REPORT IN RESPONSE TO NRC STAFF

RECOMMENDED REQUIREMENTS FOR RESTART OF THREE MILE ISLAND NUCLEAR STATION UNIT 1

Met-Ed / GPU

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1.0 INTRODUCTION AND REPORT ORGANIZATION

1.1 INTRODUCTION

Metropolitan Edison Company (Met-Ed) applied for a license to construct and operate Three Mile Island Nuclear Station Unit 1 (TMI-1) on May 1, 1967 (TMI-1 is jointly owned by Met-Ed, Jersey Central Power and Light (JCP&L), and Pennsylvania Electric Company (Penelec) but operated by Met-Ed. Met-Ed, JCP&L and Penelec are wholly owned subsidiaries of General Public Utilities (GPU).) Following issue of the Atomic Energy Commissions (AEC) Safety Evaluation Report (February 5, 1968 as supplemented April 26, 1968) and hearings before the Atomic Safety and Licensing Board (ASLB) the AEC issued a permit to construct TMI-1 (CPPR-40) on May 18, 1968.

On March 2, 1970, Met-Ed filed the Final Safety Analysis Report (FSAR) and Operating License Application for TMI-1. The application was for operation at a core power level of 2535 megawatts thermal (MWt) based on Babcock & Wilcox (B&W) analyses performed for a core power level of 2568 MWt. Based on its SER issued June 11, 1973, the AEC issued Operating License DPR-50 on April 19, 1974.

TMI-1 achieved initial criticality on June 5, 1974 and was declared "Commercial" on September 2, 1975. Since commercial operation TMI-1 has been refuelled five times. The unit was ready to begin operation on the fifth core on March 28, 1979 when the TMI-2 accident occurred. Until conditions at TMI-2 were fully understood Met-Ed decided to keep TMI-1 shutdown. On April 16, 1979, Met-Ed committed to providing the NRC with significant advance notice prior to startup of TMI-1.

On June 28, 1979, Met-Ed informed the NRC that TMI-1 would not be started up until certain plant modifications were completed. The NRC issued an Order on July 2, 1979 that TMI-1 remain shutdown until after a public hearing and further commission order. The Commission issued a further Order and Notice of Hearing on August 9, 1979 which included a list of requirements which the Director of NRR had recommended as a condition for restart of TMI. This report addresses these recommended requirements, except that the requirements for a demonstration of managerial capability and financial resources and of financial qualifications will be separately addressed.

1.2 REPORT ORGANIZATION

This report is composed of eleven (11) sections which, combined, cover the August 9, 1979 Order requirements. All requirements of a related nature are discussed in a single section. For example all requirements related to plant hardware modifications are presented in Section 2 and referenced by other Sections as appropriate. Section 10 provides a discussion of how a requirement is met or where in the report the discussion can be found.

1.3 ABBREVIATIONS

Abbreviations or Acronyms are frequently used throughout this report. The ones more commonly used are defined below:

ACRS	Advisory Committee on Reactor Safeguards
B&W	Babcock & Wilcox
CRDM	Control Rod Drive Mechanism
DH	Decay Heat
ECCS	Emergency Core Cooling System
ES	Engineered Safeguards
FSAR	Final Safety Analysis Report
HPI	High Pressure Injection
ICS	Integrated Control System
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
MU	Makeup
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PORV	Power Operated Relief Valve
PRZR (PZR)	Pressurizer
psig	pounds per square inch gauge
QA	Quality Assurance
RB	Reactor Building
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SFAS (ESFAS)	Safety Features Actuation System
TMI	Three Mile Island

?.0 PLANT MODIFICATIONS

2.1 GENERAL

The plant design features related to safe operation have been described in detail in the TMI-1 FSAR and in various submittals to the NRC since the issuance of TMI-1's Operating License on April 19, 1974. Further modifications are being made to the plant in response to the staff's recommendations contained in the Commission's Order dated August 9, 1979. These modifications are described below and will be completed before startup of TMI-1 or shortly thereafter. Modifications to be completed before startup are described in Section 2.1.1 and modifications which may be completed later are described in 2.1.2. In addition Section 2.1.3 describes certain additional modifications not included in the staff's recommendations. These modifications were proposed by Met-Ed in its June 28, 1979 letter to NRC and are to be completed prior to restart of TMI-1.

2.1.1 Short-Term Modifications

2.1.1.1 Control Grade Reactor Trip on Loss of Feedwater/Turbine Trip

2.1.1.1.1 System Description

The B&W designed Control Grade Reactor Trip System will be installed at TMI-1 in order to implement a reactor trip upon loss of both main feedwater pumps or upon a turbine trip. This system utilizes existing trip signals for loss of feedwater pumps and turbine trip from within the ICS system. These signals are OR'ed and utilized to trip both AC CRD trip breakers. This will result in a trip of the reactor.

2.1.1.1.2 Design Bases

The B&W Control Grade Reactor Trip System is designed to provide a reactor trip upon loss of main feedwater pumps or a turbine trip as an anticipatory trip. This would preclude reactor trips on high pressure for the anticipated transients conditions, thus, reducing any challenges to the PORV/Pressurizer Safety Valves. This system is designed as an interim system until a "safety" grade reactor trip system is available (See Section Later).

2.1.1.1.3 System Design

The B&W Control Grade Reactor Trip System is intended to provide a reactor trip upon loss of feedwater or turbine trip for the period of time that is required to install a "safety" grade reactor trip. The control grade scheme utilizes trip signals originating within the ICS which have proven reliability. These same signals supply the input to the ICS to provide for the automatic runback of the plant for thse transients. The B&W Control Grade Reactor Trip provides a bypass arrangement in order to allow for power escalation, starting the main turbine and normal shutdown of the main turbine.

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2.1.1.1.4 System Operation (See Figures 2.1.1-1, 2 and 3)

Relays 86X-TT and 86-FPF including the field contacts associated with them already reside in the ICS. Note that the field contacts close to convey the respective trip states. The trip of both main feedwater pumps energizes relay 86/FPF. Upon loss of both main feedwater pumps, relays 86-1/RTFT and 86-2/RTFT de-energize due to the opening of contact from 86/FPF. When relays 86-1/RTFT and 86-2/RTFT de-energizes 2 output contacts cause a reactor trip by means of the CRD trip breakers.

A trip of the main turbine cuases closure of the contact in series with the existing relay 86X/TT. Relay 86X/TT energizes, opening a contact in series with relays 86-1RT/FT and 86-2/RFT. De-energizing relays 86-1/RTFT and 86-2/RTFT causes a reactor trip as described above. A bypass feature has been designed to allow for power escalation, normal starting and stopping the main turbine. The main turbine trip bypass will be placed in effect only by automatic action w hen the reactor power is equal to or less than 20%. When reactor power increases above 20%, the turbine bypass feature is automatically removed.

Assuming reactor power is above 20%, then if power is decreased to 20% or below, relay 97/NPL-2 closes a contact energizing relay 86/TTBP. Relay 86/TTBP closes a contact and implements the turbine trip bypass. Relay 86/TTBP annunciates in the control room that the turbine trip is bypassed. When reactor power is increased above 20%, the turbine tripbypass is removed.

In order to allow for normal start-up of the main feedwater pumps, a bypass scheme is provided. The bypass is placed in effect only by automatic action when reactor power is equal to or less than 10%. When reactor power increases above 10%, the bypass is automatically removed. When reactor power is decreased to 10% or less, relay 96/NLP-2 closes a contact energizing relay 86/FTBP. Relay 86/FTBP closes a contact and implements the main feedwater pump bypass. Relay 86/FTBP annunciates in the control room that the main feedwater pump trip is bypassed. When reactor power increases above 10%, the main feedwater bypass is automatically removed.

2.1.1.1.5 Design Evaluation

The B&W Control Grade Reactor Trip scheme provides an anticipatory trip to the reactor, reducing the number of reactor trips on high pressure. This scheme utilizes ICS trip signals which have been proven during past operating experience.

2.1.1.1.6 System Safety Evaluation

This system interfaces with the existing ICS and CRD trip breakers; however, its operation does not interfere with any provide for a fail-safe arrangement, i.e., the reactor will trip. This scheme does not reduce any margin of safety and, in fact, enhances nuclear safety.

2.1.1.1.7 Start-up Testing

This scheme will be tested during installation to verify its operation prior to start-up.

2.1.1.2 Position Indication for PORV and Safety Valves

2.1.1.2.1 System Description

The purpose of this modification is to provide the Control Room Operator with information on the status of the pressurizer electromatic relief valve RC-RV2 and the pressurizer code safety valves RC-RV1A and RC-RV1B. Discharge flow will be measured by differential pressure transmitters connected across elbow taps downstream of each of the valves. In addition, the electromatic relief valve will be monitored by accelerometers mounted on the valve. These will detect flow if the valve opens. Alarms and indications will be provided in the control room to inform the operator if any of these valves are open.

2.1.1.2.2 Design Bases

A reliable and unambiguous indication will be provided to the Control Room Operator if the pressurizer electromatic relief valve or code safety valves open. The monitoring system will remain functional in containment conditions associated with any transient for which valve status is required by the operator. Redundant and diverse means will be provided for monitoring the electromatic relief valve (RC-RV2). The monitoring systems will remain functional during a loss of off-site power. All equipment inside containment will be seismically mounted. The integrity of existing safety related systems will not be impaired by this modification.

2.1.1.2.3 System Design

All of the system components have been selected for reliable operation and, where applicable, for operation under adverse conditions inside containment. The differential pressure transmitter model has been qualified for nuclear applications inside containment. The accelerometers and their associated electronics have been previously qualified for nuclear applications in the Loose Parts Monitoring System. The monitoring systems will be supplied from on-site electrical power supplies. Diverse and redundant means will be used for monitoring of the electromatic relief valve. Both differential flow measurement and acoustic detectors will be provided.

2.1.1.2.4 Design Evaluation

Elbow taps are widely used for flow measurement in fluid systems and a great deal of empirical data is available for calculating expected differential pressure across elbow taps for given flow conditions. Calculations have been made, using conservative assumptions, to demonstrate that a satisfactory signal will be generated when any of the valves open. Calculations have been made for saturated, liquid and two phase flow. Tests run by B&W at their Alliance facility have confirmed the feasibility of this approach.

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Acoustic monitoring of the electromatic relief valve makes use of well proven equipment and techniques which have been used in the B&W Loose Parts Monitoring System. Tests run on this valve at the B&W Alliance facility demonstrated that the acoustic monitoring system gave satisfactory results.

2.1.1.2.5 Safety Evaluation

Instrument taps will be installed on elbows in the discharge piping of pressurizer code safety valves RC-RV1A and RC-RV1B and electromatic relief valve RC-RV2. This piping is classified as N2, Seismic I. Analysis has been performed to demonstrate that this modification will not degrade the integrity of the existing pipe. The pipe classification has been maintained up to and including the instrument root valves. The mounting of new equipment which will be located in the vicinity of safety related systems has been analyzed to ensure that no hazardous missiles will be generated in a seismic event. It has been concluded that this modification will not degrade any safety related systems.

2.1.1.2.6 Instrumentation

The output signals from the three differential pressure transmitters will be displayed on indicators in the control room. They will be calibrated in "inches of water". Each signal will also go to an alarm bistable. A control room alarm will be initiated if any of the signals exceed a pre-determined value. This will alert the operator that one of the valves is open. The differential pressure signal will also be monitored by the plant computer for logging, trending, and alarm functions.

The outputs from the accelerometers which will be mounted on RC-RV2 will be processed by monitoring equipment installed in the existing Loose Parts Monitoring Cabinet. An output signal indicative of flow through the valve will be displayed and recorded locally. A control room alarm will be initiated if flow is detected. This signal will also be monitored by the plant computer for logging, trending, and alarm purposes.

2.1.1.3 Emergency Power Supply Requirements for Pressurizer Heaters, PORV, Block Valve, and Pressurizer Level Indication

2.1.1.3.1 Pressurizer Heaters

2.1.1.3.1.1 System Description

The purpose of this modification is to provide redundant emergency power for the 126 KW of pressurizer heaters required to maintain natural circulation conditions in the event of a loss of offsite power. A manual transfer scheme will be installed to transfer the source of power for 126 KW of pressurizer heaters from the balance of plant (BOP) source to a "Red" engineered safeguards (ES) source. A similar manual transfer scheme will be installed to transfer the source of power for 126 KW of pressurizer heaters from the BOP source to a "Green" ES source. Each manual transfer scheme will have double isolation on each end of the transfer and have mechanical key interlocks to govern the order of the transfer procedure. Figure 2.1-4 is a schematic representation of these transfer schemes.

2.1.1.3.1.2 Design Basis

Babcock and Wilcox has recommended that at least 126 KW of pressurizer heaters be restored from an assured power source within two hours after a loss of offsite power. Separation and isolation of Class IE equipment and circuits from non-Class IE equipment and circuits will be in accordance with Regulatory Guide 1.75.

2.1.1.3.1.3 System Design

Existing spare Class IE 480 volt circuit breakers on the "Red" and "Green" ES systems will be utilized for the two transfer schemes. Four new three-phase Class IE qualified disconnect switches will be utilized. A disconnect switch will be installed in series with each of the above mentioned Class IE 480 volt circuit breakers. A disconnect switch will also be placed in series with each of two non-Class IE pressurizer heater group circuit breakers. A pressurizer heater group corresponds to 126 KW and there are a total of thirteen such groups. Class IE qualified power cable will connect the load sides of the disconnect switches as shown in Figure 2.1-4. Class IE qualified under-voltage relays will be installed on each ES bus. They will initiate tripping of the ES circuit breaker to the pressurizer heaters when the bus voltage drops below its set point. The set point will be chosen so that starters on the ES bus can pickup if energized and the voltage at the ES motors is not lower than their ratings allow. The remainder of the electrical power distribution system to the pressurizer heaters will remain as it presently exists.

2.1.1.3.1.4 System Operation

All pressurizer heaters will be powered from the BOP electrical power distribution system when offsite power is available. Upon a loss of offsite power, manual transfers will enable each of the onsite emergency diesel generators ("Red" and "Green") to provide power to 126 KW of pressurizer heaters when the diesel generators can accommodate that load. Procedures will call for tripping non-essential loads to accomplish this within the two-hour requirement. Mechanical key interlocks will dictate that the order of events in the transfer from BOP to ES power source will be as follows:

A. Open the pressurizer heater group circuit breaker.
B. Open the disconnect switch associated with that pressurizer heater group circuit breaker.
C. Close the disconnect switch associated with the ES circuit breaker.
D. Close the ES circuit breaker.

When offsite power is restored, the reverse procedure will be used to transfer back to the BOP source.

2.1.1.3.1.5 Safety Evaluation

The manual transfer scheme design provides double Class IE separation of the ES system from the BOP system - the ES circuit breaker and the disconnect switch. Taking into account the single failure criteria, faults on the BOP system will, at most, cause the loss of one 480 volt ES system. The transfer scheme design also precludes the connection of the "Green" ES system to the "Red" ES system.

2.1.1.3.1.6 Inservice Testing Requirements

The emergency diesel generator loading procedure will be rewritten to incorporate this modification. Therefore, these transfer schemes will be tested when the emergency diesel generators are tested.

2.1.1.3.2 Power Operated Relief Valve (PORV)

The present plant design is such that emergency diesel generator power will be supplied to the PORV (RC-V2) upon loss of offsite power. The PORV is powered from the 250 VDC Distribution Panel IC which in turn is powered from the "Red" and "Yellow" ES batteries and ES Battery Chargers 1A, 1C, and 1E.

2.1.1.3.3 Block Valve

The present plant design is such that emergency diesel generator power will be supplied to the block valve (RC-V3) upon loss of offsite power. The block valve is powered from the 480 V Engineered Safeguard Valve Control Center 1C.

2.1.1.3.4 Pressurizer Level Instrumentation

The present plant design is such that emergency diesel generator power will be supplied to the pressurizer level instrumentation power supplies (RC-1-LT1, RC-1-LT2, RC-1-LT3) upon loss of offsite power. The pressurizer level instrumentation power supplies are part of the ICS, NNI System, and are powered from the 120 volt ICS, NNI Power Distribution Panel ATA. That panel is, in turn, powered from the 120 volt Vital Distribution Panel VBA.

2.1.1.4 POST LOCA HYDROGEN RECOMBINER SYSTEM

2.1.1.4.1 System Description

The purpose of this modification is to provide a system which shall serve as a means of controlling combustible gas concentrations in containment following a loss of coolant accident (LOCA). After a LOCA, the containment atmosphere of a PWR is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen and water droplets containing boron, sodium-hydroxide and/or sodium thiosulfate. During and following a LOCA, the hydrogen concentration in the containment results from radiolytic decomposition of water, zirconium-water reaction and aluminum reacting with the spray solution.

If excessive hydrogen is generated it may combine with oxygen in the containment atmosphere. The capability to mix the combustible atmosphere and prevent high concentrations of combustible gases in local areas is provided by the reactor building ventilation system. The hydrogen combiner system must be capable of reducing the combustible gas concentrations within the containment to below 4.1 volume percent.

The recombiner shall be capable of removing containment air mixed with hydrogen, recombine the hydrogen and exhaust the processed air back into the containment. This system is not required during normal plant operation.

2.1.1.4.2 Design Basis

The recombiner system shall meet the design and quality assurance requirements for an engineered safety feature in terms of redundancy for active components, electrical power and instrumentation. The design bass for the system shall be a loss-of-coolant accident (LOCA) with hydrogen generation rates calculated in accordance with NRC Regulatory Guide No. 1.7.

The hydrogen recombiner to be utilized for the system shall be the Rockwell International, Atomics International Div. recombiner unit purchased for TMI Unit No. 2.

One hydrogen recombiner will be installed prior to restart. The second (redundant) recombiner need not be installed, however, the piping system, electrical power supplies and structural provisions shall be installed and available. The second hydrogen recombiners shall be installed after an accident within the time period available before they need to be operational.

2.1.1.4.2 The system will be designed to meet the criteria of NRC Regulatory Guide 1.7, the acceptance criteria of SRP 6.2.5, NUREG 0578 (July 1979), 10CFR50 Appendix A-General Design Criteria for containment design and integrity and 10CFR100 Reactor Site Criteria for limits of offsite releases.

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2.1.1.4.3 System Design

The system design provides an installed and a location with installed piping for a future redundant hydrogen recombiner. The recombiners will be located in the Intermediate Building at floor elevation 305 ft., in the Leak Rate Test equipment area, as shown in Fig. 2.1-6. This system will utilize the existing "Containment Vessel Leak Rate test" penetrations (nos. 415 and 416) as shown diagramatically in Fig. 2.1-7.

Since only active component failure needs to be considered, common containment penetrations will be utilized for the redundant recombiners. All active components will be redundant and will be provided with independent power supplies.

All system components forming the containment boundary will meet the containment isolation criteria and will be designed to Safety Class 2 per ANSI B-31.7. All system supports will be designed for the DBE as seismic class S-I. The recombiners will be powered from Class 1E power sources. The inside containment isolation valves will be solenoid, dc power, operated valves.

The recombiner cooling air will be discharged directly to the outside environment. An evaluation will be performed to demonstrate that potential releases of intermediate building air used for recombiner cooling will not result in off site releases in excess of 10CFR100.

2.1.1.4.4 System Operation

The system is designed to maintain the hydrogen concentration inside containment below the 4.1 percent by volume, lower flammability limit of hydrogen.

Based on the hydrogen generation rate calculated in accordance with NRC Reg. Guide 1.7, the hydrogen recombiner should start processing the containment gases when the hydrogen concentration reaches 3 percent by volume of the total containment.

The recombiner is placed into operation by opening the containment isolation valves after having sampled the containment atmosphere and then turning on the recombiner from its remote-local panel. Local monitoring of the control panel is required until the reaction chamber reaches the required temperature for a self sustaining reaction between hydrogen and oxygen. Once the system is in a recombination mode, only periodic inspection at the control panel is required. A single remote recombiner alarm is provided in the main control room to advise the operator of an operating problem with the recombiner.

When the hydrogen concentration has dropped to an acceptable level, the system is shutdown and the containment isolation valves are closed.

2.1.1.4.5 Safety Evaluation

The hydrogen recombiner system is designed as a nuclear safety class 2, sesimic class S-I system with class 1E power supply.

Containment integrity is normally maintained by double valve isolation (with a valve inside and another outside containment). While the recombiner is being utilized for post-LOCA hydrogen control, containment integrity at the penetration is maintained by a single, manually operated, locked closed valve located outside of containment and the redundant isolation is provided by a blind flange also located outside containment.

In order to insure the ability to draw and return containment atmosphere, considering single active failure of the power operated inside containment isolation valve, two such valves are provide per penetration with each of a redundant pair of valves powered fromalternate dc pwoer supplies. These isolation valves are designed to fail closed on loss of power in order to maintain containment integrity.

All other active components have redundancy by virtue of the redundant recombiner skid and control panel. Each panel may be powered by either the "Red" or "Green" Engineered Safeguards System power supply.

Off site releases due to leakage and discharge to the atmosphere with the recombiner cooling air will be evaluated to demonstrate these releases to be below the 10CFR100 limits.

2.1.1.4.6 Inservice Testing Requirements

No inservice testing is required for the Hydrogen Recombiner System. However, normal inspection, testing and maintenance will be performed in accordance with standard plant operating procedures.

2.1.1.5 CONTAINMENT ISOLATION MODIFICATIONS

2.1.1.5.1 System Description

The functional requirements of the additional containment isolation signals are the following:

1. Provide diverse containment isolation signal from the applicable reactor trip, high radiation, 1500 psig SFAS, or pipe break signal. These signals will assure that radioactie material is not transferred out of the reactor building before a 4 psig isolation signal is reached.

2. All lines open to the containment atmosphere or connected directly to the RCS (either normally or intermittently which can result in transfer of radioactivity outside containment), which are neither part of the Emergency Core Cooling Systems nor support for RCP operation, should be isolated on reactor trip.

