UNITED ST
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WASHINGTON, C

Reference 39

June 4, 1979

LER MONTHLY REPORT

The enclosed computer listing, as identified on the attached sheet, provides information concerning Licensee Event Reports (LERs) entered into the data base during the month of May.

If you desire additional information or special searches, please feel free to contact me on 301/492-7785.

Sincerely,

Eugenia L. Boyle

Eugenia L. Boyle
Automated Systems Branch, DTS
Office of Management and
Program Analysis

Enclosure:
As stated

8008110 253

FACILITY/SYSTEM/COMPONENT/ COMPONENT SUBCODE/CAUSE CODE/ CAUSE SUBCODE/MANUFACTURER	DOCKET NO./ LER NO./ CONTROL NO.	EVENT DATE/ REPORT DATE/ REPORT TYPE	EVENT DESCRIPTION/ CAUSE DESCRIPTION
THREE MILE ISLAND-1 RESIDUAL HEAT REMOV SYS + CONT VALVES GATE COMPONENT FAILURE MECHANICAL WALWORTH CO.	05000289 79-005/03L-0 025501	022579 032279 30-DAY	DURING THE REFUELING OUTAGE PERFORMING LEAKAGE SURVEILLANCE ON DECAY HEAT REMOVAL SYSTEM TOTAL MEASURED LEAKAGE OF 8.9 GAL/HR. EXCEEDED T.S. SECTION 4.5.4.1. LIMIT OF 6.0 GAL/HR. EVENT REPORTABLE PER T.S. SECTION 6.9.2.B(4). EXCESSIVE LEAKAGE FROM VALVE PACKING GLANDS IN VALVES DH-V-15 A/B, DH-V-6A, DH-V-5A AND BS-V-3B. BORON WAS REMOVED FROM VALVE GLANDS AND PACKING GLANDS WERE ADJUSTED. LEAKAGE WAS VERIFIED WITHIN T.S. LIMITS.
THREE MILE ISLAND-2 MAIN STEAM SYSTEMS + CONTROLS VALVES GATE DEFECTIVE PROCEDURES NOT APPLICABLE ITEM NOT APPLICABLE	05000320 78-069/99X-0 023543	120278 022879 OTHER	WHILE IN MODE 1 ON DECEMBER 2, 1978, WHILE SWITCHING FROM THE STARTUP TO THE MAIN FEEDWATER REGULATING VALVES, A REACTOR TRIP OCCURRED FOLLOWED BY SAFETY INJECTION ACTUATION DUE TO OVERFEEDING THE STEAM GENERATORS. SINCE SAFETY FEATURE SYSTEMS FUNCTIONED AS DESIGNED, THIS EVENT DID NOT AFFECT THE HEALTH AND SAFETY OF THE PUBLIC. THIS EVENT OCCURRED DUE TO THE MAIN FEEDWATER REGULATING VALVE BEING PINNED OPEN. PROCEDURES HAVE BEEN REVISED TO PRECLUDE REOCCURRENCES.
THREE MILE ISLAND-2 ENGNRD SAFETY FEATR INSTR SYS INSTRUMENTATION + CONTROLS SWITCH OTHER NOT APPLICABLE BARTON INSTRU CO., DIV OF ITT	05000320 79-008/03L-0 025504	011779 020979 30-DAY	DURING INSPECTION OF EQUIPMENT & CABLES IN CONTROL BUILDING AREA ON 1/17/79 DISCOVERED SETPOINTS OF 2 FEEDWATER LINE RUPTURE DETECTION PRESSURE SWITCHES (FW-DPIS-7883-1 & FW-DPIS-7883-2) OUTSIDE T.S. ALLOWABLE LIMITS SPECIFIED IN SECTION 3.3.2.1 (196 PSID VS 192 PSID). NO EVENT OCCURRED SUBSEQUENT TO OUT-OF-TOLERANCE CONDITION OF SWITCHES WHICH WOULD HAVE REQUIRED THEM TO BE OPERABLE, AND SINCE VARIANCE FROM LIMIT WAS ONLY 2% NO EFFECT ON PUBLIC HEALTH AND SAFETY. INSTRUMENT SETTINGS MAY HAVE CHANGED FROM INSTRUMENT DRIFT OR STEAM LEAKAGE. CALIBRATION OF THESE INSTRUMENTS WILL BE CHECKED IN FUTURE TO DETERMINE DRIFT CHARACTERISTICS. PRESENT PLAN IS TO REPLACE SWITCHES DURING FEEDWATER ISOLATION MODIFICATION SCHEDULED FOR FIRST REFUELING. SWITCHES RECALIBRATED AND TESTED SATISFACTORILY.
THREE MILE ISLAND-2 STATION SERV WATER SYS + CONT COMPONENT CODE NOT APPLICABLE SUBCOMPONENT NOT APPLICABLE DEFECTIVE PROCEDURES NOT APPLICABLE ITEM NOT APPLICABLE	05000320 79-007/03L-0 025343	012679 022679 30-DAY	IN MODE 5 TRAVELLING WATER SCREENS WERE FOUND INOPERABLE DUE TO SIGNIFICANT BUILD UP OF DEBRIS CAUSING A HIGH DIFFERENTIAL LEVEL ACROSS THE IDLE SCREEN SYSTEM. BECAUSE NO EVENT OCCURRED WHICH REQUIRED EMERGENCY USE OF RIVER WATER SYSTEMS AND BECAUSE SUFFICIENT FLOW TO THE RIVER WATER PUMP IN OPERATION AT THE TIME EXISTED, THIS EVENT DID NOT HAVE AN ADVERSE EFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC. PROCEDURES DID NOT REQUIRE ONE OF THE SCREENS TO BE CONTINUOUSLY OPERABLE DURING PERIODS WHEN LARGE AMOUNTS OF DEBRIS ARE PRESENT IN THE RIVER. AFFECTED SCREENS WERE CLEANED AND RETURNED TO SERVICE. PROCEDURES TO BE CHANGED TO ENSURE AT LEAST ONE SCREEN REMAINS IN CONTINUOUS SERVICE DURING PERIODS OF HIGH DEBRIS ON THE RIVER.

