability of the emergency feedwater System is contained in IE Bulletin 79-05A, Item 8, as follows:

> "Prepare and implement immediately procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.
>
> When at least one 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generator for cooling within 12 hours or at the maximum safe shutdown rate."

The guidance contained in IE Bulletin 79-05A has been incorporated in Technical Specification 3.4.1. Technical Specifications 3.4.3 and 3.4.6 have been rewritten to incorporate remedial action in the event that the condensate storage tanks and/or the main steam safety valves are inoperable. For both the condensate storage tank and the main steam safety valves, remedial action is consistent with NRC guidance as reflected in the B&W Standard Technical Specifications. Operability requirements for the automatic EFW initiation channels are contained in Technical Specification Table 3.5-1, Item 2, "Emergency Feedwater System." Operability of the EFW flow instrumentation is required by Technical Specification 3.5.5 (Table 3.5-2).

With regard to surveillance requirements, proposed Technical Specification are presented as follows:

(1) Technical Specification 4.9.1.1 which requires testing of the emergency feedwater pumps every three months, as modified, would require pump testing every 31 days and also require verification of specific pump flow values during the testing. The flow testing would be based the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Article IWP-3210.

(2) Technical Specification 4.9.1.2 requires that, during testing of the EFW, if one steam g erator flow path is made inoperable, a qualified, dedicated individual in communication with the control room, be maintained at the EFW manual valves. In the event that EFW is required to function, the individual would promptly realign the manual valves to operational positions.

(3) Technical Specification 4.9.1.3 requires valve verification (correct position and locked, if appropriate) for valves in the Emergency Feedwater System, every 31 days. In addition, locked valves could only be maintained in an unlocked condition under administrative control.

(4) Technical Specification 4.9.1.4 requires a test, quarterly of automatic pump start logic and automatic valve lineup following an Emergency Feedwater actuation signal. In addition, the operability of the manual control valve station will be verified.

(5) Technical Specification 4.9.1.5 requires testing of the EFW control valves on a quarterly basis.

(6) Item 10F of NRC's October 26 1979 letter to Mr. R. C. Arnold requires a, "...Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary EFW system water source to the steam generators. The flow test should be conducted with EFW system valves in their normal alignment." This test is incorporated in Technical Specification 4.9.1.6.

(7) Item 50 of Technical Specification 4.1.1 requires monthly testing, and calibration each refueling period, for the Emergency Feedwater flow instrumentation. The surveillance would not be required when T_{ave} is less than 250°F since the reactor would be shutdown and this safety function not needed.

(8) Technical Specification 4.5.1.1 incorporates the motor driven feedwater pumps into the list of equipment whose operation is verified during the testing of the Emergency Diesel Generators. In this case, an additional test signal is required to start the pumps since the pumps do not start directly on loss of AC power.

(9) With regard to surveillance on the automatic initiation circuitry for the Emergency Feedwater System, Technical Specification 4.1.1 (Table 4.1-1, Item 51) provides test and calibration frequencies.

Conclusion

In conclusion, with regard to the modifications to the emergency feedwater systems and associated Technical Specifications:

(1) The probability or consequences of accidents previously evaluated have not increased. The more conservative Limiting Condition for Operation and Surveillance Requirements for the Emergency Feedwater System provide increased assurance that the system will operate properly, when required.

(2) No accidents, other than those previously considered, will be introduced. The modifications to the Emergency Feedwater System could only effect the non-operation or spurious operation of the system; both of these conditions have been previously evaluated. The only aspect of the modification with the potential for effecting other systems involves the loading of the motor driven feedwater pumps on the emergency Diesel Generators. An analysis of the diesel generator loading indicates that the maximum load, with the Emergency Feedwater pumps is less than the 2000 hour rating of 3000 Kw. The proper Diesel Generator loading sequence with the emergency feedwater pumps will be verified prior to startup and every 18 months thereafter. Other aspects of the Emergency Feedwater System will be tested prior to startup, and periodically thereafter.

(3) No safety margins have been reduced. The modifications to the Emergency Feedwater System did not involve any changes which resulted in a decrease in capacity of this system to perform its designed function.