3. In order to maintain non-ECCS support services fo RCP operation, the following service lines should be classified as Seismic Category I and closed on the following signals, provided that the piping is protected from pipe whip and/or jet impingement (see Fig. 2.3-5), Deletion of 4 psig RB Isolation Signal Logic):

    a. Reactor coolant pump seal return should be isolated on 30 psig reactor building pressure signal or by the operator through remote manual operation on high radiation alarm.

    b. Nuclear Services Closed Cooling (NSCC) water and Intermediate Closed Cooling (ICC) water should be isolated by the following:

        i. With the exception of the ICC supply to the CRDM coolers, the 30 psig reactor building pressure signal shall isolate these services into containment.

        ii. The ICC supply to the CRDM coolers shall be isolated by the 4 psig reactor building pressure signal.

        iii. Pipe line break isolation shall be provided by either a line break detection and isolation signal or by upgrading the pipe lines to seismic category I and demonstrating that the piping is protected from or can withstand jet impingement and that the only other pipe that can break the NSCC and ICC pipes by pipe whit is the Reactor Coolant piping.

    c. Normal fan cooler coils should be isolated on 30 psig reactor building pressure signal and the piping upgraded to seismic category I.

In order to utilize specific systems which have been automatically isolated, an isolation signal override capability is required. The isolation signal override shall be either on a total basis or on an individual penetration basis dependent on the isolation signal source and the penetration which is to be opened. See Table 2.3-1 for a listing of penetrations and the required isolation override requirements.

The radiation monitoring shall be accomplished at the locations indicated on Table 2.3-2.

High Radiation alarms shall be provided in the control room for each radiation monitor that provides a high radiation closure signal nd for the RC pump seal return line. Each alarm window shall also identify the valves which it is closing or is to be closed by the operator.

4. Specific requirements for each containment isolation valve are tabulated in attached Table 2.3-3. This table identifies the isolation signal for each valve and pipe upgrading requirements for each piping system.

5. Before the existing 4 psig reactor building pressure isolation signal may be deleted from the plant design, the piping system must be evaluated, utilizing the logic shown in attached Figure 2.3-5, to demonstrate that containment integrity will be maintained.

6. Containment isolation signal override capability will be provided in accordance with attached Table 2.3-1 which lists the following types of overrides:

   a. Individual Isolation Signal Bypass - This override shall be capable of bypassing only the specific isolation signal to the appropriate valves associated with only the penetration which it is desired to open. This type of override is noted by an "I" on Table 2.3-1. The initiating isolation condition may still exist when utilizing this override.

   b. Common Isolation Signal Bypass - This override shall be a common override capable of bypassing only the specific isolation signal to all of the appropriate valves associated with the various penetrations which may be desired to open by the operator. The common isolation signal bypass shall also provide the override for the individual isolation signal bypass. This type of override is noted by a "c" on Table 2.3-1. The initiating isolation condition may still exist when utilizing this override.

   c. Automatic Isolation Signal Bypass - The isolation signal for this type of override shall automatically be cleared although the initiating isolation condition may still exist. This will allow the operator to simply push

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the valve switches to "open" position in order to re-open the valves. This feature is used only for the RC system letdown isolation valves after they have been closed by a reactor trip only. This type of override is noted by an "A" on Table 2.3-1.

d. No Bypass Capability - This override shall not permit the operator to re-open the valve unless the initiating condition is removed. If the isolation valves have been re-opened and the initiating condition re-occurs then the valves shall again be isolated.

The containment isolation overrides shall be on an individual signal source basis such that overriding the isolation signal due to one source will still allow the valves to be isolated by a second isolation source if it is activated.

2.1.1.5.2 Design Bases

1. The diverse containment isolation system shall meet the single failure criterion of IEEE No. 279.

2. Redundancy of sensors, measuring channels, logic, and actuation devices shall be maintained and not be degraded by the modifications.

3. Electrical independence and physical separation shall be in accordance with IEEE-383, where practicable. If not possible, existing physical separation criteria will be maintained.

4. Switches, independent of the automatic instrumentation, shall be provided for manual control of all containment isolation valves modified.

5. Manual testing facilities shall be provided for on-line testing to prove operability and to demonstrate reliability. Plant operation should not be adversely affected.

6. All new instrumentation shall meet the environmental and seismic requirements of IEEE-323.

7. The status of all containment isolation valves shall be provided in the control room and not be affected by the modifications.

8. Non-safety related radiation isolation signal will meet all of the above criteria with the following exceptions:

    a. The system will not be seismically qualified.

    b. Testability requirements of IEEE-279 will be met to the extent practicable.

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2.1.1.5.3 Design Evaluations and Systems Operation

In order to cover a broader spectrum of events for which containment isolation is desirable, the reactor trip signal is used as a diverse containment isolation signal. Since a reactor trip signal occurs on low pressure (1800 psig) it is anticipatory of SFAS and occurs prior to SFAS initiation. Therefore the NRC directive would be fulfilled in a conservative way by the reactor trip signal rather than the SFAS signal.

The use of the RPS system would provide isolation for the following events:

a. Rod withdrawal accidents
b. Loss of coolant flow
c. Feedwater line break or loss of feedwater
d. Small steam line break accident outside containment (isolation of containment lines is still desirable)
e. Ejected rod accident
f. Boron dilution accident
g. Cold water addition
h. Iodine spikes or crud burst after trip
i. Loss of offsite power or station blackout

The 1500 psig SFAS signal would not isolate containment for items a, b, c, f, g, h and i. Isolation on 1500 psig SFAS for items d and e would not cover a full spectrum of events.

As discussed above, lines which will be isolated on reactor trip are:

a. reactor building sump
b. RCDT gas vents and liquid discharge
c. RCS sample lines
d. containment purge lines
e. RCS letdown
f. demineralized water
g. OTSG sample lines (due to primary to secondary leaks)

Closure of these paths by a signal that is not dependent on building pressure assures that there will be no uncontrolled radioactivity release from containment for design basis events.

With the exception of the letdown and the demineralized water valves, the above lines are normally isolated. If these lines receive an isolation signal after a reactor trip the plant condition is not degraded. The letdown lines is normally open, and it is immediately closed by operator action after reactor trip.

Special design provisions will be taken with letdown line isolation. If neither 4 psig building pressure nor high radiation exists, the operator will be able to reopen the valve on demand. If eithe rof these signals does exist, however, the operator can only reopen the letdown valve by overriding the closure signal to the valve.

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The demineralized water line is normally open to provide purging of the reactor coolant pump number 3 seal. The purging prevents boron building in the seal. Loss of this, function is not a concern. Westinghouse, the pump manufacturer, has stated that loss of seal purging has been determined not to affect the seal; in fact, at the owners discretion, some pumps are being operated without the purge water connected.

Individual high radiation signals will be used to prevent releases outside containment for the:

1. Reactor building sump drain
2. Reactor coolant system letdown line
3. Reactor coolant drain
4. Reactor building purge (monitor already exists)
5. Reactor coolant sample lines
6. OTSG sample lines
7. Reactor coolant pump seal return (alarm only)
8. Intermediate closed cooling water

Intermediate closed cooling water will be isolated on high radiation in order to prevent inadvertent releases due to letdown cooler leakage into the ICCW system. Isolation of the ICCW system will not jeopardize operation of the reactor coolant pumps. The pumps can run for approximately one week with only seal water providing the cooling for the pump seals. Plant operating procedures will be revised in order to address reinitiation of ICCW cooling of the seals for periods longer than one week.

Individual raidation isolation have been chosen in lieu of a general radiation isolation signal for the following reasons. First, reactor trip isolation will be anticipatory of a high radiation condition. Second, individual isolation is more sensitive to isolating the source of activity. For example, a general radiation signal based on dome activity would not detect a source of activity being added to the RCDT.

Once containment isolation is completed, certain lines may have to be reopened in order to support post trip or post accident operation. Table 3 of Appendix A provides a list of override capability for each of the lines receiving either: reactor trip, high radiation, or 4 psig or 30 psig building pressure isolation signals.

Plant procedures will govern the conditions under which any of these overrides are utilized. In genera, the prerequisite for override is a determination that neither an accident condition nor a radiation hazard exists. If either of these conditions exist, then specifics as to if or when the isolation can be bypassed will be developed on a case by case basis.

Individual reactor trip override capability has not been supplied for all lines except RCS letdown. When a stable post trip condition is achieved, the operator can bypass the containment isolation signal at the system level in order to reestablish control of these systems.

2.1.1.5.4 References

1. Letter from Boyce Grier, of US NRC, to all owners of B&W reactors dated April 5, 1979, IE Bulletins 79-05A, 79-05B, 79-05C.

2. 10CFR50, Appendix A, General Design Criteria 55, 56, and 57.

3. B&W Company, Nuclear Power Generation Division, dated 5/22/79, "Recommendations for Short-Term Changes to Containment Isolation Systems as a result of the Three Mile Island Unit 2 Accident."

4. B&W Company, Nuclear Power Generation Division, dated 5/22/79, "Recommendations for Long-Term Changes to be Considered to Containment Isolation Systems."

5. U.S. Nuclear Regulatory Commission. Standard Review Plan Section 6.2.4, Containment Isolation System, U.S. Nuclear Regulatory Commission.

6. U.S. Nuclear Regulatory Commission. TMI Lessons Learned Task Force Status Report and Short Term Recommendations. NUREG-0578, July 1979.

2.1.1.5.5 Safety Evaluation

The selective addition of the containment isolation signals on high radiation, reactor trip and 30 psig building pressure does not compromise plant safety for the following reasons:

1. The system is designed as safety grade and single failure proof (except for high radiation isolation). Thus, the system will perform its safety function when required. Thus the probability of containment isolation occurring when needed is increased.

2. Spurious initiation of an isolation signal will not introduce transients into the plant that are of significance. Thus, no new accidents/transients are introduced into the plant design.

Finally, the design meets the intent of all NRC directives to Met-Ed regarding containment isolation namely the addition of isolation on high radiation, and low RCS pressure. The design meets the requirements of Standard Review Plan 6.2.4 to the extent practicable.

2.1.1.6 Instrumentation to Detect Inadequate Core Cooling

2.1.1.6.1 System Description

The purpose of this modification is to provide instrumentation for detection of inadequate core cooling as required by paragraph 2.1.3.b of NUREG 0578. It consists of the following parts:

A. Connecting in-core thermocouples to plant computer.
B. Providing a wide range reactor outlet (T_H) temperature measurement.
C. Providing control room indication of reactor coolant saturation pressure margin.

2.1.1.6.2 Design Bases

This modification is to provide the control room operator with information to assist in identifying inadequate core cooling conditions. High quality, control grade instrumentation shall be provided. To the extent practicable, sufficient redundancy shall be provided to allow surveillance of instruments by comparing different channels and to furnish the operator with alternate information if one channel is disabled. Instrumentation shall be available after a loss of offsite power. This modification shall not degrade the integrity of any safety-related system or any existing instruments which are required for safe and reliable operation of the plant.

2.1.1.6.3 System Design

2.1.1.6.3.1 In-Core Thermocouples

The in-core thermocouples are presently cabled from the reactor up to electrical containment penetrations but have not been terminated at the penetrations. The existing chromel-alumel conductors will be spliced to copper wires and run to adjacent penetrations which have spare conductors. Connections will be made to the computer with copper wires. Temperature detectors will be used to monitor the copper to chromel-alumel junctions so that compensation can be made in the computer. This method was necessary since thermocouple extension wire penetrations were not available. The method of connection is shown in Figure 2.1-8. All splices will be made inside penetration terminal boxes and will be protected by means of heat shrink tubing. All 52 of the in-core thermocouples will be brought to the computer. This will provide some redundancy since loss of a few thermocouples will not impair the operator's ability to assess core conditions.

2.1.1.6.3.2 Wide Range T_H

The present control grade Reactor Outlet Temperature channels (T_H) have a range of 520-620°F. They are used for plant control.

This modification will provide a wide range 120-920° T_H output from the same RTD bridge without changing the range or accuracy of the existing signal to the control system. This will be done by installing a new, specially modified converter module across the output of the RTD bridge in parallel with the existing output module. The new signals will be connected to the computer and will also be used as inputs to the saturation pressure instruments described in 2.1.1.6.3.3 below. This modification will be made to four T_H channels, two in each Reactor Coolant loop. A block diagram of the new arrangement is shown in Figure 2.1-9.

2.1.1.6.3.3 P_{SAT} Margin Indication

In order to aid the operator in detecting inadequate core cooling, an instrument will be provided which will display in the control room the margin between the actual primary plant pressure and the saturation pressure (P_{SAT}) for the existing reactor coolant temperature (T_H). P_{SAT} will be computed using the wide range T_H signals described in 2.1.1.6.3.2 above. The computed P_{SAT} will be compared to the actual plant pressure and the P_{SAT} margin will be displayed in the control room. An alarm will be initiated if the margin falls below an acceptable value. Redundancy will be provided by computing P_{SAT} independently for each reactor coolant loop. The more conservative of the parameters for each loop (higher temperature, lower pressure) will be automatically selected for the computations. In addition, the plant computer, using the same inputs, will independently compute P_{SAT} and P_{SAT} margin for logging, trending, and alarm. A block diagram of the system is shown in Figure 2.1-10.

2.1.1.6.4 Design Evaluation

2.1.1.6.4.1 In-Core Thermocouples

The copper to chromel-alumel junctions which have been created by this modification will cause offsets in the thermocouple measurements. However, providing temperature measurements at the junctions will enable the computer to compensate for these offsets, preserving the accuracy of the in-core temperature readings. The splices will be protected against potential degradation by covering them with a heat shrinkable tubing which has been qualified for use inside containment.

2.1.1.6.4.2 Wide Range T_H

Tests have been run to demonstrate that the modified converter module will give an accurate output over the desired range of 120-920°F. The tests also showed that addition of the new equipment will not degrade the existing narrow range 520-620°F control signal. This addition will be implemented with Bailey Controls Company type 820 hardware which has a history of reliable operation in nuclear plants.

2.1.1.6.4.3 P_{SAT} Margin Indication

The modification will provide continuous indication of P_{SAT} margin to the operator. The addition will be implemented with reliable, conservatively applied solide state equipment.

2.1.1.6.5 Safety Evaluation

None of the modifications described involves any safety related instrumentation or control channels. It has been concluded that these modifications will not degrade any safety related systems.

2.1.1.7 Auxiliary Feedwater Modifications

To be provided later.

2.1.2 <u>Long-Term Modifications</u>

To be provided later.

2.1.3 Met-Ed Initiated Modifications

To be provided later.

Reactor Trip on Loss of FWP or Turbine Trip

(New Equipment)

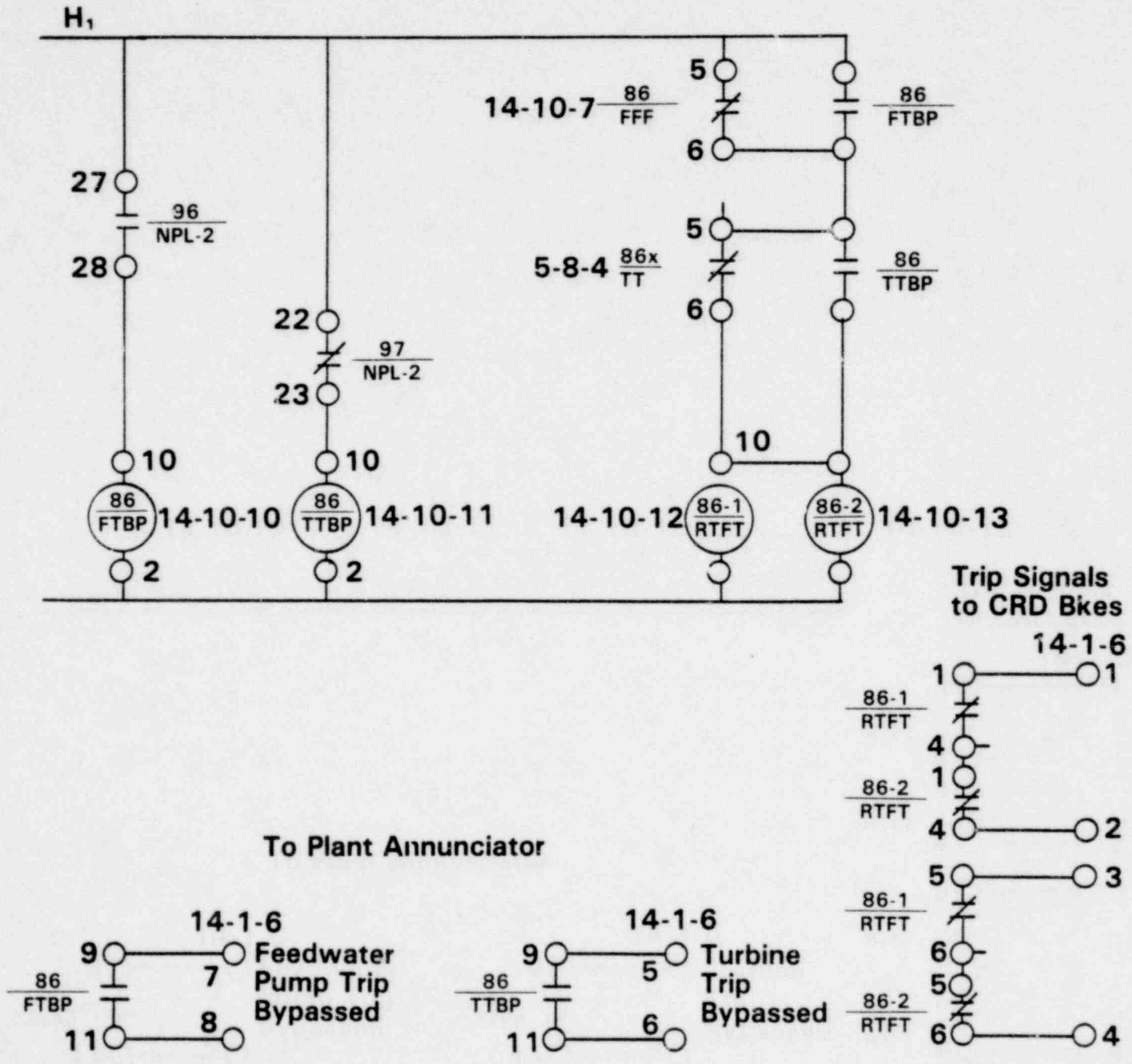

Figure 2.1-1

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Reactor Trip on Loss of FWP or Turbine Trip

(Existing Equipment)