FACILITY/SYSTEM/COMPONENT/ COMPONENT SUBCODE/CAUSE CODE/ CAUSE SUBCODE/MANUFACTURER	DOCKET NO./ LER NO./ CONTROL NO.	EVENT DATE/ REPORT DATE/ REPORT TYPE	EVENT DESCRIPTION/ CAUSE DESCRIPTION
THREE MILE ISLAND-2 RESIDUAL HEAT REMOV SYS + CONT COMPONENT CODE NOT APPLICABLE SUBCOMPONENT NOT APPLICABLE DEFECTIVE PROCEDURES NOT APPLICABLE ITEM NOT APPLICABLE	05000320 79-009/03L-0 025333	013079 022679 30-DAY	PREPARING TO ENTER MODE 4 FOUND THAT SURVEILLANCE REQUIRED BY T.S. 3.1.2.1 FOR MODE 5 HAD NOT BEEN PERFORMED AFTER MAKEUP PUMPS HAD BEEN TAGGED OUT SUBSEQUENT TO ENTRY INTO MODE 5. BECAUSE NO CORE ALTERATIONS WERE PERFORMED OR POSITIVE REACTIVITY CHANGES MADE, THIS EVENT DID NOT HAVE AN ADVERSE EFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC. LACK OF CLARITY IN THE SHUTDOWN PROCEDURE WHICH DID NOT ADEQUATELY SPECIFY PERFORMANCE OF THIS EVENT RELATED SURVEILLANCE. THE SURVEILLANCE PROCEDURE WAS COMPLETED SATISFACTORILY AND UNIT ENTERED MODE 4. THE SHUTDOWN PROCEDURE WILL BE REVISED TO CLARIFY NEED TO VERIFY VALVE LINE-UP FOR BORON INJECTION FLOW PATH WHILE IN MODE 5.
THREE MILE ISLAND-2 REACTIVITY CONTROL SYSTEM COMPONENT CODE NOT APPLICABLE SUBCOMPONENT NOT APPLICABLE PERSONNEL ERROR LICENSED & SENIOR OPERATORS ITEM NOT APPLICABLE	05000320 79-010/01T-0 025334	021479 022679 2-WEEK	WHILE IN MODE 1 FOUND BORON CONCENTRATION IN THE BORIC ACID MIX TANK WAS GREATER THAN THAT REQUIRED BY T.S. 3.1.2.9 AND THAT THE LCO ACTION STATEMENT HAD NOT BEEN INVOKED. BECAUSE A REDUNDANT SOURCE OF BORON WAS AVAILABLE AND BECAUSE NO EVENT OCCURRED WHICH REQUIRED BORON INJECTION, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC. THIS EVENT WAS CAUSED BY UNIT PERSONNEL FAILING TO RECOGNIZE THAT THE ACCEPTANCE CRITERIA OF THE SURVEILLANCE PROCEDURE HAD NOT BEEN MET. THE PERSONNEL INVOLVED WILL BE COUNSELLED TO MORE CAREFULLY REVIEW SURVEILLANCE RESULTS VS. ACCEPTANCE CRITERIA.