Based upon the above, we conclude that the modifications to the Emergency Feedwater System, and associated Technical Specifications, do not involve any unreveiwed safety questions.

Technical Specification Change Request No. 103G - TMI-1/TMI-2 Separation

The licensee requests that the attached revised pages replace page 6-17 of the existing Technical Specifications, Appendix of the operating license. The licensee also requests that the attached pages 3-95, 3-96, and 4-8a be added to Appendix A. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

Item II.4 of the NRC's August 9, 1979 "Order and Notice of Hearing," requires that, "The licensee shall demonstrate that decontamination and/or resotration operations at TMI-2 will not affect safe operations at TMI-1. The licensee shall provide separation and/or isolation of TMI 1/2 radioactive liquid transfer lines. Fuel handling areas, ventilation systems, and sampling lines."

Section 7.2 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" describes a plan to separate TMI-1/TMI-2 interfaces tnat have the potential of transferring significant quantities of contamination as a result of restoration activities at TMI-2.

Safety Analysis Justifying Changes

The major pathways for potential transfer of contamination from TMI-2 to TMI-1 are the waste management interconnections. The following TMI-1/TMI-2 waste management interconnections have been identified:

(1) Unit 2 Reactor Coolant Bleed Holdup Tank to the Unit 1 Reactor Coolant Waste Evaporator.

(2) Unit 1 Miscellaneous Waste Evaporator to the Unit 2 Evaporator Condensate Test Tank.

(3) Unit 2 Neutralizer Tanks, Contaminated Drain Tanks, Reactor Coolant Bleed Holdup Tanks, Auxiliary Building Sump Tanks and Miscellaneous Waste Holdup Tanks to the Unit 1 Liquid Waste Disposal System.

(4) Unit 1 Evaporator Concentrate to the Unit 2 Evaporator Concentrate.

(5) Unit 1 Spent Ion Exchange Resin to the Unit 2 Spent Ion Exchange Resen.

Technical Specification 4.1.2 (Table 4.1-2, Item 13) requires the isolation devices (valves, blank flanges, etc.(on the above interconnections to be verified to be isolated, by visual inspection, on a semiannual basis. Technical Specification 3.19 requires that, if an isolation device is found to be open without prior NRC authorization, a "Thirty Day Written Report" must be prepared. In addition, proposed Technical Specification 3.19.2 requires NRC approval prior to creation of additional TMI-1/TMI-2 system interconnections that can transfer potentially significant quantities of contamination.

In conclusion, the proposed TMI-1 Technical Specifications 3.19 and 4.1.2 for the TMI-1/TMI-2 interconnections provide assurance that:

(1) System interconnections that could potentially transfer significant quantities of contamination from TMI-1 to TMI-2 will remain closed.

(2) If permission is received from the NRC to open system interconnections, these interconnections will be used in accordance with approved procedures.

(3) No new system interconnections, with the potential for transferring significant quantities of contamination from TMI-2 to TMI-1, will be created without prior NRC approval.

The above controls limit releases from TMI-1 to materials under control at TMI-1 and thus to previously evaluated quantities and concentrations of contamination.

Technical Specification Change Request No. 103H - Setpoints Associated with ECCS Analyses

The licensee requests that the attached revised pages replace pages 2-7, 2-8, 2-9, 2-9a, and 3-37 of the existing Technical Specifications, Appendix A of the operating license. Margin bars on the enclosed pages indicate changes from existing Technical Specifications.

Reason for Change Request

The Low Reactor Coolant System Pressure Channel setpoint, which is used as input to the ESAS logic, is determined based on a generic LOCA analysis. The generic LOCA analysis for TMI, referenced as "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, has referenced the Low Reactor Coolant System Pressure setpoint as 1600 psig compared with the Technical Specification value of 1500 psig. (The value actually used in the BAW-10103 calculations, however, was 1350 psig to conservatively account for instrument inaccuracy.)