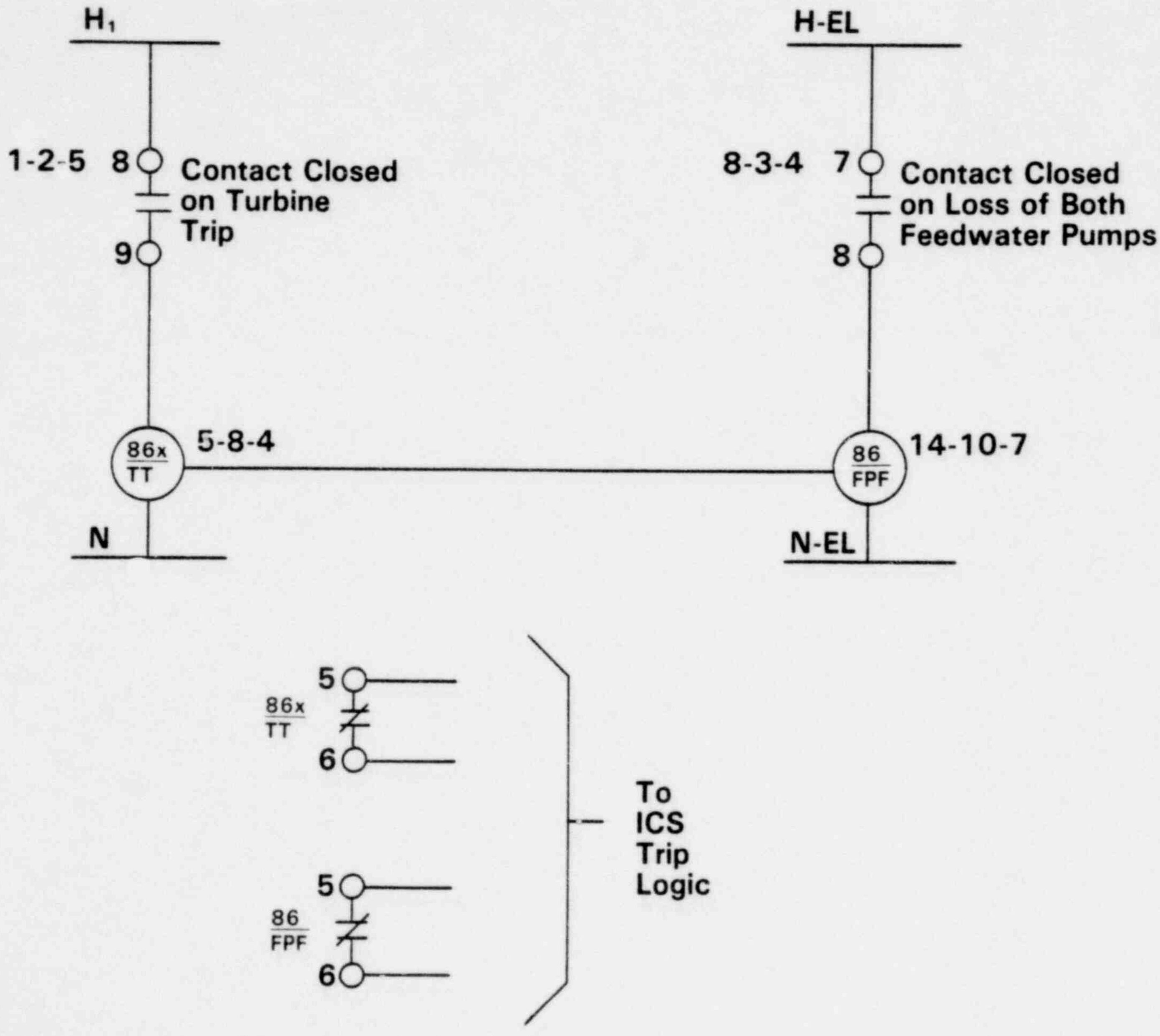

Figure 2.1-2

Reactor Trip on Loss of FWP or Turbine Trip

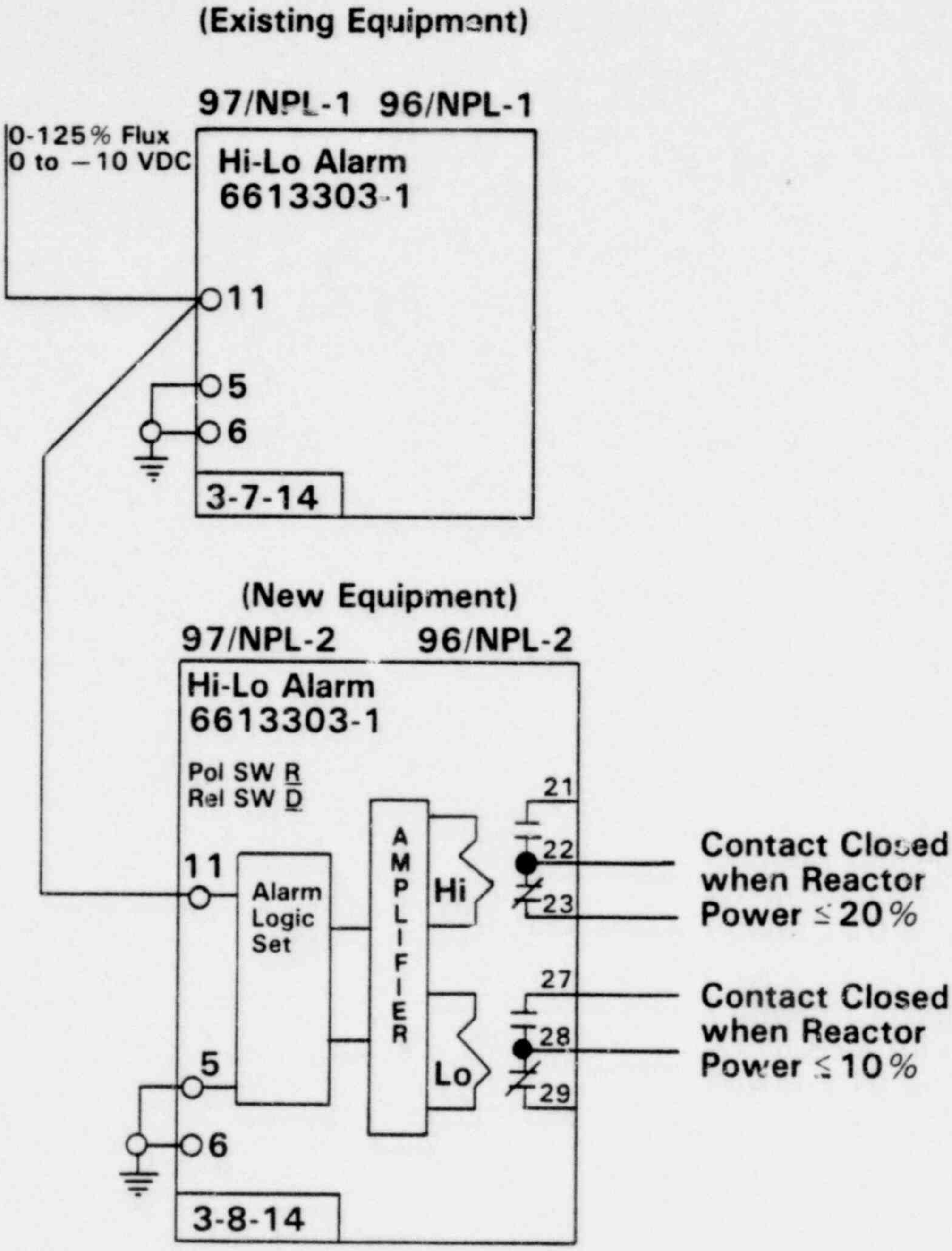

Figure 2.1-3

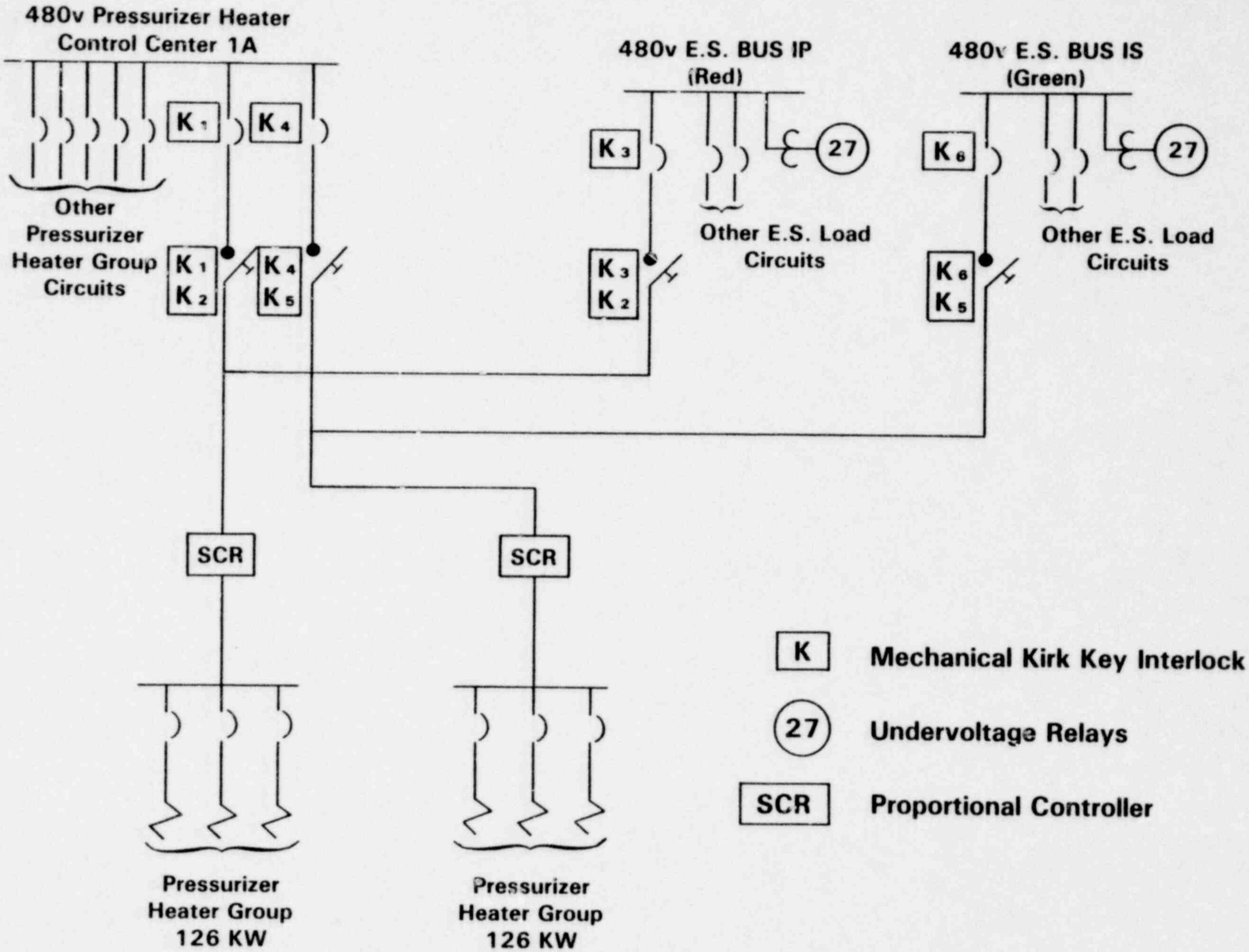

Figure 2.1-4

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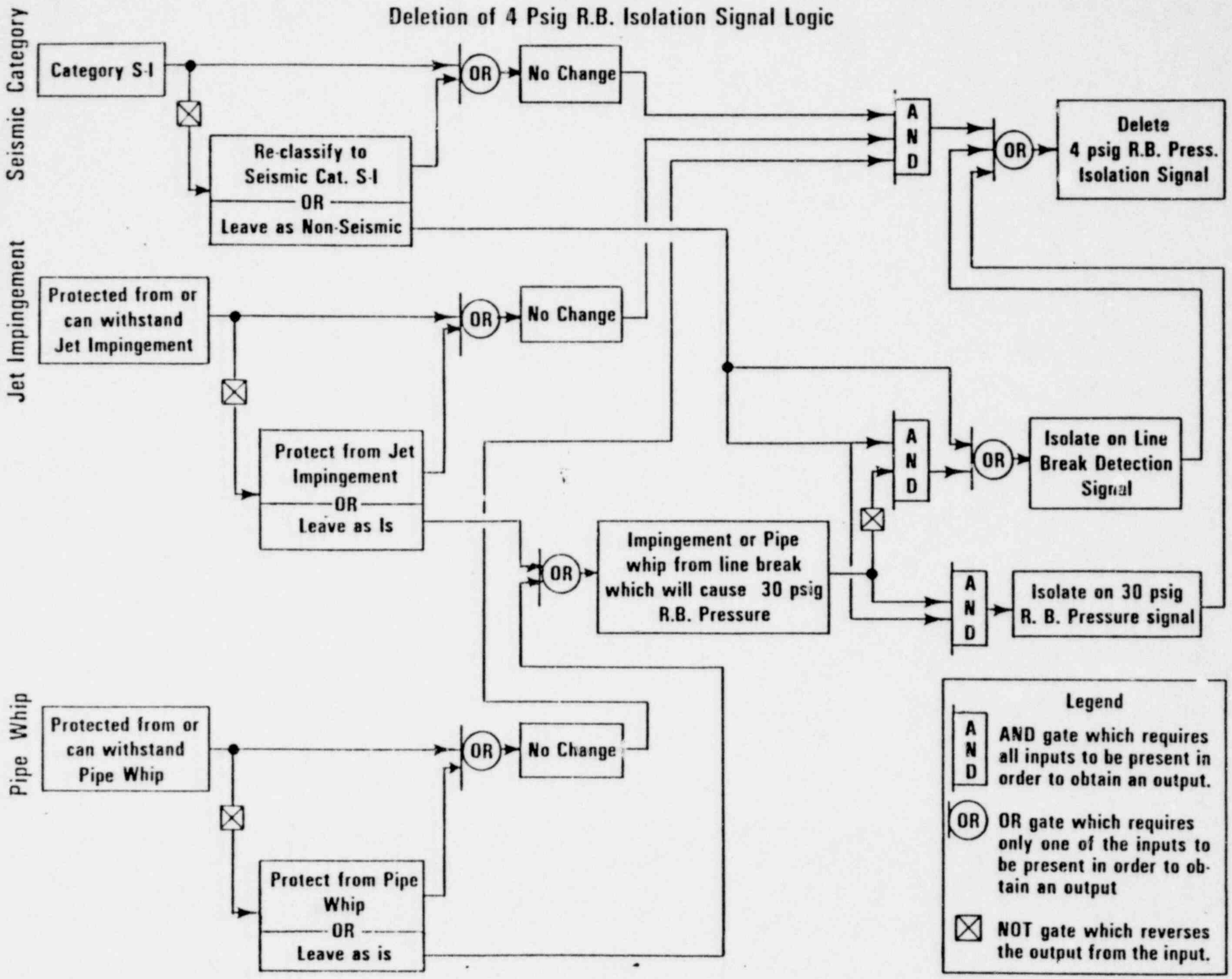
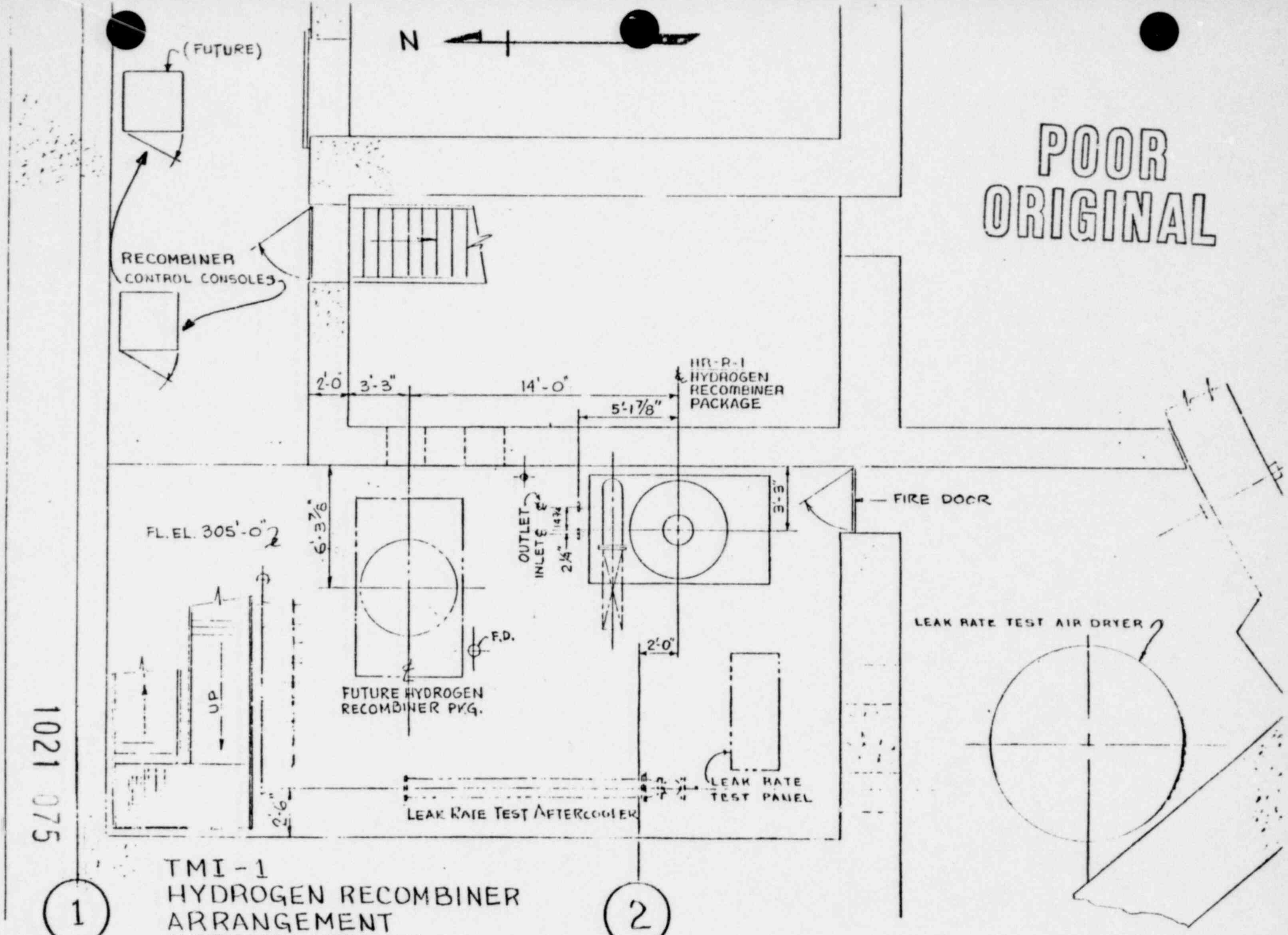
Figure 2.1-5

Deletion of 4 Psig R.B. Isolation Signal Logic

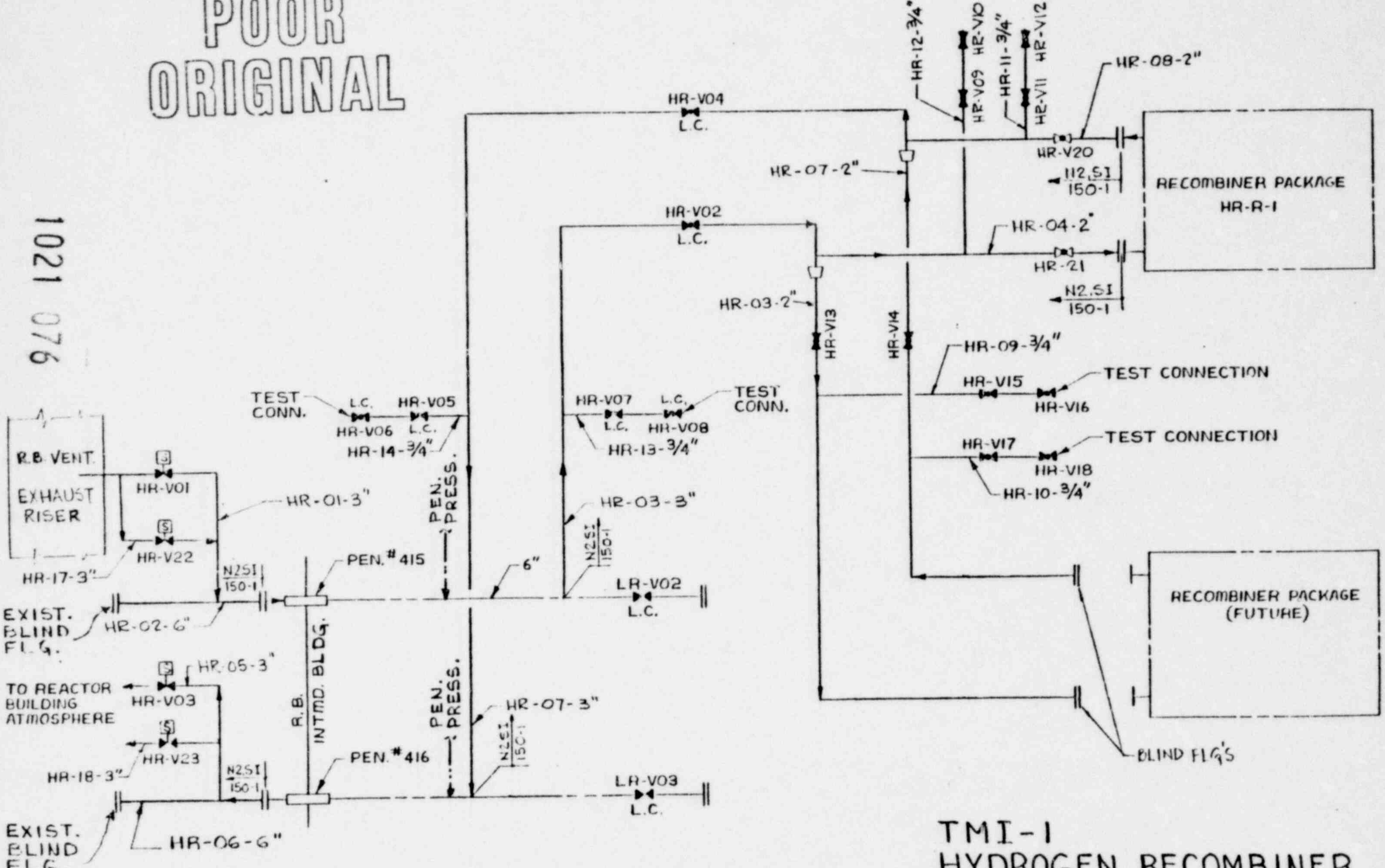


1021 074



TMI-1
HYDROGEN RECOMBINER
ARRANGEMENT
INTERMEDIATE BLDG.

FIG. No. 2.1-6



TMI-1
HYDROGEN RECOMBINER
SYSTEM DIAGRAM

FIG. No. 2.1-7

DRAWN CHK'D.

DESIGN LEADER ENG.

MANAGER APPROVAL

ENG. MECH.

Service

DWG. NO.

REV.

SCALE:

AUTH. NO.

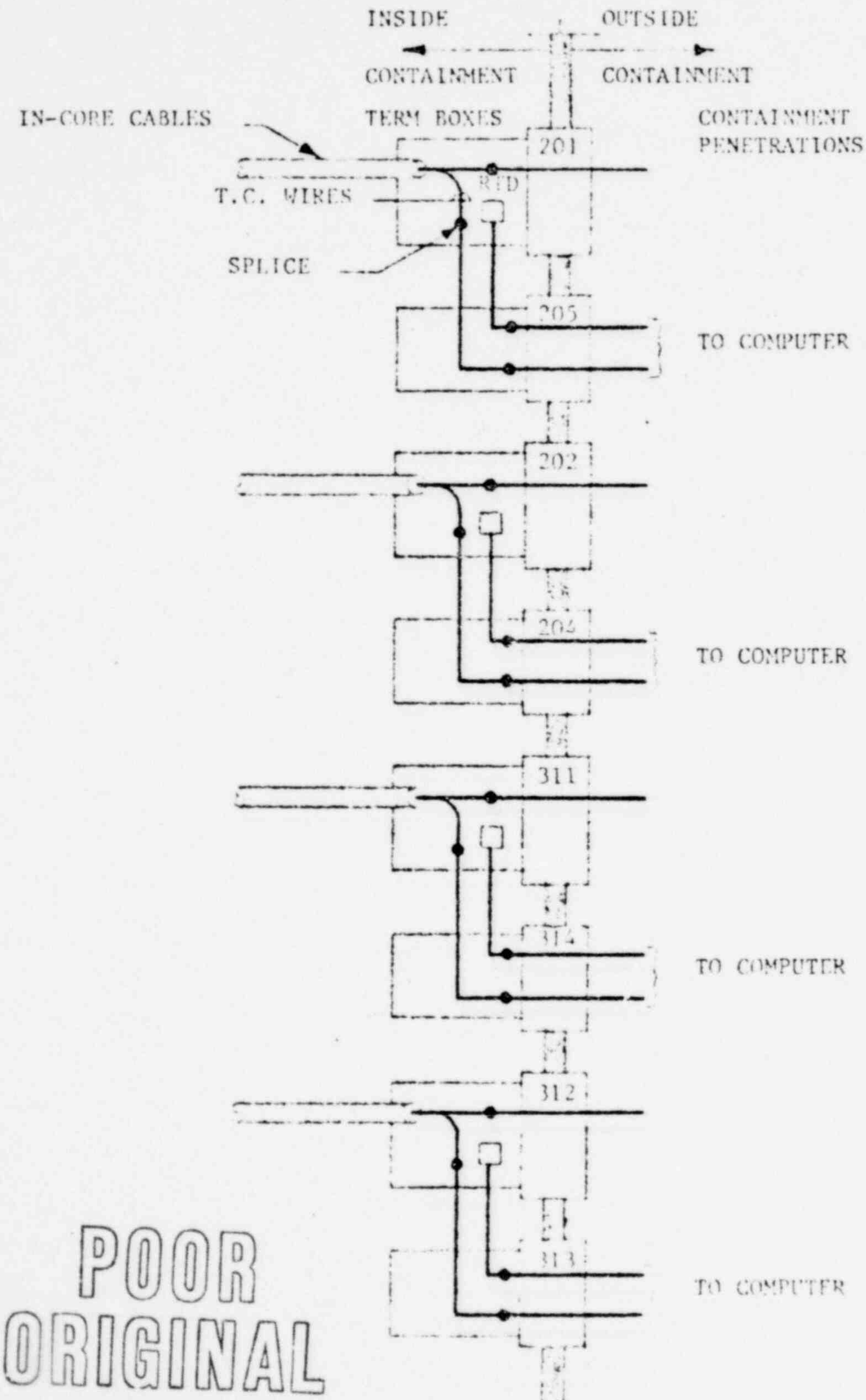


Figure 2.1-8

CONNECTION OF IN-CORE THERMOCOUPLES TO COMPUTER

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DWG. NO.