THREE MILE ISLAND-2 EMERG GENERATOR SYS + CONTROLS ENGINES,INTERNAL COMBUSTION SUBCOMPONENT NOT APPLICABLE OTHER NOT APPLICABLE FAIRBANKS MORSE	05000320 79-011/03L-0 025415	021779 031679 30-DAY	OPERATING DIESEL GENERATOR DF-X-1B WITH UNIT IN MODE 1 ON 02-17-79 AND 02-21-79 D.G. OUTPUT POWER COMMENCED FLUCTUATING UNTIL REVERSE POWER CAUSED TRIP. DF-X-1B WAS RESTARTED IMMEDIATELY FOLLOWING BOTH OCCURRENCES AND SUCCESSFULLY LOADED TO 3MW FOR ONE HOUR CONFIRMING OPERABILITY AND NEGATING THE NEED FOR PERFORMING T.S. 3.8.1.1 ACTION A. BOTH OFFSITE CIRCUITS AND REDUNDANT D.G. WERE OPERABLE AND EVENT DID NOT AFFECT PUBLIC HEALTH AND SAFETY. THOROUGH VISUAL INSPECTIONS FAILED TO REVEAL A DEFINITE CAUSE. WIRING CONNECTIONS TO GOVERNOR ACTUATOR WERE TIGHTENED AND VOLTAGE READINGS TAKEN TO INSURE OPERABILITY OF GOVERNOR. ACTUATOR OIL CHANGED AS AN ADDITIONAL CORRECTIVE MEASURE. WIRING CONNECTIONS PROBABLE CAUSE WILL BE PERIODICALLY INSPECTED.
TURKEY POINT-3 FIRE PROTECTION SYS + CONT PUMPS CENTRIFUGAL PERSONNEL ERROR NONLIC. OPERATIONS PERSONNEL FAIRBANKS MORSE	05000250 79-001/03L-0 025414	021379 031379 30-DAY	DURING AN INSURANCE INSPECTION, THE A FIRE PUMP WAS FOUND TO BE INOPERABLE. SUBSEQUENT INVESTIGATION DETERMINED THAT THE PUMP MAY HAVE BEEN INOPERABLE FOR 8 DAYS DUE TO ITS POWER SUPPLY BREAKER NOT HAVING BEEN CLOSED FOLLOWING A MAINTENANCE CLEARANCE. INOPERABILITY OF A FIRE PUMP IN EXCESS OF 7 DAYS IS REPORTABLE PURSUANT TO T.S. 3.14.2.B.1. THE BREAKER WAS CLOSED AND THE PUMP TESTED. THE PUMP WAS THEN DECLARED OPERABLE. THE CAUSE OF THE FAILURE WAS IMPROPER OPERATION OF THE POWER SUPPLY BREAKER BY THE OPERATOR FOLLOWING A MAINTENANCE CLEARANCE, I.E., THE BREAKER WAS CHARGED BUT NOT CLOSED. THE LONG TERM CORRECTIVE ACTION IS THE ADDITION OF A LOG ENTRY REQUIRING EACH SHIFT TO VERIFY POWER AVAILABLE TO THE FIRE PUMPS.