In addition, Technical Specification 2.3.1 (Table 2.3-1, Figure 2.3-1) provides a value of 1800 psig for the RPS Low Reactor Coolant Pressure trip setpoint. The B&W generic ECCS analysis, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103, Rev. 2, April 1976, referenced a value of 1900 psig for the Low Reactor Coolant System Pressure Trip setpoint based on trip on variable pressure/temperatures at full power. To cover operation at less than full power Technical Specification 2.3.1 increase the Low Reactor Coolant System Pressure Trip Setpoint from 1800 psig to 1900 psig.

Safety Analysis Justifying Changes

Revised Technical Specification 3.5.3.1 requires the Low Reactor Coolant System Pressure HPI/LPI initiation setpoint to be raised to 1600 psig. In the event of a LOCA, the only impact of the 100 psig increase in the minimum Low Reactor Coolant System Pressure setpoint would be to initiate actions, based on this signal, at an earlier time in the accident (e.g., both HPI and LPI pumps would start earlier in the accident.)

With regard to the Low Reactor Coolant Pressure trip setpoint the principal reason this setpoint is to maintain thermal margins for the fuel by preventing the minimum DNB ratio from decreasing below the safety limit of 1.3; the transient analysis for TMI-1 is based on an 1800 psig Low Reactor Coolant Pressure trip setpoint. The Low Reactor Coolant Pressure trip setpoint is also credited in the ECCS analysis since a reactor trip is part of the assumed LOCA scenario. By increasing the Low Reactor Coolant Pressure setpoint from 1800 psig to 1900 psig, the reactor would trip earlier in the LOCA scenario. Increasing the Low Reactor Coolant System trip setpoint has the effect of increasing the margin to DNB following a trip on low pressure; the reactor would trip earlier on low pressure and thus the final minimum DNB would be higher (more conservative) than if the reactor tripped at 1800 psig.

In conclusion, with regard to increasing the Low Reactor Coolant Pressure trip (RPS) setpoint from 1800 psig to 1900 psig and increasing the minimum Low Reactor Coolant System (ESAS) channel setpoint from 1500 psig to 1600 psig:

(1) The probability or consequences of accidents previously considered have not increased. For any accident that involves a pressure decrease, the reactor will trip earlier in the transient, and the ECCS system initiated earlier in the transient, and thus the result of the accident will be less severe.

(2) No accidents of a type not previously evaluated will occur. The change in the Low Reactor Coolant System Pressure Channel Setpoint and the Low Reactor Coolant Pressure trip setpoint would have only a small impact on the timing of initiation of safety features.

(3) No safety margins have been reduced. Raising the minimum setpoint to 1600 psig would slightly increase the LOCA margins. With regard to the Low Reactor Coolant Pressure trip setpoint, for pressure reduction transients, the minimum DNB following the reactor trip will be higher (more conservative) and for the LOCA, the peak clad temperature and other system parameters will be more favorable.

Based upon the above, we conclude that increasing the Low Reactor Coolant System trip setpoint from 1800 psig to 1900 psig, and increasing the minimum Low Reactor Coolant System Pressure setpoint, does not involve unreviewed safety questions.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10 % above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to value $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of < 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

References

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

(6) Technical Specification Change Request No. 31, January 16, 1976, and Technical Specification Change Request No. 84, June 23, 1978.

(7) "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103, Rev. 2, Babcock and Wilcox, April 1976.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip set point. The trip setting limit for high reactor coolant system pressure has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (6)Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

The high pressure trip setpoint was subsequently lowered from 2390 psig to 2300 psig. The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

The low pressure (1900 psig) and variable low pressure (11.75 T_{OUT} 5103) trip setpoint were established to maintain the DNB ratio greater that or equal to 1.3 for those design accidents that result in a pressure reduction (3,4). The B&W generic ECCS analysis however, assumed a low pressure trip of 1900 psig and is therefore the basis of low pressure reactor trip. Figure 2.3-1 shows the high pressure, low pressure, and variable low pressure trips.

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RTD is 520° to 620°F. The trip setpoint of the channel is 619°F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1°F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip set point of 620°F.