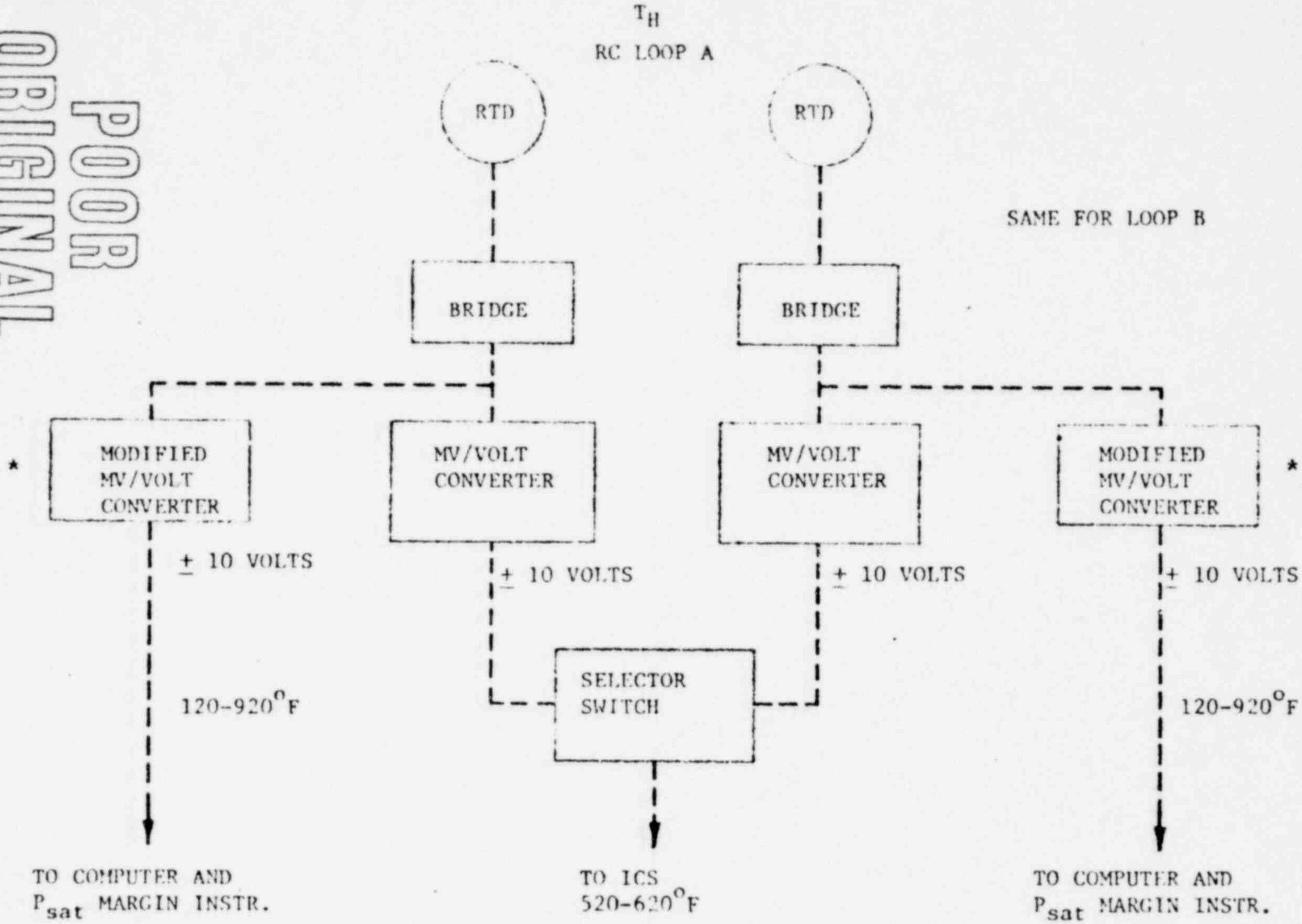
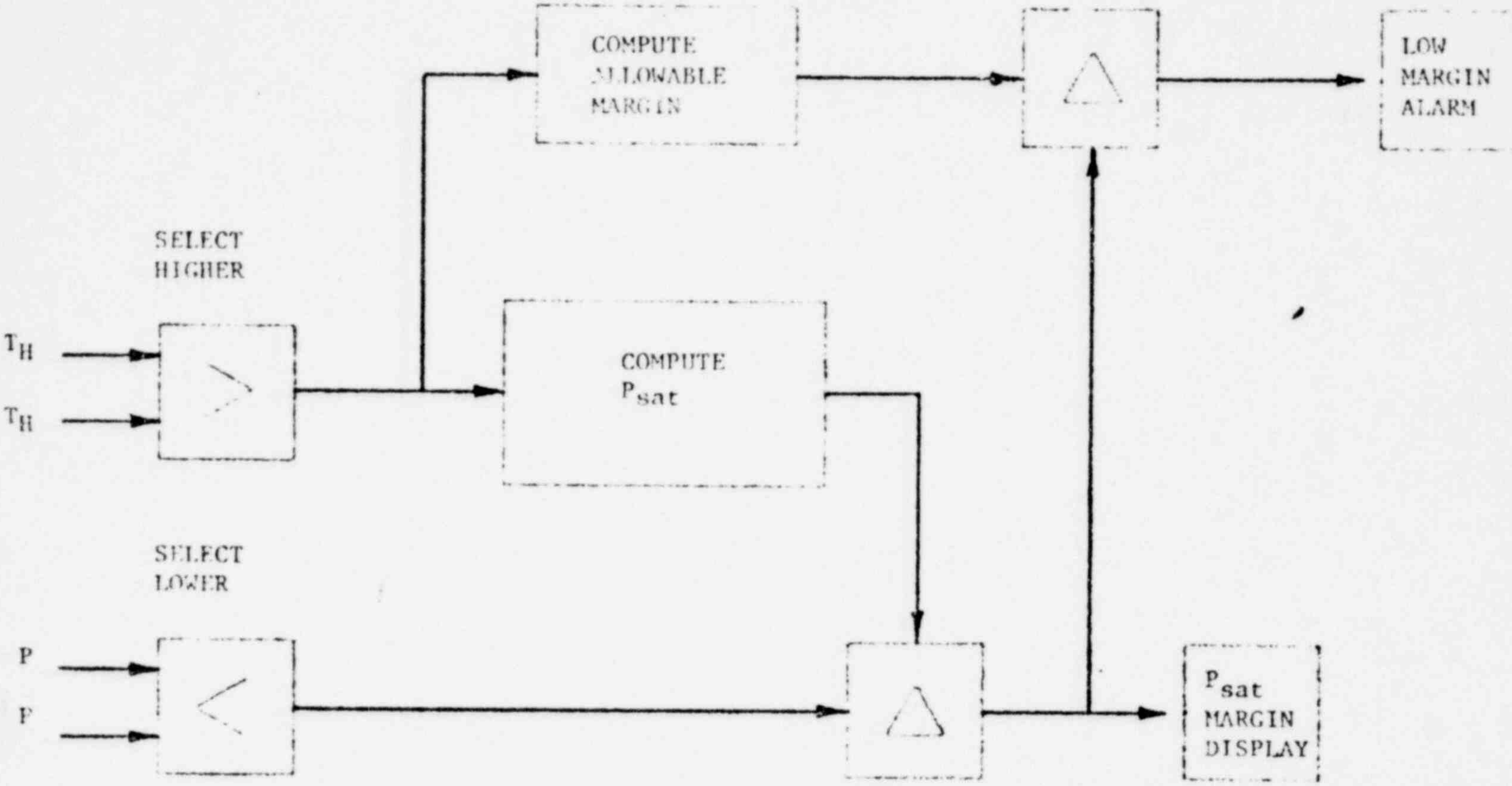


Figure 2.1-9

WIDE RANGE T_H MEASUREMENT



TYPICAL FOR LOOPS A & B

Figure 2.1-10

P_{sat} MARGIN DISPLAY

POOR ORIGINAL

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DRAWN CHK'D.
DESIGN LEADER ENG.
MANAGER APPROVAL
ENG. MECH.
DWG. NO.
SCALE:
AUTH. NO.
REV.
Service

DWG. NO.

THREE MILE ISLAND UNIT NO. 1

Table 2.1-1

List of Isolation Signal Override Capability

		Isolation Signal					
	Penetration No.	Reactor Trip	High Radiation	4 psig Building	30 psig Building	1500 psig (SFAS)	Line Break
Containment Air Sample	108	N/A	N/A	I	N/A	N/A	N/A
R.B. Sump	353	C	I	I	N/A	N/A	N/A
RCDT	330,331	C	I	I	N/A	N/A	N/A
RCS Sample	328	C	I	I	N/A	N/A	N/A
R.B. Purge	336,423	C	NO	NO	N/A	N/A	N/A
RCS Letdown	309	A	I	I	N/A	N/A	N/A
Demin Water	307	C	N/A	C	N/A	N/A	N/A
OTSG Sample	213, 214	C	I	I	N/A	N/A	N/A
NSCCW	346, 347	N/A	N/A	N/A	NO	N/A	I
ICCW	302, 333, 334	N/A	I	N/A	NO	N/A	NO
R.B. Air Coolers	431, 422	C	N/A	C	NO	C	N/A
R.C. Pump Seal Return	329	N/A	N/A	N/A	NO	N/A	N/A

Legend
- C = Common Signal Bypass; initiating isolation condition may still exist.
- I = Individual isolation signal bypass capability; procedures governing override to be developed.
- A = Automatic isolation signal bypass.
- NO = No bypass capability; initiating condition must clear to allow reopening of valve.
- N/A = Not applicable.

Note: For combinations of initiating signals that are allowable, refer to Table 1 of Appendix A.

THREE MILE ISLAND UNIT NO. 1

Table 2.1-2

LIST OF CONTAINMENT ISOLATION VALVES REQUIRING MODIFICATIONS

Penetration No.	Service	System	Valve Tag No.	Valve Type	Line Size, In.	Method of Actuation	Normal Valve Position	Post Accident Position Existing	Post Accident Position Modified	Valve Actual Position Indication	Valve Actuation Signal Source Existing	Valve Actuation Signal Source Modified	Notes
108	Containment Air Sample	RM	CM-V1	Ball	1	Air	Open	Closed	Closed	Yes	1,10	1,6,10	System must be leak rate tested at appropriate pressure (see RS-7).
			CM-V2	Ball	1	Air	Open	Closed	Closed	Yes			
			CM-V3	Ball	1	Air	Open	Closed	Closed	Yes			
			CM-V4	Ball	1	Air	Open	Closed	Closed	Yes			
213	Steam Generator Sample	CA	CA-V4A	Globe	3/8	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	No B&W recommendation
			CA-V5A	Globe	3/8	Air	Open	Closed	Closed	Yes			
214	Steam Generator Sample	CA	CA-V4B	Globe	3/8	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	No B&W recommendation
			CA-V5B	Globe	3/8	Air	Open	Closed	Closed	Yes			
302	Intermediate Cooling Water Outlet Line	IC	IC-V2	Gate	6	EMO	Open	Closed	Open/Closed	Yes	1,10	4,7,8,9,10	See Note (1) below
			IC-V3	Gate	6	Air	Open	Closed	Open/Closed	Yes			
307	Demin. Water to Reactor Building	CA	CA-V189	Gate	2	Air	Open	Closed	Closed	Yes	1,10	1,5,10	
309	Letdown Line to Purification Demineralizers	MU	MU-V2A	Globe	2-1/2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	
			MU-V2B	Globe	2-1/2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	
			MU-V3	Gate	2-1/2	Air	Open	Closed	Closed	Yes	1,10	1,6,10	
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1	Globe	3/8	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,6,10	
			CA-V2	Gate	3/8	Air	Closed	Closed	Closed	Yes			
			CA-V3	Globe	3/8	EMO	Closed	Closed	Closed	Yes			
			CA-V13	Globe	3/8	EMO	Closed	Closed	Closed	Yes			
329	Reactor Coolant Pump Seal Return	MU	MU-V25	Globe	4	EMO	Open	Closed	Open/Closed	Yes	1,7,10	3,7,8,10	B&W does not address need on radiation signal. Hi Rad alarm will be provided
			MU-V26	Gate	4	Air	Open	Closed	Open/Closed	Yes			
330	Reactor Coolant Drain Tank Vent	WDG	WDG-V3	Globe	2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,10	
			WDG-V4	Gate	2	Air	Open	Closed	Closed	Yes			
331	Reactor Coolant Drain Tank Pump Discharge	WDL	WDL-V303	Gate	4	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			WDL-V304	Gate	4	Air	Closed	Closed	Closed	Yes			
333	Intermediate Cooling Water Supply Line	IC	IC-V4	Gate	6	Air	Open	Closed	Open/Closed	Yes	1,10	4,7,8,9,10	B&W does not address need to classify lines as Seismic Category I. Also see Note (1) below.
334	Intermediate Cooling to CRDM Cooling Coils	IC	IC-V6	Gate	3	Air	Open	Closed	Closed	Yes	1,10	1,4,7,9,10	See Note (1) below.

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Table 2.1-2 (Cont'd.)

LIST OF CONTAINMENT ISOLATION VALVES REQUIRING MODIFICATIONS

Penetration No.	Service	System	Valve Tag No.	Valve Type	Line Size, In.	Method of Actuation	Normal Valve Position	Post Accident Position Existing	Post Accident Position Modified	Valve Actual Position Indication	Valve Actuation Signal Source Existing	Valve Actuation Signal Source Modified	Notes
336	Reactor Building Outlet Purge Line	AH	AH-V1A	Butter-fly	48	Air	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			AH-V1B	Butter-fly	48	EMO	Closed	Closed	Closed	Yes			
346	Reactor Coolant Pump Motor Cooling Water Supply	NS	NS-V15	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10	See Note (1) below
347	Reactor Coolant Pump Motor Cooling Water Return	NS	NS-V4	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10	See Note (1) below
			NS-V35	Gate	8	EMO	Open	Closed	Open/Closed	Yes			
353	Reactor Building Sump Drain	WDL	WDL-V534	Gate	6	Air	Closed	Closed	Closed	Yes	1,10	1,4,5,10	B&W does not address need on radiation signal
			WDL-V535	Gate	6	Air	Closed	Closed	Closed	Yes			
421	Reactor Building Normal Air Coolers Supply Line	RB	RB-V2A	Gate	8	EMO	Open	Closed	Open	Yes	1,10	7,8,10; or 1,2,10; or 1,5,10	Retain 4 psig signal unless coils and piping inside R.B. are made Seismic Category I
422	Reactor Building Normal Air Coolers Return Line	RB	RB-V7	Gate	8	Air	Open	Closed	Open	Yes	1,10	7,8,10; or 1,2,10; or 1,5,10	Retain 4 psig signal unless coils and piping inside R.B. are made Seismic Category I
423	Reactor Building Inlet Purge Line	AH	AH-V1C	Butter-fly	48	EMO	Closed	Closed	Closed	Yes	1,4,10	1,4,5,10	
			AH-V1D	Butter-fly	48	Air	Closed	Closed	Closed	Yes			

Valve Actuation Signal Source

1) 4 psig reactor building pressure isolation
2) 1500 psig (SFAS) isolation
3) Radiation alarm, operator action required
4) High radiation (non-safety) isolation
5) Reactor trip isolation
6) Override capability on individual valves
7) Classify line to Seismic Category I
8) 30 psig reactor building pressure isolation
9) Line break isolation signal or protect from pipe whip and jet impingement
10) Remote manual control

Notes:

(1) See explanation in text of TDR - No. TMI-157 pg. 10, para IV 3) a) ii) and iii) regarding line break isolation. A line break isolation is not required provided the line can withstand, or is protected from, jet impingement and the only pipe whip that can break it is the R. C. piping.

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Table 2.1-2 (Cont'd.)

LIST OF CONTAINMENT PENETRATIONS REQUIRING ISOLATION ON HI-RADIATION

Penetration No.	Service	System	Isolation Valve Tag No.	Radiation Detector Location	Type of Monitor
213 and 214	Steam Generator Sample	CA	CA-V4A -V5A -V4B -V5B	Locate the monitors outside the R.B. on the sampling line downstream of the containment isolation valve and upstream of connection for Turb. Plant sampling	Strap on GM (New)
309	Letdown Line to Purification Demineralizers	MU	MU-V2A -V2B	Utilize existing Rad. Monitor RM/L-1 located outside R.B.	Inline (Existing)
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1 -V2 -V3 -V13	Locate the monitor outside the R.B. between the isolation valve and the sample cooler.	Strap on GM (New)
329	Reactor Coolant Pumps Seal Return	MU	MU-V33A -33B -33C -33D	Locate the online radiation monitor downstream of the containment isolation valves outside of the R. B. for Alarm Operator action is required to close valves.	Strap on GM (New)
330 and 331	Reactor Coolant Drain Tank Vent Reactor Coolant Drain Tank Pump Discharge	WDG WDL	WDG-V3 -V4 WDL-V303 -V304	Locate the monitor on the outside of the tank.	Area Monitor, strap on GM (New)
336 and 423	Reactor Building Outlet and Inlet Purge Lines	AH	AH-V1A -V1B -V1C -V1D	Utilize the existing purge outlet line Rad. Monitor RM/A-9 located outside of R.B.	Inline (Existing)
353	Reactor Building Sump Drain	WDL	WDL-V534 -V535	Locate a liquid radiation monitor in the R.B. Sump	Sump Liquid Monitor (New)
302 333 and 334	Intermediate Cooling Supply & Return	IC	IC-V2,3 -V4,6	Locate the radiation monitor on the 6" IC return line between valve IC-V3 and the 2" pump recirc. line.	Strap on GM (New)

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3.0 PROCEDURAL MODIFICATIONS

3.1 GENERAL

The preparation, review, approval and distribution of procedures at Three Mile Island is accomplished in accordance with the requirements giver in Technical Specification Section 6 and Administrative Procedure 1001 "TMI Document Control."

The TMI Technical Specification establishes a Plant Operations Review Committee (PORC) and the requirement that PORC review each nuclear safety related procedure and administrative policy. The PORC signs Nuclear Safety Related Procedures and recommends approval by the Unit Superintendent. Administrative Procedure 1001 establishes the format, content, review and approval requirement of all procedures. AP 1001 further establishes the requirement that all procedures relating to nuclear safety be reviewed every two years and defines the mechanism for that review.

The PORC is composed of an inter-disciplinary team of engineers and plant technicians who advise the Unit Superintendent on all matters related to nuclear safety. Section 6 of the Technical Specifications defines the composition of PORC as follows:

a. Unit Superintendent
b. Supervisor of Operations
c. Supervisor of Maintenance
d. Unit Electrical Engineer
e. Unit Mechanical Engineer
f. Unit Nuclear Engineer
g. Unit Instrument and Control Engineer
h. Supervisor of Radiation Protection and Chemistry
i. PORC Chairman (Unit Superintendent-Technical Support)
j. Station Engineers assigned by the Unit Superintendent

A Quorum consists of four members, at least one of whom shall be either the chairman or Vice Chairman of the committee and the quorum is limited to no more than one alternate.

To accomplish the review and revision of TMI-Unit 1 procedures they were divided into two groups. The first group (Table 3.1-1) is required to be reviewed prior to restart of Unit 1 and the second group (Table 3.1-2) will be reviewed in a timely manner not necessarily prior to restart. As reference material to accomplish this review the following sources were used.

1. B&W recommendations

2. NuReg 0560 Staff report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactor Designed by Babcock & Wilcox Company.

1021 084

3. NuReg 0578 TMI-2 Lessons Learned Task Force status report and short term recommendations.

4. I&E Bulletins 79-05, 05A, 05B, 05C.

5. ACRS Interim Report #3.

6. Order and Notice of Hearing dated August 9, 1979.

7. ACRS Recommendations.

In addition to the above documents TMI-Unit 1's emergency procedures were used at the B&W Simulator by the operating staff during training exercises. Information and recommendations from these training sessions were fed back as revisions to procedures.

This procedure review was started in early May, even though it was recognized that a need for further revision would occur as systems were modified in preparation for restart of Unit 1 and as recommendations changed. The review and revision of Group 1 procedures (required prior to startup) will be completed before startup. The Group 2 procedure review started in August and is anticipated to be completed in 1980.

3.1.1 Emergency Procedures

The Emergency Procedures have been and are being revised to include the following:

1. An objective statement was added to the follow-up action.

2. Incorporation of the philosophy of re-checking key parameters using alternative indicators where alternives are available.

3. Incorporation of the philosophy of using multiple plant parameters to judge reactor coolant conditions (I&E Bulletin 79-05A, Item 4d).

4. Stressing the heat transfer aspect of maintaining adequate core cooling at all times (I&E Bulletin 79-05A, Item 3 and NuReg 0578, Item 2.1.9b).

5. Incorporation of NRC Bulletin guidance including adequate sub-cooling, immediate trip of RC Pumps, non-defeat of E.S. Equipment unless continued operation results in unsafe plant conditions and recognition and prevention of void formation.

6. Incorporation of the lessons learned task force's recommendation on operator performance during small break loss of coolant accident; improving operator recognition and response to conditions of inadequate core cooling.

3.1.2 Administrative Procedures

The Administrative Procedures have been and are being revised to include the following:

1. Formalizing shift relief procedures through the use of turnover checklists; requiring signatures of both oncoming and offgoing shifts and listing safety related systems removed from or returned to service (I&E Bulletin 79-05A, Item 10 and NuReg 0578, Item 2.2.1c).

2. Incorporation into surveillance procedures major valve and switch position checks of alternate trains of emergency equipment prior to performance of surveillance testing (I&E Bulletin 79-05A, Item 10).

3. Assurance that surveillance procedures require a specific signed switch and/or valve alignment steps to be used to restore emergency systems to service (I&E Bulletin 79-05A).

4. Verification by inspection of the operability of redundant safety related systems prior to removal of any safety related systems from service for maintenance or surveillances and fication by inspection of the operability prior to return to service after testing.