METROPOLITAN EDISON COMPANY SUBSIDIARY OF

Reference 40

POST OFFICE BOX 542 READING, PENNSYLVANIA 19603 TELEPHONE 215 – 929-3601

July 7, 1978
GQL 1133

Director of Nuclear Reactor Regulation
Attn: S. A. Varga, Chief
Light Water Reactors Branch #4
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

IE FILE CO

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Docket No. 50-320
Operating License No. DPR-73
Technical Specification Change Request No. 014

Enclosed are three signed originals (sixty conformed copies sent separately) of Technical Specification Change Request No.014 requesting amendment to Appendix A of Operating License No. DPR-73.

Also enclosed is one signed copy of Certificate of Service for proposed Technical Specification Change Request No.014 to the chief executives of the township and county in which the facility is located.

Sincerely,

Signed J. G. Herbein

J. G. Herbein
Vice President

Dupe of 7904200118 10

JGH:JRS:cjg

Enclosures:
1) Technical Specification Change Request No. 014
2) Certificate of Service for Technical Specification Change Request No. 014
3) Check No. #36315

COPY SENT REGION I

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT

Operating License No. DPR-73
Docket No. 50-320
Technical Specification Change Request No. 014

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-73 for Three Mile Island Nuclear Station Unit 2. As a part of this request, proposed replacement pages for Appendix A are also included.

METROPOLITAN EDISON COMPANY

By Signed J. G. Herbein
Vice President

Sworn and subscribed to me this 7th day of July, 1978.

Original Signed By
G. J. TROFFER
Notary Public

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-320
LICENSE NO. DPR-73

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No.014 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 2, has, on the date given below, been filed with the U. S. Nuclear Regulator 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. Weldon B. Arehart
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, Pennsylvania 17057

Mr. Harry B. Reese, Jr.
Board of County Commissioners
of Dauphin County
Dauphin County Court House
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By Signed J. G. Herbein
Vice President

Dated: July 7, 1978

Three Mile Island Nuclear Station Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

Technical Specification Change Request

The licensee requests that the attached changed pages replace 2-5, 2-6, 3/4 2-13, 3/4 3-3, and 3/4 3-13, and that the attached changed figures replace figures 2.1-1, 2.1-2, 2.2-1 and 2.2-2. Also included for your information and use are changed pages B 2-1, B 2-2, B 2-6 and figure 2.1, all of which are part of the "Bases" Section of the TMI-2 Technical Specifications. These pages will be distributed to Technical Specification (T.S.) copyholders upon approval of this change request. However, in accordance with 10 CFR 50.36, the bases are not considered part of the T.S., and therefore, changes in them do not require your approval.

Reasons for the Change Request

Due to the recent concern over wear encountered in the Fuel Assembly (FA) holddown latch assemblies in units similar to TMI-2, caused by vibration of Burnable Poison Rod Assemblies (BPRAs) and Orifice Rod Assemblies (ORAs), it is believed necessary to install retainers on the BPRAs and on two modified ORAs and to remove the remaining ORAs. Installation of the BPRA retainers reduces hot assembly flow by less than 1% and removal of all 40 ORAs would increase bypass flow by only 1.6%. These effects on the system flow characteristics are very slight and when they are combined with the increased flow presented in the attached proposed changes, the DNBR safety margin is actually increased. Although the present safety margins are adequate to compensate for the changes in flow distribution brought about as a result of these core alterations, we feel that system flow should be increased to ensure even larger DNBR safety margin.

Several changes included in this change request are not related to ORA removal and BPRA retainer installation. The first of these changes is the increase in RCS Pressure - low trip setpoint in Table 2.2-1 from 1800 psig to 1900 psig. This change is being made for greater operating flexibility and to allow for a backup function in case of a small break LOCA although it will most likely never be used because the RCS pressure - variable low should occur first. It is also being made to increase the margin to HPI so that a rapid depressurization will not unnecessarily inject HPI as frequently as would occur with less margin. As a result of this increase in the RCS Pressure - low trip setpoint, it is correspondingly necessary to increase the manual bypass by 100 psi to ≤1820 psig to incorporate 1820 psig as the new high pressure trip during startup in shutdown bypass. This will enable startup to be performed more easily and will continue to maintain the same margin previously used to allow for instrument errors. Since the change in RCS Pressure - low trip setpoint is in a conservative direction, it is bounded by previous analyses and therefore does not require an additional safety analysis.