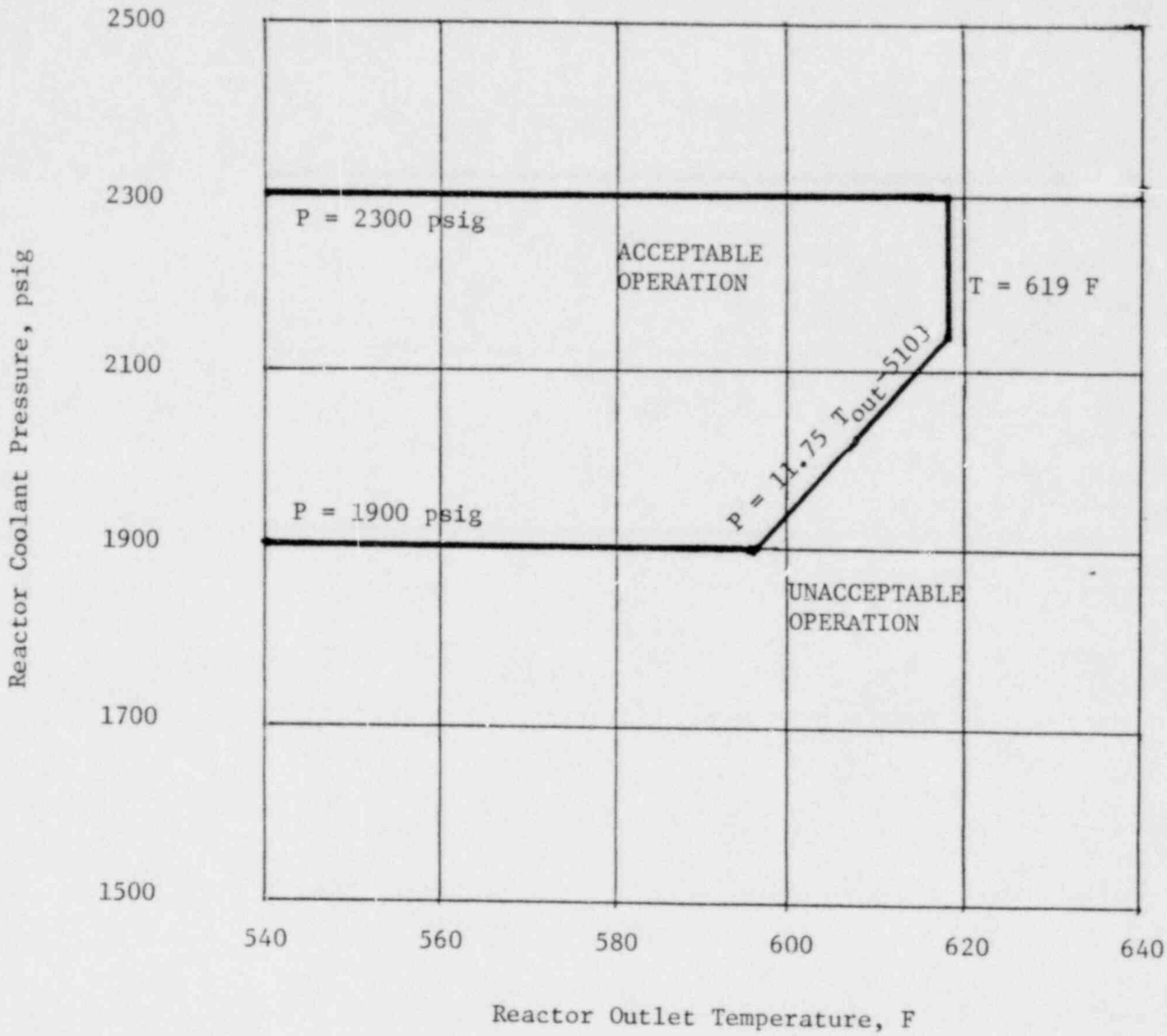
The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

TABLE 2.3

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow (2) and imbalance max. of rated power	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (5) on pump monitors, Max. % of rated power	NA	NA	91%	Bypassed
4. High reactor coolant system presure, psig max.	2300	2300	2300	1720(4)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Variable low reactor coolant system pressure psig, min.	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	Bypassed
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only during reactor shutdown
(4) Automatically set when other segments of the RFS (as specified) are bypassed
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation
(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable connectors.



TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1