3.1.3 Surveillance/Preventative Maintenance/Corrective Maintenance Procedure

These procedures have been and are being revised to include the following:

1. Assurance that no more than one (1) safety train is defeated during maintenance or surveillance testing (I&E Bulletin 79-05A, Item 10).

2. Incorporation into surveillance procedures major valve and switch position checks of alternate trains of emergency equipment prior to performance of surveillance testing (I&E Bulletin 79-05A, Item 10).

3. Assurance that surveillance procedures require a specific signed switch and/or valve alignment steps to be used to restore emergency systems to service (I&E Bulletin 79-05A).

4. Verification by inspection of the operability of redundant safety related systems prior to removal of any safety related systems from service for maintenance or surveillances and verification by inspection of the operability prior to return to service after testing.

3.1.4 Operating Procedures

The Operating Procedures have been and are being revised to include:

1. Changes necessary to conform to plant modifications

2. Incorporation of a natural Circulation Procedure (I&E Bulletin 79-05B, Item 1)

GROUP 1 PROCEDURES

TABLE 3.1-1

PROCEDURE NO.	TITLE	PROCEDURE NO.	TITLE
AP 1012	Shift Relief and Log Entries	SP 1300-3E	Spent Fuel Cooling Pump Functional Test
EP 1202-4	Reactor Trip	AB 1203-24	Steam Leak
OP 1102-1	Heat Up	OP 1102-4	Power Operation
EP 1202-26A	Loss of Feedwater to OTSG	AP 1041	ISI System List & Retest Requirement
		AB 1203-41	Low Grid Volts
EP 1202-26B	Loss of Feed to One Steam Generator	SP 1300-3F	Motor Driven EFWPS Functional Test
OP 1103-5	Pressurizer Operation	AP 1036	Instrument OOS Control
AB 1203-15	Loss of Reactor Coolant Makeup	EP 1202-3	Turbine Trip
OP 1104-2	Makeup & Purification Demin System	SP 1300-3G	Turb. Driven EFWP Functional Test
AP 1002	Switching & Tagging	AP 1037	Control of Caution and DNO Tags
EP 1202-37	Cooldown from Outside Control Room	EP 1202-5	OTSG Tube Rupture
SP 1300-1	ISI Inspection Program	AP 1044	Prompt Report Procedure
AB 1203-1	Load Rejection	OP 1106-6	EFW
SP 1300-3C	DHCCWP's Functional Test Recirc Mode	EP 1202-11	Hi Activity
AP 1028	Operator at the Controls	OP 1102-10	Shutdown
SP 1300-3A	RB Spray Pump Functional Test	Report	Past Transient Report Review
SP 1300-3B	DHR Pump Functional Test	EP 1202-12	Excessive Rad. Levels
AB 1203-10	Unanticipated Criticality	OP 1102-16	Natural Circulation
SP 1300-3D	DHRW Pump Functional Test	1300-3H	Makeup Pump and Valve Functional Test
OP 1102-2	Startup	OP 1102-11	Cooldown

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POOR ORIGINAL

GROUP 1 PROCEDURES

TABLE 3.1-1

PROCEDURE NO.	TITLE
1300-3I	NSRWP and Valve Functional Test
EP 1202-35	Loss of Decay Heat Removal
EP 1202-14	Loss of Flow
SP 1300-3J	NSCCWP and Valve Functional Test
EP 1202-29	Pressurizer System Failure
OP 1103-11	Drain & N2 Blanketing RC System
SP 1300-3K	RB Emerg. Clg. Functional Test
EP 1202-36	Loss of Instrument Air
SP 1300-3L	Screen Wash Pump Functional Test
OP 1103-2	Fill and Vent
SP 1300-3M	Screen House Vent Pump Funct. Test
EP 1202-2A	Sta Blackout & Sta Blackout with Loss of Both
SP 1300-3N	Chilled Water Pump Func Test
OP 1103-6	RCP Operation
SP 1300-3P	ISI Misc. Valves
EP 1202-6	Loss of RC Coolant
OP 1103-8	Appraoch to Criticality
SP 1300-3Q	ISI Test Valves Normal OP
SP 1300-3R	ISI Test Inaccessible Valves
OP 1104-4	Decay Heat Removal
SP 1301-1	Shift and Daily Checks

PROCEDURE NO.	TITLE
OP 1105-16	RPS, NNI, ESAS, NI Switch Lineups
SP 1303-5.4	EFW Pumps
1106-1	Turbine Generator
1303-11.22	M.S. Isolation Valves
1107-1	Normal Electric
1106-2	Condensate System
S101	F.W. Pump Turbines
1106-3	Feedwater System
S105	Exercise FW Pump Emergency Governor
1107-2	Emergency Elect.
1107-3	Diesel Generator
1010	T.S. Surveillance
OP Procedure	Lock Valve List
AP 1009	Station Organization & Chain of Command
1016	Operations Surveillance
1026/1407-1	Corrective Maintenance
1027	Preventative Maintenance
AP 1013	By Pass of Safety Function & Jumper Control
Draft	Solid System Operation
AP 1004	Emergency Plan & Procedures
AP 1014	Call Standby Personnel to Plant

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1001	Document Control
1003	Radiation Protection Manual
1005	Security Procedures/Plan
1006	TMI Retraining Program
1007	Control of Records
1008	Good Housekeeping
1011	Controlled Key Locker Control
1018	Quality Control Warehousing
1019	Qualification of Personnel Performing Special Procedures
1020	Cleanliness Requirements
1021	Plant Modifications
1022	Control of Measuring Test Equipment
1023	Test Equipment Recall
1024	Control of TMI Q. C. Records
1025	Special Nuclear Material Accountability
1030	Control of Access To Primary System Openings
1032	Dissemation of Information
1033	Operating Memo's and Standing Orders
1034	Control of Combustible Materials (Unit #2 Only)
1035	Control of Transient Combustible Materials
1038	Administrative Controls - Fire Prot. Prog. Plan
?	Unit #2 ISI Systems List and Retest Req. Unit #2 Only
1043	Engineering Change Modifications

PROCEDURE NO.	TITLE
1101-1	Plant Limits and Precautions
1101-2	Plant Setpoints
1101-2.1	Radiation Monitoring System Setpoints
1101-2.2	Transient Combustible List
1101-3	Containment Int. & Access Limits
1101-4	Balance Of Plant Setpoints
1102-12	Hydrogen Addition & Degasification
1102-13	Decay Heat Removal By OTSG
1102-14	Reactor Bldg. Purging & Venting
1102-15	Fill & Drain Fuel Transfer Canal
1103-4	Soluble Poison Concentration Control
1103-15	Reactivity Balance
1103-15	Heat Balance Calculations
1104-1	Core Flood System
1104-3	Condensate Chemical Feed
1104-5	Reactor Building Spray System
1104-6	Spent Fuel Cooling System
1104-8	Intermediate Cooling System
1104-9	Circ. Water (Inc. Nat Draft Cooling Tower and Amertap)
1104-11	Nuclear Service Closed Cooling Water System
1104-12	Secondary Services Closed Cooling Water System

GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1104-13	Decay Heat Closed Cycle Cooling System
1104-14A	Steam Gen Compartment System
1104-14B	Operating Floor Ventilation System
1104-14C	Reactor Compartment System
1104-14D	RB Recirculation System
1104-14E	Industrial Cooler System
1104-14F	Reactor Building Unit Heater System
1104-15A	Aux & Fuel Handling Bldg. Sup & Exhaust System
1104-15B	Spent Fuel Pump Area
1104-15C	Nuclear Service Closed Cooling & Decay Heat Pump Area
1104-16	Penetration Cooling System
1104-18	Makeup Demineralizer Neutralizing Tank Disch. Proc.
1104-19	Control Building Ventilation System
1104-20	Fluid Block System
1104-21	Penetration Pressurization System
1104-22	Cycle Make-Up Pretreatment
1104-23	Cycle Make-Up Demineralizer
1104-24A	Cycle Makeup Water Pretreatment House
1102-24B	Intake Screen & Pump House Ventilation
1104-24C	River Water Intake Chlorin. House
1104-24D	Service Building
1104-24E	Circ. Water Pump House
1104-24F	Circ. Water Chlorinator House
1104-24G	Substation Relay Control House

PROCEDURE NO.	TITLE
1104-24H	Intermediate Bldg. (Inc. Emer. FW Pump Area)
1104-24I	Turbine Bldg. & Heater Bay
1104-24J	Service Water Post Cooling Tower Pump House
1104-24L	Sewage Pumping H & V
1104-23M	Diesel Generator Bldg.
1104-25	Instrument & Control Air System
1104-26	Nitrogen Supply System
1104-26A	Nitrogen Blanketing Feedwater Heaters
1104-27	Waste Disposal - Gaseous
1104-28	Packaging & Solid. of Solid & Liquid Radwaste
1104-28B	Solid Rad Waste Dis. Sys. Compacting Radio-active Waste
1104-28C	Disposal of Dewatered Resin and Precoat
1104-29	Liquid Waste Disposal System
1104-29A	Reactor Coolant Cleanup Proc.
1104-29B	Reactor Coolant Evap. Processes
1104-29C	Spent Fuel Cleanup Process
1104-29D	Decay Heat Cleanup Processes
1104-29E	Bleed & Feed Processes
1104-29H	Transfers, Cleanup and Evap. of Water From Misc. Waste Storage Tank
1104-29I	RC Drain Tank Transfers
1104-29K	Transfers, Cleanup and Evap. of Water From Laundry Waste Storage Tank
1104-29L	Neutralizing Processes
1104-29M	Neutralizing Waste Clean-up and Evap.
1104-29N	Transfers, Cleanup and Evap. of Decant From Spent Resin & Used Precoat Storage Tanks

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1104-29P	Startup and Operation of the RC Evaporator
1104-29Q	Startup and Operation of the Misc. Waste Evap.
1104-29R	Concent. Waste Storage Tank Evap.
1104-29S	Transfers From Waste Evap Condensate Storage Tanks
1104-29T	Rad Waste Transfers From Unit #2 to Unit #1
1104-29U	Not In Use - For Information Only
1104-29W	Misc. Transfers
1104-29Y	Precoat Filter Operation
1104-30	Nuclear River Water
1104-31	Secondary Service River Water
1104-32	Decay Heat River Water System
1104-33	Screen House Equip
1104-34	Turbine Oil Conditioner and Supply
1104-35	Circulating Water Chlorination & Chemical Addition System
1104-36	River Water Chlorination System
1104-37	Mechanical Draft Cooling Tower
1104-38	Reactor Bldg. Emerg. Cool. River Water System
1104-40	Plant Sump & Drainage System
1104-42	Station Services Air
1104-43	Nuclear Plant Sampling
1104-44	Turbine Plant Sampling
1104-45	Fire Protection System
1104-46	Electric Heat Tracing
1104-47A	Reclaimed Water System
1104-47B	Chemical Addition Nuclear

PROCEDURE NO.	TITLE
1104-48	ScreenHouse Ventilation Equipment R.W.
1104-49	Auxiliary System Operating Procedure - Domestic Water
1104-50A	Industrial Waste Treatment System
1104-50B	Industrial Waste Filter System
1104-51	Cation Demineralizer Resin Replacement
1104-52	Resin Regeneration and Replacement for Deborating Demin.
1104-53	Resin Replace. for Waste Evap. Cond. Demin. (Mixed Bed)
1104-54	Loading and Flushing Makeup & Purif. Demineralizers
1104-55	Reactor Bldg. Atmosphere Cleanup System
1104-56	CO_2 Fire Extinguishing System
1104-57	Fuel Oil Storage and Transfer System
1104-58	Sewage Lift System (SL)
1104-60	Preoperation Chemical Cleaning Basin
1105-8	Radiation Monitoring System
1105-9	Control Rod Drive System
1105-10	Computer
1105-11	Sec. Plant & Aux. Sys. - NNI Inst.
1105-12	Communications System
1105-13	Security Systems
1105-14	Loose Parts Monitor System
1105-15	Transient Monitor
1105-16	HI Band Radio System

GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1106-4	Auxiliary Boilers
1106-5	Turbine Bypass
1106-7	Stator Cooling System
1106-8	Hydrogen Seal Oil and Gas System
1106-9	Turbine Lube Oil Pump System
1106-10	Turbine Gland Steam Supply System
1106-11	Isolated Phase Bus Duct Cooling
1106-12	Extraction Steam, Heater Vents & Drains
1106-13	Powdex System
1106-14	Main Steam
1106-15	Main & Auxiliary Vacuum System
1106-16	OTSG Secondary Fill Drain & Layup
1106-17	Turbine High Pressure Fluid
1106-19	Auxiliary Steam Cross Connection
1107-4	Electrical Distribution Panel Listing
1107-5	Electrical Distribution Component Listing

PROCEDURE	TITLE
1202-2	Sta Blackout & Station Blackout With Loss Of Both Diesel Generators
1202-8	CRD Equip Failures - CRD Malfunction Action
1202-13	Plant Response to Penetration of Protected Area
1202-17	Loss of Intermediate Cooling System
1202-30	Earthquake
1202-31	Fire
1202-32	Flood
1202-38	Nuclear Service River Water Failure
1203-5	High Cation Conductivity In The Condensate and/or Feedwater System
1203-7	Hand Calc. For Quad Power Tilt & Core Power Imbal.
1203-10	Unanticipated Criticality
1203-16	RC Pump and Motor Malfunction
1203-19	River Water System Failure
1203-20	N.S.C.C. System Failure
1203-21	S.S.C.C. System Failure
1203-28	Post Accident H_2 Purge
1203-34	Control Building Ventilation System
1203-40	Vibration & Loose Parts Monitor System

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1300-1AD	Continuous Wet Fluorescent Magnetic Particle Procedures for Pressure Retaining Bolting 2" Diameter and Greater
1300-1AG	Ultrasonic Examination of Crum Pressure Housing Welds
1300-1D	Ultrasonic Examination of Reactor Flange to Vessel Weld and Flange Ligaments
1300-1E	Manual Exam of Piping Butt Welds & Long Welds
1300-1F	Ultrasonic Examination of Reactor Closure Head to Flange Welds
1300-1G	Automatic Ultrasonic Examination of Reactor Closure Head Crum Nozzle and Penetration Welds
1300-1H	Ultrasonic Examination of Piping and Nozzle Bi-Metallic Welds
1300-1J	Ultrasonic Examination of Pressurizer Vessel Welds
1300-1K	Ultrasonic Examination of Steam Generator Upper and Lower Head to Tubesheet Welds and Support Structure Welds
1300-1L	Manual Ultrasonic Exam Proc for Steam Gen Nozzle Inner Radius
1300-1Q	Ultrasonic Examination of Assembled Flywheels on Reactor Coolant Pump Motor Rotors
1300-1T	Manual Ultrasonic Exam. Proc for Pressure Retaining Bolting (Bolts, Studs & Nuts)
1300-1U	Ultrasonic Examination of Reactor Vessel Closure Head Pressure Retaining Studs and Nuts
1300-1V3	Inservice Inspection of Reactor Vessel Control Rod Drive Nozzle and Pressure Housing Welds

PROCEDURE NO.	TITLE
1300-1V6	Inservice Inspection of Reactor Vessel Closure Head Studs, Nuts and Washers
1300-1V7	Inservice Inspection of Reactor Vessel Pressure Retaining Bolting Below 2" Dia. Control Rod Drive Gasketed Joint Bolting
1300-1V8	Inservice Inspection of Reactor Vessel Closure Head Cladding
1300-1V10	Inservice Inspection of Reactor Vessel and Internal Structures
1300-1V11	Inservice Inspection of Pressurizer Circumferential and Logitudinal Seam Welds
1300-1V13	Inservice Inspection of Pressurizer Heater Bundles and Manway Bolting
1300-1V14	Inservice Inspection of Pressurizer Bolting Below 2" Dia. Relief and Safety Valve Bolting
1300-1V15	Inservice Inspection of Pressurizer Integrally Welded Vessel Supports
1300-1V16	Inservice Inspection of Pressurizer Vessel Cladding
1300-1V17	Inservice Inspection of Steam Generator Tubesheet to Head Weld
1300-1V18	Inservice Inspection of Steam Generator Inlet and Outlet Nozzles
1300-1V19	Inservice Inspection of Steam Generator Manway Bolting
1300-1V20	Inservice Inspection of Steam Generator Inspection Opening Bolting (Bolting Below 2" Dia)
1300-1V21	Inservice Inspection of Steam Generator Integrally Welded Vessel Supports
1300-1V22	Inservice Inspection of Steam Generator Vessel Cladding

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1300-1V23	Inservice Inspection of Reactor Coolant and Associated System Piping Welds
1300-1V24	Inservice Inspection of Integrally Welded Pipe Supports
1300-1V25	Inservice Inspection of Piping Supports and Hangers not Welded to a Pressure Boundary
1300-1V29	Inservice Inspection of Reactor Coolant Pump Bolting 2" Diameter and Greater
1300-1V30	Inservice Inspection of Integrally Welded Reactor Coolant Pump Support Welds
1300-1V31	Inservice Inspection of Reactor Coolant Pump Support and Restraint Systems
1300-1V32	Inservice Inspection of Valve Internals for Valves Greater Than 3" Normal Size
1300-1V33	Inservice Inspection of Valves Pressure Retaining Bolting Less than 2" in Dia.
1300-1V34	Inservice Inspection of Valve Supports
1300-1V35	Inservice Inspection of Letdown Cooler Primary Manifold Longitudinal Seam and Manifold to Support Sleeve Welds
1300-1V36	Inservice Inspection of Class 2 Support Components not Welded to a Pressure Boundary
1300-1V37	Visual Inspection of ISI Class 3 Systems
1300-1X	Liquid Penetrant Examination Procedure
1300-1Y	Ultrasonic Examination of Letdown Cooler Manifold and Sleeve Welds
1300-1Z	Ultrasonic Exam. of Integrally Welded Pipe Supports
1300-2	Reactor Building Structural Integrity Anchor Bolt Surveillance Program
1300-2X	Inservice Inspection of Class 2 Support Attach. Welds to the Pressure Retaining Boundary

PROCEDURE NO.	TITLE
1300-4A	Hydrostatic Test for ISI
1301-4.1	Weekly Surveillance Checks
1301-5.1	BAMT Temp Channels, RBAST Temp Channel
1301-5.3	Incore Newtron Dectors-Monthly Check
1301-6.7	Monitoring of Silt Buildup In River Water Scrnhouse
1301-7.1	Inspection of Dikes
1301-9.2	Control Rod Program
1301-9.5	Reactivity Anomaly
1301-9.7	House Pump Floor Intake, Silt Accumulation
1301-9.8	Core Power Map Distribution
1301-9.9	Hydraulic Shock & Sway Suppressors
1301-12.1	Fire System Header/Nozzle Inspection
1301-12.2	Hose Station Inspection
1301-12.3	Fire System Hose Station Insp. Freq. 18 Mo.
1301-12.4	Fire System Hose Station Func. Test (Freq. 3 Yrs.)
1302-1.1	Power Range Calibration
1303-1.1	Reactor Coolant System Leak Rate
1303-1.2	R.C. Flow Surveillance
1303-3.1	Control Rod Movement
1303-4.14	R. B. Spray System Logic Channels
1303-4.16	Emergency Power System
1303-4.17	Main Steam Isolator Vlvs Required Interval-Monthly
1303-4.18	4 KV ES BUS Undervoltage Relay Test

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1303-5.1	R.B. Cooling & Isolatn Sys. Logic Channel & Comp. Test
1303-5.2	Loading Sequence & Somponent Test & HPI Logic Channel
1303-5.4	Emergency Feedwater Pumps
1303-5.5	Control Room Emergency Filtering System (OP. Tests)
1303-6.1	R.B. Integrated Leakage Rate Test
1303-6.2	Hydrogen Purge Operating Test
1303-8.1	Reactor Coolant System Test
1303-8.4	Reactor Building Spray System
1303-9.9	Functional Testing of Hydraulic Snubbers
1303-10.1	R.B. Purge System
1303-11.1	Control Rods (Drop Times)
1303-11.4	Refueling System Interlocks
1303-11.6	Spent Fuel Cooling System
1303-11.8	High and Low Pressure Injection
1303-11.9	R.B. Emergency Cooling System
1303-11.10	ES System Emerg. Loading Seq. & PWR Trans. Test
1303-11.16	Decay Heat Removal System Leakage
1303-11.18	R.B. Local Leak Rate Testings
1303-11.19	Turbine Overspeed Testing
1303-11.20	Reactor Building Access Hatch Interlocks Freq. 6 Mo.
1303-11.21	Core Flooding System Valve Operability Test
1303-11.22	Main Steam Isolation Valves
1303-11.24	RB Local Leakage Penetration Pressurization
1303-11.25	R.B. Local Leakage Access Hatch Door Seals
1303-11.26	R.B. Isolation Valve Cycle Test Required Interval Refueling Period
1303-12.3	Venting of MU Pumps and HP I Lines Frequency-Refueling Interval
1303-12.4	Venting of DH Pumps and LP I Lines Refueling Interval
1303-12.13	Fire System-Flush at 2" Drains Deluge/ Sprinkler Syst.
1303-12.14	Fire Protection Instrumentation Non-Supervised Circuts Test
1303-12.16	Fire System Testing Air Tunnel Deluge Funct. Test
1303-12.17	Fire System Testing Miscellaneous Deluge Fuact. Test
1303-12.18	Fire System Nozzle Flair Test
1305-1.1	Weather Station Daily Checks Required Interval Daily
1301-2	Boric Acid Mix Tank OR Reclaimed Boric Acid Tank
1301-3	Reactor Coolant System
1301-3E	Reactor Coolant System (Radiochemical Analysis)
1301-4.4	BWST
1301-4.5	Secondary Coolant Activity
1301-4.8	Primary Coolant Isotopic Iodine Analysis
1301-5.6	Core Flooding Tank Water Sample
1301-5.7	Spent Fuel Pool
1301-6.5	Sodium Hydroxide Tank Concentration
1301-6.6	Sodium Thiosulfate Tank Concentration
1301-9.10	River Water Discharge Sampling

GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1301-4.6	Station Storage Batteries-Required Interval Weekly
1301-5.8	Station Batteries
1301-6.2	Strong Motion Accelerometer Battery Checks
1303-11.11	Station Batteries (Load Test)
1303-12.11 A/B	Halon System Pres. and Weight Checks
1301-8.1	R.B. Annual Inspection
1301-9.1	R.B. Structure Integrity Tendon Surv. Program
1301-9.12	Sulfate Ion Accountability
1301-8.2	Diesel Generator, Annual Inspection
1301-10.1	Internal Vent Valve Inspect. & Exercise
1303-11.2	Pressurizer Code Safety Valves Setpoint Verification
1303-11.3	Main Steam Safety Valves
1302-3.1	Quarterly Calibration Radiation Monitoring Systems
1302-3.2	Strong Motion Accelerometer
1302-5.1 & 5.5	R.C. Temp Channels & Pressure Temp Comparator
1302-5.2 & 5.3	RPS High and Low RC Pressure Channel Calibration
1302-5.4	RC Flux Flow
1302-5.6	Pump - Flux Comparator
1302-5.7	High R.B. Pressure Channel
1302-5.8	H.P. & L.P. Injection Analog Channels
1302-5.10	R.B. 4 PSIG Channels
1302-5.11	R.B. 30 PSIG Pressure Channels

PROCEDURE NO.	TITLE
1302-5.12	Pressurizer Temp. & Level Channels
1302-5.13	Control Rod Absolute Position
1302-5.14	Control Rod Relative Position
1302-5.15	Core Flood Tanks, Pressure & Level Channels
1302-5.17	Makeup Tank Level Channels
1302-5.18	High/Low Pressure Inj. Sys., Flow Channels
1302-5.19	Borated Water Storage Tank Level Indicator
1302-5.20	B A M T Level & Temp Channel
1302-5.21	Reclaimed Boric Acid Mix Tank Level & Temp Channel
1302-5.22	Containment Temperature
1302-5.24	Environmental Monitors Cal.
1302-5.25	R.B. Sump Level
1302-5.26	OTSG Level Channel Calibration
1302-5.27	Sodium Thiosulfate Tank Level
1302-5.28	Sodium Hydroxide Tank Level Ind.
1302-5.32	Waste Gas Compressor Pressure Switch Checks
1302-6	Calib of Non Tech Spec Instr Used For Tech Spec Comp
1302-14.1	Calibration of Inservice Insp. Related Instruments
1303-4.1	RX Protect Sys
1303-4.11	High and Low Pressure Injection Analog Channels
1303-4.13	R.B. Emergency Cooling & Isolation System Analog Channels
1303-4.15	Radiation Mon. Systems
1303-7.1	Intermediate Range Channel
1303-7.2	Source Range Channel

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
1303-11.23	Reactor Bldg. Local Leakage-Fluid Block System
1303-12.5	CO2 Fire Protection System Test
1303-12.8A	Fire Protect Instr. Funct Test Cont Bldg Elev 355'
1303-12.8B	Fire Protect Instr. Funct Test Cont Bldg Elev 338'
1303-12.8C	Fire Protect Instr. Funct Test Cont Bldg Elev 322'
1303-12.8D	Fire Protect Instr. Funct Test Diesel Generators
1303-12.8E	Fire Protect Instr. Funct Test Screen House
1303-12.8F	Fire Protect Instr. Funct Test Aux & Fuel Handl Bldg.
1305-1.2	Wind Speed and Wind Direction Calibration
1305-1.3	Vertical Temperature Calibration
1302-5.30	Diesel Generator Protective Relaying
1302-5.31	4160 V, D & E BUS Undervoltage Relay System

PROCEDURE NO.	TITLE
3301-M1	Fire System Valve Lineup Verification
3301-M2	Emergency Plant Radiation Instrumentation Check
3301-Q2	Specific Gravity Check-Diesel Fire Pumps
3301-R1	Fire Service Diesel Engine Inspection
3301-W1	Fire System Water Source Level Check
3301-W2	Fire System Diesel Battery Check
3302-R1	Emerg Plant Rad Instr Calib Reqd. Inter-Refueling Int
3302-SA1	Meteorological Instrumentation Calibration
3303-A1	Fire System Valve Cycling
3303-A2	Fire System Main Header Flush and Loop Test
3303-M1	Fire Pump Periodic Operation
3303-Q1	Fire Pump Diesel Fuel
3303-R1	Fire Pump Start Circut
3303-R2	Fire Pump Capacity Testing
3303-3Y1	Fire System Capability Test
3325-SA1	Chemical Release Inventory
3391-SA1	Fire Hydrant Inspection
3106-1	Unit #1 & Unit #2 Condensate Cross-Connect

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GROUP 2 PROCEDURES

TABLE 3.1-2

PROCEDURE NO.	TITLE
#1	Procurement
#2	Expenditure and Shipment of Material
#3	Turn In of Material
#4	Material Storage
#5	Material Receipt
#6	Approval of Invoices & Local Purchase Orders
#7	Ultrasonic Calibration Standards Control
#8	Transfer of Material from Unit 2 to Unit 1
#9	Letters of Justification

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4.0 EMERGENCY PLANNING

4.1 Introduction

To be provided later.

5.0 THREE MILE ISLAND NUCLEAR STATION ORGANIZATION

5.1 GENERAL

Metropolitan Edison Company (Met-Ed), Jersey Central Power and Light Company (JCP&L), and Pennsylvania Electric Company (Penelec), operating subsidiaries of the General Public Utilities Corporation (GPU), are joint owners and applicants for the Operating License for the Three Mile Island Unit 1.

The Senior Vice President - Met-Ed, Vice President GPU, is responsible for the control of generating station operations. The centralized control supervision, coordination and planning of all aspects of TMI-1 operation rests with the Vice President, Nuclear Operations of Met-Ed.

5.2 STATION ORGANIZATION

The Vice President, Nuclear Operations utilizes the following management staff in carrying out his responsibilities:

° Director - Technical Support
° Superintendent - Unit 1
° Superintendent - Unit 2
° Manager - Training
° Manager - Support Services and Logistics

The Three Mile Island Nuclear Station organization as shown in Figure 5.2.1 will function in five main areas: Unit 1 operations and preventative maintenance, Unit 2 oeprations and preventative maintenance, technical support, training, and support services including Health Physics, Chemistry, and Security.

The Operations Group under the Unit 1 Superintendent will be responsible for the day-to-day operation of the unit. Unit 1 will have a Shift Foreman directing the operations of each shift through the Control Room operators and Auxiliary Operators. A maintenance force supporting TMI-1 in the areas of electrical, mechanical and instrument control preventative maintenance and surveillance will also report to the Unit 1 Superintendent. This maintenance force will be suppliemented by additional forces under the Director of Maintenance - GPU for corrective maintenance assignments.

The Technical Support Group under the Director - Technical Support will consist of lead engineers in such disciplines as nuclear, mechanical, electrical, and instrument and control engineering to whom other engineers and analysts assigned to TMI-1 will report. In addition, Technical Engineers will be assigned on each operating shift to maintain technical liaison and coordination between operating shift personnel and the technical support engineering staff.

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The Training Department will function primarily in the three main areas of operator training, technician training and accelerated retraining of operators. The operator training section is organized to support both licensed operator and non-licensed operator training. The technician training section will support training of technicians in both the maintenance and health physics areas. The accelerated re-training program section is designed to present an augmented training program as a result of the TMI-2 accident.

The Operations and Maintenance group for TMI-2 will be responsible for the day-to-day operations and preventive maintenance and surveillance of Unit 2.

The Support Services and Logistics group will function in the areas of facilities, office management, personnel, station security and health physics and chemistry.

The following subsections detail the functions and responsibilities of various station supervisory personnel.

5.2.1 Vice President - Nuclear Operations

The Vice President - Nuclear Operations reports to the Senior Vice President - Met-Ed and has the management responsibility for the overall direction of station operations. This responsibility consists of daily operations, maintenance and site engineering related activities.

5.2.2 Unit Superintendent

The Unit Superintendent reports directly to the Vice President-Nuclear Operations and assists him in the overall operation and maintenance of the Unit. He has direct responsibility for operating the unit in a safe, reliable, and efficient manner.

He bears the responsibility for compliance with the operating license. He is responsible for the supervision of the Operations and Preventative Maintenance groups.

5.2.3 Supervisor of Operations

The Supervisor of Operations has the responsibility for directing the actual day-to-day operation of the unit. He reports direclty to the Unit Superintendent. The Supervisor of Operations coordinates operations, related maintenance activities with the Supervisor of Maintenance, and Supervisor of Preventative Maintenance.

5.2.4 Shift Supervisor

The Shift Supervisor is responsible for the broad perspective of station operations during his assigned shift and he reports direclty to the Supervisor of Operations (NUREG 0578-Section 2.2.1.a). He directs the activities of the Shift Foreman on his shift and is cognizant of maintenance activities being performed while he is on duty. The Shift Supervisor has the authority and obligation to shut down the unit if, in his judgment, conditions warrant this action and is responsible for the station during emergency situations from the Control Room until relieved.

5.2.5 Shift Foreman

The Shift Foreman is responsible for the actual operation of the unit during his assigned shift. He reports directly to the Shift Supervisor. He directs the activities of the unit operators on his shift and is cognizant of all maintenance activities being performed while he is on duty. The Shift Foreman on duty has both authority and the obligation to shut down the unit if, in his judgment, conditions warrant this action.

5.2.6 Supervisor Preventative Maintenance

The Supervisor of Preventative Maintenance reports to the Unit Superintendent and is responsible for organizing and conducting preventative maintenance and surveillance for the Unit. Operation relted maintenance activities are coordinated with the Supervisor of Operations. Corrective maintenance for the station will be performed under the direction of the Director of Maintenance - GPU which is shown on Figure 1.2.

5.2.7 Director - Technical Support

The Director of Technical Support will report to the Vice President-Nuclear Operations and is responsible for the coordination of the technical engineering staff including the Nuclear Engineering, Mechanical Engineering, Electrical Engineering, Instrument and Control Engineering and Shift Technical Engineers.

5.2.8 Shift Technical Engineer

The Shift Technical Engineer (NUREG 0579-Section 2.2.1.b) reports to the Director-Technical Support. He is responsible for providing on shift engineering, technical and administrative support to the Operations staff personnel. He will provide direct technical oversight of the plant reactor performance and associated safety systems in order to improve the safety of unit operations and maintenance performance.

5.2.9 Manager of Support Services and Logistics

The Manager of Support Services and Logistics reports to the Vice President - Nuclear Operations. In this position he is responsible for coordination of facility functions such as office management, facilities, personnel, station security and health physics and chemistry.

5.2.10 Supervisor - Radiation Protection and Chemistry

The Supervisor - Radiation Protection and Chemistry reports to the Manager of Support Services and Logistics and is responsible for the radiation protection and chemistry programs for Unit 1 and common station programs. He provides administrative and technical guidance in the ares of radiation protection, radioactive waste, respiratory protection, health physics engineering including ALARA programs, dosimetry control, and chemistry. A separate health physics organization udner the direction of the Manager of Waste Management is responsible for Unit 2 related health physics activities.

5.2.11 Manager - Training

The Manager - Training reports to the Vice President - Nuclear Operations. In this position he is responsible for the operator training, technician training, and accelerated operator retraining. The technical training section will include training for maintenance and health physics technicians. The operator accelerated re-training program is a braod program based upon changes and lessons learned as a result of the TMI-2 accident. The training department will be augmented by outside consultants as necessary (see Section 6.0).

5.3 STATION SUPPORT ORGANIZATION

The facility organization is supplemented by the resources of General Public Utilities. The GPU Station Support Organization, shown in Fugure 5.3-1, will function in the five main areas of: corrective maintenance, TMI-2 recovery, technical functions, environmental health and safety and reliability engineering.

The Director of Maintenance reports to the Senior Vice President-Met-Ed-Vice President - GPU and is responsible for corrective maintenance and construction at Three Mile Island including coordination with outside contractors as necessary.

The Superintendent of Maintenance reports to the Director of Maintenance and is responsible for Met-Ed and contractor corrective maintenance on TMI-1 and 2. Operations related corrective maintenance activities are coordinated through the respective supervisors of maintenance with the unit supervisors of operations.

A Supervisor of Maintenance on each unit reports to the Superintendent of Maintenance and is responsible for Met-Ed corrective maintenance in the mechanical, electrical, and instrument and control areas for the respective units. Lead foreman on each unit in the various maintenance areas report to the supervisors of maintenance as shown in Figure 5.3-1.

The Director of TMI-2 Recovery reports to the Senior Vice President Met-Ed-Vice President GPU and is responsible for the licensing and environmental safety of generating stations.

The Director - Environmental Health and Safety reports to the Senior Vice President Met-Ed-Vice President GPU and is responsible for the licensing and environmental safety of generating stations.

The Director Technical Functions reports to the Senior Vice President-Met-Ed, Vice President GPU. In this position he will be responsible to provide a centralized technical capablity to support generating facilities. This capability will include general mechanical, civil, electrical and instrumentation and engineering mechanics areas to assist i' the solution of plant operating problems. In addition, this position will be responsible for supporting GPU nuclear plants in the areas of nuclear fuel management, process computer, control and safety analysis, and plant operational analysis. In addition, TMI Engineering Management section has been organized to be the focal point for the coordination of all out of plant technical support for TMI operations.

The Director of Reliability Engineering reports to the Senior Vice Presidet-Met-Ed Vice President GPU. In this position, he is responsible for the activities of the systems laboratories and the quality assurance functions including the quality control support on site.

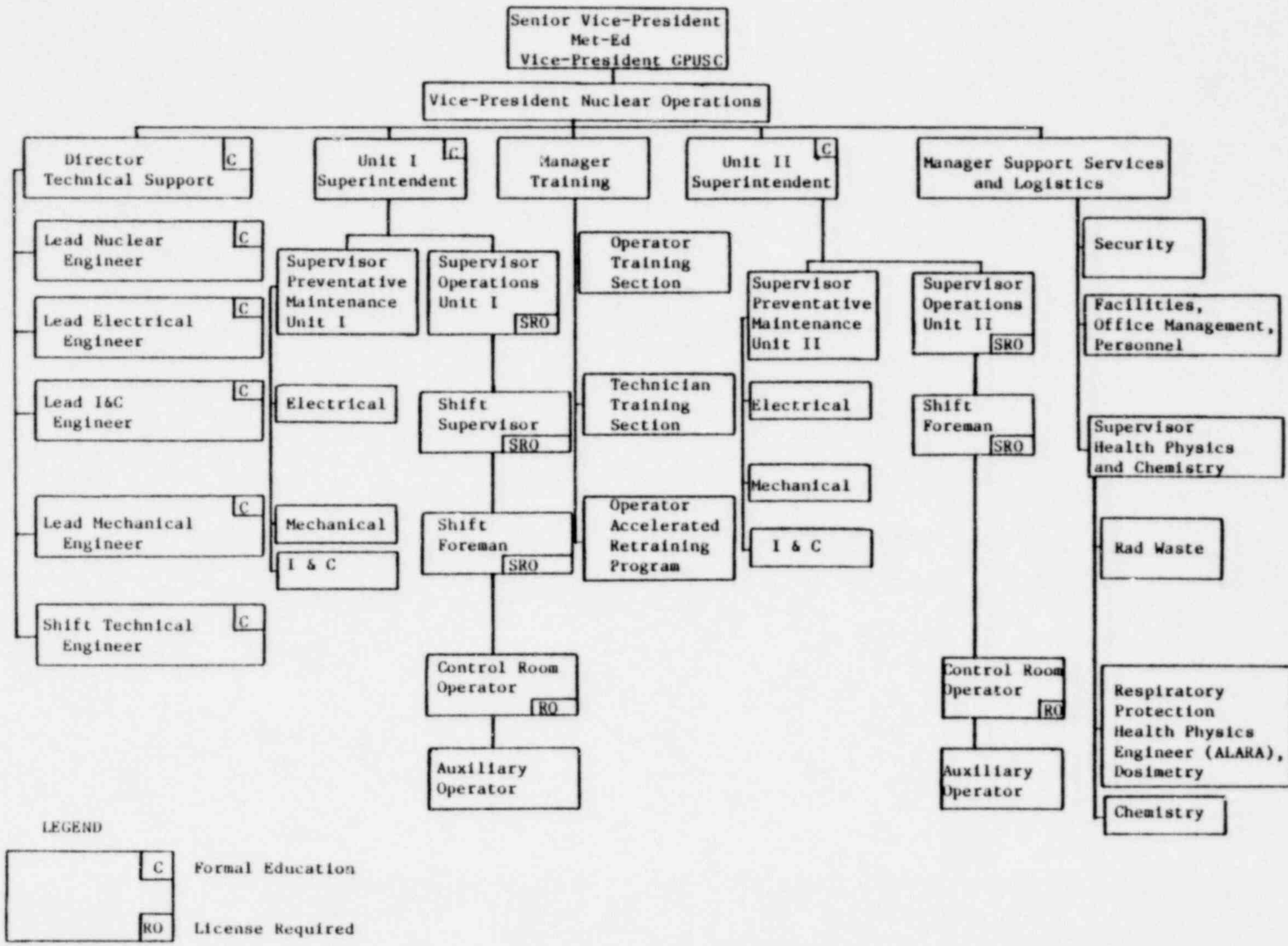

Figure 5.2-1 Station Organization

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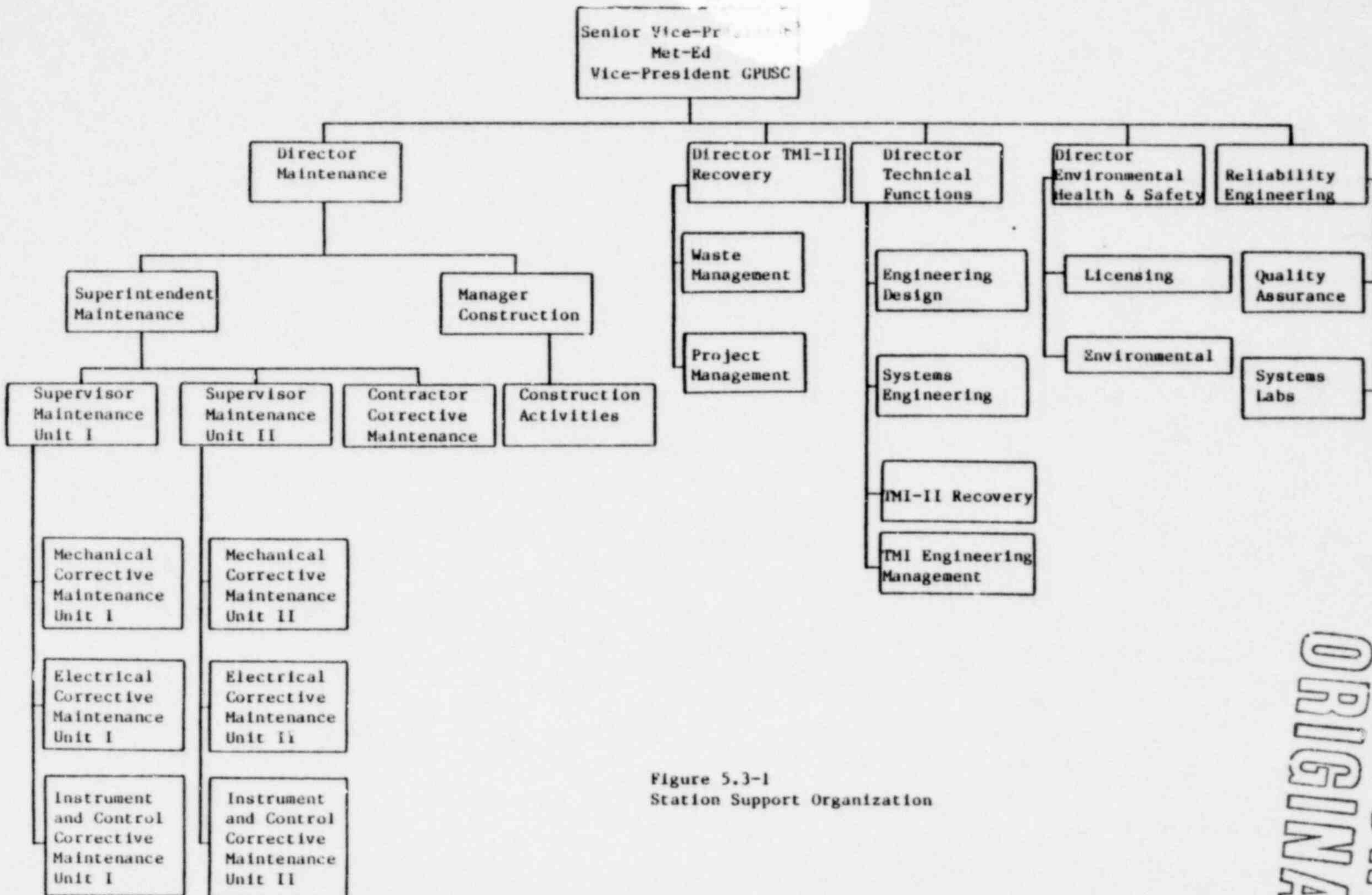

Figure 5.3-1
Station Support Organization

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6.0 OPERATOR ACCELERATED RETRAINING PROGRAM (OARP)

6.1 INTRODUCTION

In preparation for restarting TMI-I, a retraining program for TMI-I Reactor Operators and Senior Reactor Operators is being implemented. Several training issues considered as prerequisites to resuming power operation at TMI-I have been identified and addressed in the Operator Accelerated Retraining Program Objectives. The successful completion of the Operator Accelerated Retraining Program and subsequent evaluation process is required of all personnel who will be assigned as Reactor Operators and Senior Reactor Operators at TMI-I during the resumption of power operation.

The Operator Accelerated Retraining Program includes over sixty (60) presentations and/or practice sessions involving over two-hundred hours of training. Included in the program are at least twenty (20) hours of training directly involved with analyzing and handling abnormal and emergency situations at the Babcock and Wilcox Nuclear Training Center Simulator.

The Operator Accelerated Retraining Program covers topics which can be grouped into four functional areas:

- TMI Plant System Review
- TMI Plant Operational Review
- Radioactive Materials Control
- TMI Plant Transient Analysis

The combination of the Operator Accelerated Retraining Program and the previous TMI-I operator training and requalification programs can enable the safe and effective operation of the Three Mile Island Nuclear Station Unit I.

6.2 PROGRAM OBJECTIVES

The Operator Accelerated Retraining Program is designed to accomplish several objectives relating to enhancing TMI-I Reactor Operator and Senior Reactor Operator performance. The achievement of these objectives is in accordance with the performance standards specified in Section VI (Evaluation Procedure) and is a prerequisite to resuming operation of TMI-I. Program lectures which support the objectives and references for the objectives are listed in Appendix A.