The rod bow penalty which is not directly related to ORA removal has been revised to correctly reflect the NRC Rod Bow Model. The original Technical Specification was prepared using the B&W Rod Bow Model and during investigation into the removal of the ORAs was discovered and is corrected herein. The imbalance curves in Figures 2.2-1 and 2.2-2 are also being corrected to account for an initial misinterpretation of the positive offset error equation.

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.6% of RATED THERMAL POWER with four pumps operating #
	$\leq$ 78.1% of RATED THERMAL POWER with three pumps operating	$\leq$ 78.2% of RATED THERMAL POWER with three pumps operating #
	$\leq$ 50.9% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 51.0% of RATED THERMAL POWER with one pump operating in each loop #
3. RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 619.08°F #
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBLANCE (1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-2. #
5. RCS Pressure-Low (1)	$\geq$ 1900 psig	$\geq$ 1899.0 psig*; $\geq$ 1891.5 psig**
6. RCS Pressure-High	$\leq$ 2355 psig	$\leq$ 2356.0 psig*; $\leq$ 2363.5 psig**
7. RCS Pressure-Variable Low (1)	$\geq$ (13.00 T_{out} °F - 5887) psig	$\geq$ (13.00 T_{out} °F - 5887.64) psig#

TABLE 2.2-1 (Co ed)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT		TRIP SETPOINT	ALLOWABLE VALUES
8.	Nuclear Overpower based on Pump Monitors[(1)]	≤ 125% of RATED THERMAL POWER with three pumps operating	≤ 125% of RATED THERMAL POWER with three pumps operating #
		≤ 56.9% of RATED THERMAL POWER with one pump operating in each loop	≤ 57.18% of RATED THERMAL POWER with one pump operating in each loop#
		≤ 0% of RATED THERMAL POWER with two pump operating in one loop and no pump operating in the other loop	≤ 0.28% of RATED THERMAL POWER with two pumps operating in one loop and no pump operating in the other loop#
		≤ 0% of RATED THERMAL POWER with no pumps operating or only one pump operating	≤ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating #
9.	Reactor Containment Vessel	≤ 4 psig	≤ 4 psig #

[(1)] Trip may be manually bypassed when RCS pressure ≤ 1820 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is ≤ 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤ 1820 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure >1900 psig.

*Allowable value for Channel Functional Test
**Allowable value for Channel Calibration
#Allowable value for Channel Functional test and Channel Calibration

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.05% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1900 psig, and RCS Pressure-Variable Low, (13.00 T_{out} °F-5887) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protection against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpointof (13.00 T_{out} °F-5927) psig.

Nuclear Overpower Based on Pump Monitors

In conjuction with the power/imbalance/flow trips the Nuclear Overpower Based On Pump Monitors trip prevents the minimum core DNBR from decreasing belwo 1.30 by triping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure $\leq$ 1820 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER.

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of $\leq$1820 psig is imposed.

(3) The Shutdown Bypass is removed when RCS pressure > 1900 psig.

(b) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1,

TABLE 3.3-3 (Continued)