The operator Accelerated Retraining Program objectives are as follows:

A. To improve operator performance during small break loss of coolant accidents.

B. To assure that the operator can recognize and respond to conditions of inadequate core cooling.

C. To improve operator performance during transients and accidents including events that cause or are worsened by inappropriate operator action.

D. To assure that the operators have an in-depth understanding of the TMI-II accident and lessons learned.

E. To assure that operators are knowledgeable of operating procedures and actions required upon initiation of the engineering safeguards features including reactor coolant pump requirements.

F. To assure that operators understand the manometer effects of water levels in the reactor coolant system under different coolant system pressure and temperature conditions.

G. To assure that operators are aware of the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feed-water trains.

H. To assure that operators are aware of the prompt NRC notifications required in the case of serious events and significant events.

I. To provide the operators with an in-depth understanding of the methods required to establish and maintain natural circulation.

J. To assure that operators are knowledgeable of both short and long term plant systems modifications.

K. To provide the operators with a review of the major plant systems.

L. To provide specialized training on "Operations and Procedural Guidance Requirements".

M. To assure operators are fully qualified through the administration of the Company and NRC administered final written and oral examination.

N. To provide the operator with a review of major administrative, normal, abnormal, and emergency procedures.

O. To assure all licensed Unit I operators receive training on the B&W Simulator covering the TMI-II incident.

6.3 TOPICAL OUTLINE

The Operator Accelerated Retraining Program includes over sixty (60) presentations and/or practice sessions covering topics which can be grouped into four (4) functional areas:

- TMI Plant Systems Review
- TMI Plant Operational Review
- Radioactive Materials Control
- TMI Plant Transient Analysis

The program topics include coverage of essential information needed to understand TMI-I plant design and operation. Detailed information on plant systems, operating procedures, and transient analysis are also included to provide an overall understanding of safe nuclear plant operating practices.

A. TMI Plant Systems Review

Topics which provide specific plant systems information address the following areas:

- Features of Facility Design
- Instrumentation and Control
- Safety and Emergency Systems

Presentations covering specific information on system functions, capabilities, limitations, interrelationships and controls are involved.

The specific topics are:

1. Reactor Coolant System
2. Makeup and Purification System
3. Control Rod Drive System
4. Nuclear Instrumentation and In-Core Instrumentation
5. Decay Heat Removal
6. Decay Heat River System
7. Containment Isolation System

A. TMI Plant Systems Review (continued)

8. High Pressure Injection System
9. Nuclear Services Closed Cooling System
10. Decay Heat Closed Cooling System
11. Core Flood System
12. Nuclear Service River Water System
13. Reactor Building Emergency Cooling System
14. Intermediate Closed Cooling System
15. Feedwater System
16. Condensate System
17. Emergency Feedwater System
18. Main Steam System
19. Electrical Distribtution System
20. Emergency Diesel
21. Reactor Protection System
22. Ventilation
23. Hydrogen Recombiner and Hydrogen Purge
24. Emergency Safeguards Actuation System
25. Non-nuclear Instrumentation and Interlocks
26. Computer and Mod Comp
27. TMI-I Short Term Change Modifications
28. TMI-I Long Term Change Modifications

B. TMI Plant Operational Review

Topics which provide information covering the plant general operating characteristics and specific procedural guidance address the following areas:

- Heat Transfer and Fluid Dynamics
- Principles of Reactor Operation and Reactor Theory

B. TMI Plant Operational Review (continued)

- General and Specific Operating Characteristics
- Administrative Procedures, Conditions and Limitations
- Fuel Handling and Core Parameters

Presentations on plant operation are designed to give detailed information on fundamental plant operation and specific procedural guidance. The specific topics are:

1. Heat Transfer and Fluid Dynamics
2. Reactor Theory
3. Use of Procedures
4. Operating Characteristics Review-including natural circulation
5. Solid Plant Operations
6. Operational Chemistry
7. Standard and Emergency Operating Procedures-(covered in nine sections)
   (1) Administrative Procedures
   (2) Limitations and Precautions
   (3) Emergency Procedures
   (4) Emergency Feedwater Procedures
   (5) Reactor Coolant Pump Procedures
   (6) Electrical Power Emergency Procedures
   (7) Primary System Leak Emergency Procedures
   (8) Operating Procedures
   (9) Steam System Emergency Procedures
8. Technical Specifications - Limiting Conditions for Operations
9. Technical Specifications Review
10. Fuel Handling and Core Parameters
11. NRC Prompt Notification Enforcement Policy

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C. Radioactive Materials Control

Topics which provide information covering radioactive materials control address the following areas:

- Radiation Control and Safety
- Radioactive Material Handling, Disposal and Hazards
- TMI Emergency Plan

The specific topics are:

1. TMI Radiation Emergency Plan
2. Radiation Safety and Radioactive Materials Control
3. Radiation Monitoring
4. Radioactive Waste Disposal
5. Liquid and Gaseous Releases

D. TMI Plant Transient Analysis

Topics which provide information covering plant abnormal operating characteristics and plant transients address the following areas:

- TMI-II Transient
- Safety Analysis for TMI-I
- TMI Simulator Training

The specific topics are:

1. TMI-II Transient
2. Small Break Loss of Coolant Accident Operator Guidance
3. Reactor Coolant System Elevations and Manometer Effect
4. Expected Instruments and Plant Response to Transients
5. TMI Control Room Session
6. Safety Analysis Workshop

In addition to these topics, specifically designed training sessions were conducted at the Babcock and Wilcox Simulator Training Center. These training sessions involved discussion of plant transient information and simulator training sessions where specific casualty situations were handled by the trainees.

D. <u>TMI Plant Transient Analysis</u> (continued)

The topics covered included:

1. Power Distribution and Rod Withdrawal Limits
2. Heat Transfer and Fluid Flow
3. Small Break Analysis
4. Safety Analysis
5. Unannounced Casualties (conducted on the simulator)
6. Special program on the B&W Simulator covering the TMI-II accident

6.4 PROGRAM RATIONALE

The selection of topics to be included in the Operator Accelerated Retraining Program was based on several factors. During the program formulation stage, the extensive training curriculum the TMI-I Reactor Operator and Senior Reactor Operator have already completed was balanced with the training needs related to the current TMI-I and TMI-II plant status. Specific sources utilized in identifying program topics include the following areas:

A. Standard references for operator training programs considered in determining course content include:

1. 10 CFR 55 - Operator's License
2. NUREG-0094 - NRC Operator Licensing Guide
3. TMI-I FSAR
4. TMI-I Operator Requalification Program

The topics included in the Operator Accelerated Retraining Program provide for coverage of all the areas in the NRC operators written examination (10 CFR 55.21/22). In addition topics included in the program include lecture requirements in the TMI Requalification Program (10 CFR 55 Appendix A and TMI-I FSAR Section 12).

B. Other Licensed Nuclear Operator Training References

In making specific topic selections for the course content, other information sources for operator training were used. These sources include:

1. NRC Bulletin 79-05, 79-05A, 79-05B and 79-05C
2. Metropolitan Edison Company commitments on operator training (J. Herbein letter to NRC dated June 28, 1979)
3. NRC letter - Order and Notice of Hearing, August 9, 1979.
4. Selected training programs conducted at other Babcock and Wilcox incident nuclear plants since the TMI-II incident.
5. Interviews with TMI Operators
6. TMI-I plant modifications (Short Term and Long Term)
7. TMI-II incident information and other relevent License Event Reports
8. NUREG - 0578 TMI-II Lessons Learned

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6.5 INSTRUCTIONAL PROCEDURE

The Operator Accelerated Retraining Program topics are presented using a variety of instructional techniques. Instructional techniques utilized for particular program topics are selected to build comprehension of nuclear plant fundamentals, develop the ability to analyze and respond to plant operational situations, and ensure understanding of current TMI-I plant conditions and procedural guidance.

In order to achieve the retraining program goals, the instructional techniques utilized will include:

- Classroom Lectures
- Classroom Discussions
- Classroom Working Sessions
- TMI Control Room Training Sessions
- Nuclear Plant Simulator Practice Sessions
  (B&W Simulator Training Center)

A. Classroom Sessions

In preparation for the classroom presentations conducted at TMI, an extensive program development process was completed. This preparation included the involvement of a primary and backup instructor for designated training sessions. Comprehensive lesson plans developed for the training sessions ensure a well directed approach for the presentations.

1. Topic Lesson Plan Preparation

Lesson plans developed for the training sessions are in accordance with a standard format which includes all the elements of a comprehensive presentation and written guidance for carrying out a topic presentation.

Primary instructors assigned to prepare topic lesson plans have technical expertise in the specific areas covered by assigned topics. The primary instructor identified specific lesson plan objectives and developed the lesson plan material.

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A. Classroom Sessions (continued)

Backup instructors assigned to assist in preparing topic lesson plans have experience in developing technical training material.

In addition to assisting in topic lesson plan development the backup instructor also completes a Lesson Plan Development Summary which identifies essential information pertinent to the topic objective, instruction techniques, and evaluation procedures.

The combined development efforts of the primary and backup instructors is reviewed by designated training department staff members at various stages to ensure a well directed, comprehensive topic presentation is adequately supported.

2. Topic Classroom Presentation

Classroom sessions are conducted following the direction provided by the topic lesson plan and lesson plan development summary. In order to ensure a comprehensive coverage of essential information in the classroom presentation, at least two people will be involved with the presentation. The primary instructor (or a designated alternate) will present the topical information. The backup instructor (or a designated alternate) will sit in on the presentation and ensure that the essential topic information is covered during the presentation. This may involve clarifying certain points and asking specific questions related to the topic lesson objectives and support material.

In preparation for the classroom presentations, practice sessions involving the primary and backup instructor (or designated alternates) are conducted as required. The practice sessions involve discussion of lesson material and presentation techniques,

A. Classroom Sessions (continued)

and may include an abbreviated practice presentation of part of the lesson. The practice sessions serve as a means of ensuring that actual lesson presentations will meet required standards and facilitate the achievement of the lesson objectives. The required extent of the practice session will depend upon the experience level of the primary instructor in presenting similar training material.

B. Control Room and Simulator Sessions

The Control Room and Simulator Training sessions are designed to enable hands on application of guidance provided to TMI-I operators. In preparation for these sessions, specific areas of coverage were designated to ensure essential items identified and/or demonstrated for the operators.

1. Control Room Sessions

   A review with the information/instrumentation available in the TMI-I Control Room is addressed in a specific session. This supplements the references made during other topic presentations which interfaced with Control Room features. A tour of the Control Room conducted under the guidance of a lesson plan prepared by a primary and backup instructor team is designed to build the association of operational concept and guidance with actual system controls.

2. Simulator Sessions

   The B&W Simulator Training is included in the program to provide actual practice for the TMI operators in handling plant transient situations.

B. Control Room and Simulator Sessions (continued)

The training practices used during the simulator training sessions enabled the following:

- Detailed use of procedures (including follow-up actions)
- Plant casualties carried out until a stable condition is reached
- Multiple plant casualties simulated
- Watch section members handling casualties as a team, with specific job assignments made
- Casualty conditions analyzed with watchstander input, supervisor deciding course of action and supervisor directing recovery
- Watch section members evaluated as a team on specific casualty response

6.6 EVALUATION PROCEDURE

The Operator Accelerated Retraining Program is evaluated formally and informally in several manners. Continuous informal evaluation is occurring during the training sessions as the instructor and/or backup instructor gauge trainee understanding by asking questions and observing performance.

Formal evaluations of the training program, instructor delivery, trainee performance and trainee knowledge level are also conducted and analyzed. In addition, performance standards are specified for key evaluation processes.

A. Trainee Evaluation of the Program

At the completion of each week of the training program, the trainees are asked to evaluate and comment on the training sessions. This evaluation encompasses the instructors, training materials, presentation techniques, and classroom facilities. Results of these evaluations are a means of measuring the trainees reaction to the training program. Problems which are identified by these evaluations are considered and resolved by the TMI Training Department staff. Necessary changes to the program are factored into subsequent presentations. If a deficiency is deemed to be severe and cannot be otherwise compensated for, parts of the program will be repeated with appropriate modifications incorporated.

B. Presentation Evaluations

Each session of the program will be monitored and evaluated by the session backup instructor or a designated alternate. An Instructor Evaluation Form is completed for the session and a presentation grade computed. To ensure the overall quality of instruction for each session, the following minimum standards are established.

B. Presentation Evaluations (continued)

1. Individual Presentation Standard

   Presentation Grade $\geq$ 2.5 (on a 4.0 scale)

   The Presentation Grade is the average grade of all the individually graded entries on the Instructor Evaluation Form.

2. Topic Presentation Standard

   Topic Grade $\geq$ 3.0 (on a 4.0 scale)

   The Topic Grade is the average grade of all the individual presentation grades for the topic.

Presentations which do not meet the minimum standards will be subjected to the following:

1. Weaknesses found in the presentation will be discussed with the instructor.
2. Key concepts which are not adequately covered in this presentation will be presented again to the trainees in a subsequent training session.
3. Trainee performance on quiz questions on the concepts covered in the presentation will be evaluated. If trainee performance of 70% is found, the entire training session will be repeated for the affected trainees.

C. Knowledge Evaluations by Quiz

Each lesson plan for the program is developed with representative quiz questions identified. During each week of training, quizzes will be administered and utilized for evaluation of trainee knowledge level. The quizzes will meet or exceed the following quiz standards:

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1. Quizzes will be administered each week.
2. Each quiz will consist of at least ten questions.
3. At least 75% of the individual lesson plans presented during the week will have representative questions included in one or more of the quizzes.
4. A variety of question types may be used, but essay questions will predominate. Predetermined quiz question point values will be assigned for evaluation purposes.

Quizzes will be scored and a grade for each quiz determined. To ensure a satisfactory level of understanding of the weekly program material, the following minimum standard is established for each trainee's performance:

1. Individual Quiz Standard

   Individual Quiz Grade $\geq$ 80%

For trainees who do not meet this standard, the following will occur:

1. Trainee will review the program the program material by reviewing the topic lesson plan and/or handouts
2. Trainee will review the material with a designated staff member
   a. Control Room Operators and Shift Foreman will review the material with the Shift Supervisor.
   b. Shift Supervisors and licensed plant management will review the material with a designated instructor.
3. Another quiz will be administered and graded with the same standards in effect. The quiz will cover the material included in the unacceptable quiz (s) and will be composed of questions not previously used during the program.

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D. Knowledge Evaluation by Oral and Written Comprehensive Examination

1) Following the completion of the program, an Auditor Group will conduct a written and oral evaluation of the licensed trainees. The evaluation will be equivalent to an NRC administered licensing examination. It will include an expanded examination section covering the Operator Accelerated Retraining Program objectives. Each successful trainee will be required to pass the audit examination with the minimum examination standard.

For trainees who do not meet this standard, a follow on accelerated requalification program will be implemented. The program content and duration will be dictated by the nature of the audit examination failure. Following completion of the accelerated requalification program, a written and oral evaluation of the trainee will be conducted with the same minimum examination standards in effect.

2) Licensed Unit I personnel, who have successfully completed the Operator Accelerated Retraining Program will finally be required to take an NRC administered oral and written license examination.

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6.7 PROGRAM FORMAT

The Operator Accelerated Retraining Program is developed in over sixty individual lessons involving classroom presentations, TMI Control Room walkthrough and simulator training sessions. The entire program is scheduled for completion in seven modules, with a module consisting of 4 to 5 days (8 hr/day) of training.

Structuring the program into modules enable the scheduling of the presentations to occur during the six weeks cycle TMI training shift, or as a full time program. The content of each module is a selected grouping of individual lesson plans which cover material which is related to similar subjects. The modules are identified in Appendix B and are representative of the program scheduling.

A. Simulator Training Module

The initial program training module involved four and one-half days of training at the Babcock and Wilcox Nuclear Training Center. The module content included classroom training sessions and Control Room operational sessions. The individual topics were:

1. Power Distribution and Rod Withdrawal Limits (4 hours)
2. Heat Transfer and Fluid Flow (4 hours)
3. Small Break Analysis (4 hours)
4. Safety Analysis (4 hours
5. TMI-II Accident Analysis (4 hours)
6. Unannounced Casualties (16 hours)

The plant casualties included:

a. Natural Circulation Cooldown
b. Total Loss of Feedwater with no Emergency Feedwater (TMI-II Accident)
c. Station Blackout (with diesels)
d. Loss of Coolant Accident
e. Steam Generator Overfeed
f. Steam Generator Tube Leak
g. Steam Leak in the Reactor Building

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A. Simulator Training Module (continued)

The simulator training module provides an overview of guidance for operators which has resulted from analysis of the TMI-II incident and involvement in simulated plant abnormal and emergency conditions. This initial program module supplemented previous operator training and provided a reference point for subsequent program modules dealing with detailed plant systems, operator guidance and nuclear plant fundamentals.

B. TMI Module One

The first module of the program conducted at TMI involved four days of classroom training focused on nuclear plant fundamentals intergrated with specific plant operational characteristics. The individual topics are:

1. Heat Transfer and Fluid Dynamics (16 hours)
2. Reactor Theory (16 hours)

The content of module one provides an in-depth coverage of the fundamental aspects of nuclear reactor control and nuclear reactor heat removal. These topics review principles necessary for understanding the purpose and function of nuclear plant systems, operational procedures and required operator actions for safely operating TMI-I.

C. TMI Module Two

The second module of the program conducted at TMI involves three and one-half days of classroom training covering specific TMI-I plant information on selected plant transients, plant systems and the Radiation Emergency Plan. The individual topics are:

1. TMI-II Transient (4 hours)
2. Reactor Coolant System (5 hours)
3. Make-up and Purification System (4 hours)

C. TMI Module Two (continued)

4. In-Core Instrumentation (1 hour)
5. Control Rod Drive System (4 hours)
6. Nuclear Instrumentation (2 hours)
7. Integrated Control System (4 hours)
8. Radiation Emergency Plan (4 hours)
9. NRC Prompt Reporting Requirements and Enforcement Policy (0.5 hours)

The content of module two provides detailed coverage of the TMI-II Transient which occurred March 28, 1979. This puts into persepctive the plant systems and procedural training sessions included in subsequent program lessons. Detailed plant systems coverage begins in module two with sessions on key primary plant systems.

D. TMI Module Three

The third module of the program conducted at TMI involves four and one-half days of classroom training covering specific TMI-I plant systems and operational procedures. The individual topics are:

1. TMI-I Short Term Modifications (4 hours)
2. Decay Heat Removal System (1 hour)
3. Decay Heat Closed Cooling System (1 hour)
4. Decay Heat River System (1 hour)
5. Core Flood System (1 hour)
6. Containment Isolation (1 hour)
7. High Pressure Injection (1 hour)
8. Use of Procedures (2 hours)
9. Nuclear Service Closed Cooling System (1 hour)
10. Nuclear Services River Water System (1 hour)
11. Reactor Building Emergency Cooling System (1 hour)

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D. TMI Module Three (continued)

12. Intermediate Closed Cooling System (1 hour)
13. Feedwater System (1 hour)
14. Condensate System (1 hour)
15. Procedure Review-Reactor Coolant Pump Procedure (2 hours)
16. Emergency Feedwater System (2 hours)
17. Procedure Review-Emergency Feedwater Procedure (2 hours)
18. Main Steam System (1 hour)
19. Electrical Distribution (3 hours)
20. Emergency Diesel (2 hours)
21. Procedure Review-Electrical Power Emergency Procedure (2 hours)
22. Engineered Safeguards Actuation System (4 hours)

The content of module three provides detailed coverage of selected TMI-I primary and secondary plant systems. The systems covered in the program include systems essential to normal and emergency cooling of the reactor.

E. TMI Module Four

The fourth module of the program conducted at TMI involves four and one-half days of classroom training covering specific TMI-I plant systems, operational procedures and radioactive materials monitoring/control. The individual topics are:

1. Procedures Review-Primary System Leak Emergency Procedure (hours)
2. Procedure Review-Steam System Emergency Procedure (2 hours)
3. Reactor Protection System (4 hours)
4. Operating Characteristics Review including Natural Circulation (4 hours)
5. Solid Plant Operations (2 hours)
6. Procedure Review-Emergency Proceudre (2 hours)
7. Procedure Review-Operating Procedures (4 hours)

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E. TMI Module Four (continued)

8. Radiation Safety and Radioactive Materials Control (4 hours)
9. Radiation Monitoring (4 hours)
10. Radioactive Waste Disposal (4 hours)
11. Liquid and Gaseous Releases (2 hours)
12. Operational Chemistry (2 hours)

The Content of module four provides detailed coverage of selected TMI-I systems and plant proceudres. Specific attention is given to normal and abnormal plant operations characteristics and related procedural guidance. Radiation safety, radiation monitoring, and radioactive materials control is covered to review existing guidance and present modifications made at TMI following the TMI-II incident.

F. TMI Module Five

The fifth module of the program conducted at TMI involves five days of classroom training covering specific TMI-I plant systems, operational procedures, technical specifications and plant operational characteristics. The individual topics are:

1. Ventilation (3 hours)
2. Hydrogen Recombiner and Hydrogen Purge (1 hour)
3. Technical Specifications-Limiting Conditions for Operation (4 hours)
4. Technical Specifications-Definitions and Safety Limits (2 hours)
5. Procedures Review-Administrative Procedures and Limitations and Precautions (2 hours)
6. Technical Specifications Review (4 hours)
7. Non-Nuclear Instrumentation and Interlocks (4 hours)
8. Small Break Loss of Coolant Accident Operator Guidance (4 hours)

F. TMI Module Five (continued)

9. Expected Instrument and Plant Response to Transients (4 hours)

10. Reactor Coolant System Elevations and Manometer Effects (2 hours)

11. Fuel Handling and Core Parameters (4 hours)

12. Simulated Transients in Control Room (4 hours)

The content of module five provides detailed coverage of selected TMI-I Systems and plant procedures. Specific attention is given to normal and abnormal plant operating characteristics and related procedural guidance, including plant technical specifications. The TMI-I Control Room is used to develop further relationship between expected plant response to operational situations and actual control instrumentation locations and features.