TABLE NOTATION

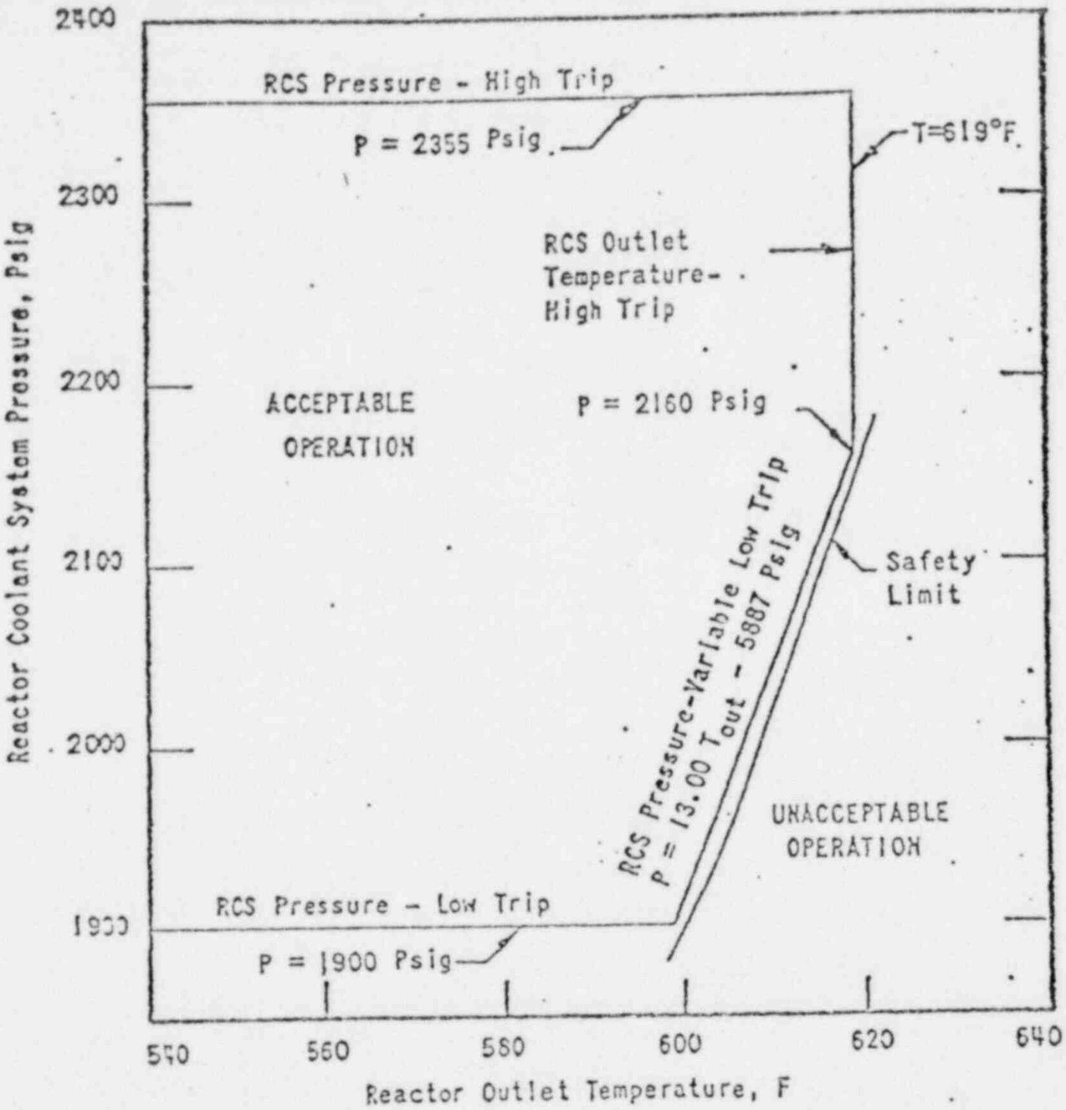
* Trip function may be bypassed in this MODE with RCS pressure below 1920 psig. Bypass shall be automatically removed when RCS pressure exceeds 1950 psig.

** 3 channels per Automatic Actuation Logic, Each R. B. Pressure High Channel trips one Safety Injection Channel and one R. B. Cooling & Isolation Channel.

*** 3 channels per Automatic Actuation Logic, R. B. Spray Valves are actuated by R. B. Cooling and Isolation.

**** Trip function may be bypassed in this mode with steam generator pressure $\leq$ 800 psig. Bypass shall be removed when steam generator pressure $\geq$ 800 psig.

The provisions of Specification 3.0.4 are not applicable.



TMI - UNIT 2
REACTOR CORE SAFETY LIMIT

Figure 2.1-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-73

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The issuance of this license amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the amended Facility Operating License No. DPR-73 is hereby amended by changing the Technical Specifications as indicated in the attachment to this license amendment.

   Paragraph 2.C.(2) of amended Facility Operating License No. DPR-73 is hereby amended to read as follows:

Dupe of
7904300090

"2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 6 are hereby incorporated in the license. Metropolitan Edison Company shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Steven A. Varga
Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

AUG 17 1978