G. TMI Module Six

The sixth module of the program conducted at TMI involves five days of classroom training covering specific TMI-1 plant modifications and extensive coverage of safety analysis for TMI-1. The individual topics are:

1. Computer and Computer Modifications (4 hours)
2. TMI-I Long Range Design Modifications (4 hours)
3. Safety Analysis Workshop (32 hours)

The content of module six provides an overview of specific changes being planned and accomplished at TMI and provides an in-depth presentation of key safety analysis areas and their implication to TMI-I plant operation. The safety analy-

G. TMI Module Six (continued)

sis training will cover several areas of integrated TMI-I plant response to normal and abnormal events and provide guidance in evaluating plant performance in real time. The fundamental principles of plant operation and plant system information will be combined with existing plant data to analyze several categories of potential abnormal operating conditions and categories of plant emergencies.

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7.0 RADWASTE MANAGEMENT

To be submitted later.

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8.0 SAFETY ANALYSIS

8.1 INTRODUCTION

Changes affecting the acceptance criteria for the TMI-1 FSAR safety analyses arise from several sources. First is the TMI-1 "Order and Notice of Hearing" (Reference 19) which contains NRC staff recommendations that certain changes be made to the plant. This order encompasses recommendations made in NRC bulletins 79-05 A, B and C and from the TMI-2 Lessons Learned Task Force NUREG-0578 (Reference 20). Most of the changes listed below are being made in response to this order. Prior to the TMI-2 accident, B&W 177 FA plants received orders requiring modifications to the high pressure injection system to accommodate certain small break LOCA's. These changes are being evaluated as well. A third source of changes has originated from plant upgrades that Metropolitan Edison believes to be justified. Some of these modifications were being evaluated prior to the TMI-2 accident on March 28, 1979. Certain analyses will be performed using the TMI-1 RETRAN computer model. These analyses will be selected in light of insight gained from TMI-2. The analyses of interest are:

1. The transition to natural circulation following loss of offsite power.

2. The feedline break accident with regard to functional requirements for the emergency feedwater system.

3. Partial and complete loss of feedwater events and their sensitivity to: PORV setpoint, emergency feedwater flow, and reactor trip on loss of feedwater/turbine trip. TMI-1 has design features which permit a turbine trip without causing a reactor trip. Since the reactor will now be tripped by a turbine trip, the following questions will be addressed in this safety analysis: 1) Should these features remain in the plant or be deleted; and, 2) what is the effect of the retention/deletion on the revised plant design? A second area of interest will be the PORV setpoint. As indicated in Reference 2, B&W analyses indicated that a setpoint of 2450 psig would prevent lifting of this valve for all transients that have been experienced at B&W plants. RETRAN analysis will be performed to determine the Tech Spec allowable setpoints considering setpoint drift, setpoint inaccuracy of both the safety and POR valves and realistic plant response to these events. Finally, the RETRAN analyses will also be used to determine if overcooling can occur as a result of EFW operation following a loss of feedwater event.

   These analyses will also support design decisions affecting plant modifications. A final source of input will come from the Abnormal Transients Operating Guidelines (ATOG) Subcommittee of the B&W 177 FA Owner's Group. Met-Ed expects to be a full participant in this group and to utilize results, as applicable, for TMI-1. The RETRAN model will be used if plant specific analyses are required in developing operator guidelines.

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The decision to investigate additional accidents/transients was arrived at by reviewing the events in Table 8.2-1. These events were developed from the TMI-1 FSAR, the Standard Format and Content Guide, Rev. 3, and NRC requests for additional information from the TMI-1 and TMI-2 dockets.

8.2 AREAS OF INVESTIGATION

The plant modifications which are being investigated are summarized below. They are grouped according to their origin.

8.2.1 Modifications Resulting from the August 9, 1979 Order

1. Change of the reactor protection trip setpoint to 2300 psig from 2390 psig. This lower trip setpoint in conjunction with the higher power operated relief valve (PORV) setpoint of 2450 psig results in a lower likelihood of PORV operation.

2. Loss of feedwater flow initiate a reactor trip.

3. Turbine trip initiate a reactor trip.

4. Emergency feedwater modifications to allow:

    a. automatic initiation of the steam and motor drive EFW upon loss of all 4 reactor coolant pumps, feedwater/steam differential and loss of main feedwater.

    b. Loading of EFW pumps on the diesel generators and deletion of the blackout start interlock.

    c. Alternate manual control for EFW system.

5. Modify the emergency feedwater system to start automatically on the following safety grade signals:

    a. low steam generator level

    b. negative differential feed to steam differential pressure.

    c. loss of all four reactor coolant pumps. Since this item is long term, plant safety will be discussed with and without these changes.

8.2.2 Modification as Result of Order of May, 1978

Modifications to the high pressure injection system. The HPI injection lines have been cross connected to assure acceptable results from a break in a high pressure injection line. Cavitating venturis have been added to provide the proper flow split in the event of an HPI line break.

8.2.3 Modification Originating from within Met-Ed

1. Upgrade instrument and valve operator reliability by the addition of heat shrink tubing.

2. The switchover of the ECCS system suction supply from the borated water storage tank (BWST) automatically rather than by operator action.

3. The reactor building spray system will be modified to delete sodium thiosulfate while retaining sodium hydroxide only. Changes will provide more equal drawdown of the BWST and NaOH tanks for a large spectrum of single failures.

4. The fuel handling building, which is presently shared between TMI-1 and 2, will have an airtight barrier partitioning the building.

8.2.4 I&E Bulletin 79-05C

Met-Ed is in the process of evaluating the response to this bulletin. It is expected that a reactor coolant pump trip will be initiated on a SFAS coincident with an indication of a large (in excess of 10-20%) void fraction. This or any other change will be evaluated with regard to their effect on the plant accident and transient analyses and plant operating guidelines.

8.3 EFFECT OF CHANGES ON SAFETY ANALYSIS

Following are summaries of the accidents listed in Table 8.2-1. Table 8.2-1 indicates where FSAR analyses took credit for non-safety grade equipment, or where mitigation is dependent on a specific operating/emergency procedure or design margin. These conclusions will continue to be revised to account for plant design changes.

The event description and mitigating equipment are for the plant design before modification. The modifications discussed in the previous secitons were considered in the review of each accident. If a modification affected that analysis, then a note as to its safety significance was made under the "conclusions" section.

8.3.1 Rod Withdrawal from Startup (FSAR Section 14.1.2.2)

1. Description

   Uncontrolled reactivity excursion starting from a subcritical condition of $1\%\Delta k/_k$ at hot standby.

2. Acceptance Criteria

   i. Limit power to design overpower (112%)

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ii. RCS pressure not to exceed code allowable of 2750 psig.

3. Mitigation

i. RPS trip on high pressure for fast power rises.

ii. Pressurizer code safety valves lift and peak pressure is limited to 2515 psia.

iii. Doppler coefficient provides a negative reactivity addition.

4. Conclusion

The FSAR analysis still bounds the modified TMI-1 plant design. The RCS high pressure trip is lower and safety margins are increased. Since no credit was taken for operation of the PORV, raising the valve setpoint does not change the analysis results. As discussed in Ref. 2, the PORV would lift for the worst case rod withdrawal accident which was analyzed in the FSAR. Nevertheless, the probability of occurrence has been decreased so that safety margins have been improved and lifting of the PORV is not likely for a broad spectrum of rod withdrawal accidents.

8.3.2 Rod Withdrawal at Power (FSAR Section 14.1.2.3)

1. Description

Accidental withdrawal of a control rod group at normal rated power, without ICS control and a 1% shutdown margin.

2. Acceptance Criteria

i. Limit power to design overpower of 112%.

ii. RCS pressure not to exceed code allowable (2750 psig).

3. Mitigation

i. RPS trips on high pressure for slow transients and high neutron flux for fast transients.

ii. Doppler and moderator coefficients provide negative reactivity addition.

4. Conclusions

The FSAR analysis bounds the modified TMI-1 plant design. Lowering of the reactor trip setpoint increases safety margins for this event. Credit was not taken for PORV operation. As discussed in Reference 2, some low worth rod withdrawals can result in PORV actuation. Nevertheless, the probability of such an occurrence is greatly decreased by the changes in the PORV and high pressure trip setpoints.

8.3.3 Moderator Dilution Accident (FSAR Section 14.1.2.4)

1. Description

Diluted makeup water is inadvertently added to the reactor coolant system at a rate of 500 gpm beginning at normal power. RCS boron concentration is at its highest initial value. The result is a reactivity insertion, increased power, pressure and temperature. The addition of one makeup tank volume of unborated water changes the shutdown margin by .8% $\Delta k/_k$.

2. Acceptance Criteria

i. Reactor power will be limited to less than the design overpower (112%).

ii. Reactor coolant system pressure will be limited to less than code allowable 2750 psig.

iii. The minimum shutdown margin will be at least 1% $\Delta k/_k$.

3. Mitigation

i. High pressure or high temperature trip.

ii. Termination of deborated water to makeup tank on reactor trip.

iii. Termination of makeup flow on high pressurizer level.

4. Conclusion

The FSAR analysis bounds the modified TMI-1 plant design. Lowering of the high pressure trip setpoint increases the safety margins for this accident. Operation of the PORV was not assumed in the original analysis, and peak pressure is 2435 psia. Therefore, the PORV setpoint will not be reached during this transient.

Reactor power is limited to 107.3%, and the final shutdown margin is greater than 1% $\Delta k/_k$ even with the most reactive rod stuck out of the core all of the acceptance criteria for this accident are met.

8.3.4 Cold Water Addition (FSAR Section 14.1.2.5)

1. Description

Startup of one or more idle reactor coolant pumps can cuase excess heat removal from the primary coolant system. This cooldown can cause positive reactivity insertions, which

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result in a power rise. The worst case event is the startup of two reactor coolant pumps from 50% power. A tripped rod worth of 1% $\Delta k/_k$ is used in the analysis.

2. Acceptance Criteria

i. Limit overpower to less than the maximum design overpower (112%).

3. Mitigation

i. RPS trip on high pressure for slow power increases or power/flow mismatch for rapid power increases.

ii. RC pump/power monitor limits initial conditions under which event can occur.

4. Conclusion

Lowering of reactor trip setpoint increases safety margins for this event. The FSAR analysis was performed without taking credit for PORV. Peak pressure did not exceed 2400 psia, hence the PORV will not lift during this event.

The FSAR analysis bounds the modified TMI-1 plant design.

8.3.5 Loss of Coolant Flow (FSAR Section 14.1.2.6)

1. Description

Fuel rods experience a limiting DNB transient when all four reactor coolant pumps trip on loss of offsite power or when one pump experiences a locked rotor resulting in an instantaneous loss of flow. The loss of flow analysis is performed from 114% normal power, nominal reactor coolant pump flow, a +2 F core inlet temperature error and a -65 psi error in pressure. Reactor trip delay is assumed to be 620 ms. and a 1% $\Delta k/_k$ subcritical margin is assumed at hot standby. The event is analyzed past the time that the minimum DNBR occurs.

The locked rotor accident is performed from an initial power level of 102% power, with a rampdown in flow from 100% to 75% in 100 ms. Temperature and pressure were the same as for the loss of flow accident. Reactor trip delay is assumed to be 650 ms.

2. Acceptance Criteria

i. DNB is greater than 1.3 for a loss of coolant flow.

ii. DNBR is greater than 1.0 for a locked rotor accident.

3. Mirigation

   i. Protection from four pump coastdown is by limitation of peaking factors, limitations on power level and the pump power monitor.

   ii. Protection for the locked rotor accident is by the flux for flow monitor initiating reactor trip.

4. Conclusions

   The FSAR analysis for the four pump coastdown terminates prior to establishing stable decay heat removal by natural circulation. The EFW system will automatically start and maintain steam generator level at 50% on the operate range. This design should result in the transition to stable conditions; a startup test will be perfromed to demonstrate this transition prior to startup.

8.3.6 Dropped Control Rod (FSAR Section 14.1.2.7)

1. Description

   A dropped control rod reduces the average coolant temperature and reduces power. A return to full power may result in high local power density and heat fluxes. The analysis is performed at rated power with the most adverse values of the moderator and doppler coefficients (EOL) Rod worth are the maximum expected for full power operation with and without Xenon. Tripped rod worth is assumed to be 1% $\Delta k/_k$.

2. Acceptance Criteria

   i. DNBR remains above 1.3.

   ii. Reactor coolant system pressure is less than code allowable (2750 psig).

3. Mitigation

   i. Integrated control system inhibits withdrawal of control rods and ramps secondary side steam demand to 60% rated power to prevent overcooling.

4. Conclusions

   This analysis has not been changed as a result of any of any TMI-1 plant design changes. Analysis results still show that the acceptance criteria are met. It should be noted that while ICS action is assumed in this analysis, acceptable results are not dependent on ICS operation. The dropped control rod analysis performed in the TMI-2 FSAR does not assume ICS action, and demonstrates that the accident acceptance criteria are met.

8.3.7 Loss of Electric Power (FSAR Section 14.1.2.8)

1. Description

Separation of the unit from the transmission network can reult in the trip of the turbine and reactor. As analyzed in the FSAR, severe transient occurs if the ICS does not run back the reactor load demand. The result is reactor trip on high pressure. Cooldown is accomplished through the atmospheric dump or steam relief valves. In the presence of failed fuel and primary to secondary leaks, this event can lead to low levels of radioactivity release.

2. Acceptance Criteria

i. DNBR shall not be less than 1.3.

ii. Reactor coolant system pressure will not exceed code allowable limits of 2750 psig.

3. Mitigation

i. Reactor trip on high pressure.

4. Conclusion

This transient has an increased safety margin over the analysis performed in the FSAR as a result of the high pressure trip setpoint reduction to 2300 psig and the anticipatory reactor trip with turbine trip. In addition, a PORV setpoint of 2450 assures that the PORV will not be activiated (Ref. 1). The addition of the reactor trip signal initiated on turbine trip results in an improved safety margin for this event.

8.3.8 Station Blackout (Loss of AC) (FSAR Section 14.1.2.8)

1. Description

All AC power to the unit is lost, with only battery power available. The reactor and turbine trip, and reactor coolant and feedwater pumps are lost. Core cooling is accomplished through heat rejection to the secondary side using the turbine driven emergency feedwater pump with steam relief to the atmosphere. The analysis is performed starting at full power 2535 Mw (t), and takes credit for a condensate inventory c 200,000 gallons. NNI and ICS instrumentation is taken credit for in controlling the plant when it is powered from the vital ac inverters.

2. Acceptance Criteria

i. DNBR is not less than 1.3.

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ii. Reactor coolant system pressure does not exceed code allowable pressure of 2750 psig.

3. Mitigation

i. Manual control of the steam driven emergency feedwater pump in accordance with plant emergency procedures 1202-2 and 2a.

ii. Steam relief through the atmospheric dump and main steam relief valves.

4. Conclusion

The FSAR analysis of this event remains bounding for the modified TMI-1 plant design. None of the plant modifications being made affect the systems and components which are necessary to mitigate this accident. Since the ICS is powered from the vital ac system, monitoring instrumentation will be powered by the station batteries. The operator will have all of the instrumentation available to bring the plant to a stable shutdown condition.

The extended analysis of this event will determine:

i. When power would have to be restored to maintain stable shutdown.

ii. RCS system pressure response without pressurizer heaters available.

8.3.9 Steam Line Failure (FSAR Section 14.1.2.9)

1. Description

A steam line rupture results in depressurization of the secondary system. This depressurization causes a primary system cooldown causing a DNBR transient and a positive reactivity addition. Blowdown can cause a significant mass and energy addition to containment. Finally, offsite doses can result from the release of secondary side steam to the atmosphere, if steam generator tube leakage exists. The FSAR analysis addresses a variety of break sizes, including the rupture of all four main steamlines outside the reactor building. HPI was not assumed to operate during this event.

2. Acceptance Criteria

i. The core will be maintained in a coolable geometry.

ii. No steam generator tube loss of integrity will result from the pressure/temperature transient.

iii. Offsite doses will be within the limits of 10CFR100.

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3. Mitigation

i. Reactor trip on low pressure or high neutron flux.

ii. Feedwater isolation of the affected OTSG as a result of low steam generator pressure.

iii. Isolation of the unaffected steam generator by the turbine stop valves.

iv. Decay heat removal through the unaffected OTSG by manual control of emergency feedwater (Procedure 2203-2.3) and either atmospheric dump valves or the turbine bypass valves if they are available.

v. Containment temperature and pressures are limited by the containment fan coolers (and reactor building spray systems) if reactor building pressure exceeds 28 psig.

4. Conclusion

Recent, detailed analyses of TMI-2 (Refs. 5 through 8) allow broader conclusions about the acceptability of TMI-1 regarding steam line break. The TMI-2 analysis considered additional single failures, the most limiting were the feedwater regulating and turbine stop valve failures. In addition, the reactor core performance was analyzed assuming that: feedwater is not isolated, offsite power is available if results are worse for that case, and both steam generators blow down outside containment. Reference 3 explains why the TMI-2 core performance analysis bounds Unit 1.

At the Cycle 5 refueling outage, the feedwater latching signal was added to the upstream block valves (FW-V-5A/B). The TMI-2 feedwater regulating valve and turbine stop valve failures cases thus bound the TMI-1 design. Although these failures are not a licensing basis for the plant, they do demonstrate the additional safety margins available in this accident.

The difference in design of the main steam isolation valves between TMI-1 and TMI-2 results in less severe containment transients for TMI-1. The Unit 1 valves are a stop/check design, so that they would prevent the blowdown of both steam generators inside containment. Since TMI-1 does not have cavitating venturi's on the emergency feedwater lines, the operator would have to isolate the affected steam generator to prevent containment overpressure. The operator would have approximately 20 minutes to perform this action.

TMI-1 has not analyzed the environmental effect inside containment for the worst case single failure (because of the stop/check MSIV's, the worst failure is the feedwater regulating valve failure). As noted previously, the blowdown will be less severe than for Unit 2. Although this issue is still being resolved, there are several reasons to expect acceptable results.

i. Heat shrink tubing is being added to splices inside containment. This change was made to TMI-2 prior to receipt of the operating license to resolve this concern.

ii. Much of the equipment which was analyzed and shown acceptable for TMI-2 is also used on TMI-1.

The radiological consequences of the unmitigated steam line break accident have also been addressed on the TMI-2 docket (Ref. 6 and 7). These analysis results demonstrate that worst case doses from a steam line break accident are within the limits of 10CFR100.

8.3.10 Steam Generator Tube Failure (FSAR Section 14.1.2.10)

1. Description

The rupture of a steam generator tube concurrent with 1% failed fuel results in the release of radioactive steam to the environment via the condenser air ejector. Leakage is greater than the capacity of the makeup system, so that the RCS depressurizes.

2. Acceptance Criteria

i. Doses are less than 10CFR100 limits.

3. Mitigation

i. Reactor trips on low pressure.

ii. High pressure injection initiates and maintains primary system pressure and inventory.

iii. Turbine trip isolates the steam generator, and the release path of steam to the environment is via the turbine bypass line, through the condenser to the air ejector.

iv. Cooldown is achieved first via the unaffected steam generator and then through the decay heat cooling system.

4. Conclusions

There have been no plant changes which could change the results of this analysis. Results are still valid and acceptable.

8.3.11 Fuel Handling Accident (FSAR Section 14.2.2.1 and Reference 8 through 10)

1. Description

Failure of a spent fuel assembly, either in the fuel handling building or inside the containment building can result in release of radioactivity to the environment. The fuel handling accident in the fuel handling building considers a 72 hr. decay period for the fuel with the release of gap activity from the entire row of fuel pins on one assembly. 100% of the noble gases and 1% of this iodine inventory is released from spent fuel pool. The fuel handling accident inside containment assumed failure of an entire assembly, filtration by the refueling canal water, and release via the purge exhaust filtration system.

2. Acceptance Criteria

i. Doses should be appropriate within the guidelines of 10CFR100 (less than 100 REM).

3. Mitigation

i. Filtration of releases through the fuel handling building ventilation system.

ii. Filtration of releases by the purge exhaust filter system for the accident inside containment.

iii. Meteorological dispersion of 6.8×10^{-4} sec/m^3 for the accident inside containment.

4. Conclusion

The plant design changes do not affect the mitigation of the fuel handling accident inside containment. Results are still within the acceptance criteria.

The partitioning of the fuel handling building between Unit 1 and Unit 2 does not affect the consequences of this accident because each unit has its own HVAC system. A Unit 1 fuel handling accident would still be mitigated by the Unit 1 ventilation system.

8.3.12 Rod Ejection Accident (FSAR Section 14.2.2.2)

1. Description

Failure of a pressure barrier component could result in the rapid ejection of a control rod from the core. A power excursion and leakage of radioactive primary system fluid to the secondary side would result. Release to the environment

2. Acceptance Criteria

i. The reactor coolant pressure boundary is not further degraded as a result of the ejected rod (no reactor vessel deformation).

ii. Offsite doses are within the limits of 10CFR100.

iii. Radially averaged enthalpy should not be greater than 280 cal/gm at any axial location in any rod.

3. Mitigation

i. The power excursion is limited by the Doppler coefficient.

ii. The power excursion is terminated by reactor trip on high pressure or high flux.

4. Conclusions

The lower high pressure trip setpoint results in increased safety margins over the FSAR analysis. Improvements to the containment isolation signal (radiation +Rx trip) make release of fluid from the containment building less likely.

8.3.13 Feedwater Line Break Accident (TMI-2 FSAR, S3-22.49)

1. Description

This event has not been analyzed for TMI-1. The following description is based on FSAR analyses for TMI-2. A loss of feedwater flow results in a loss of heat sink, primary system heatup accompanied by increased pressurizer level and pressure, and reactor trip on high pressure. A loss of feedwater can be the result of partial or complete loss of feedwater flow or a feedwater line break accident. The analysis assumes a complete loss of feedwater due to a break upstream of the first feedwater line upstream check valves. No analysis of loss of feedwater due to pump trip or valve closures was analyzed. The reactor is initially at 2772 Mw(t). Assumptions were made to provide two worst case scenarios one for containment, and one for primary system conditions.

A double ended rupture (with a blowdown area limited by the feedwater nozzle area) was analyzed; steam generators are assumed to have a fouled inventory of 62,500 lbs., and emergency feedwater is assumed to be at full flow within 40 seconds. The loss of feedwater is not directly calculated but taken as a conservative loss of heat demand (100-0% in 5 seconds for the affected generator and 100-0% in 20 seconds for the unaffected generator).

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2. Acceptance Criteria

i. Core thermal power shall not exceed 112% of rated power.

ii. Reactor coolant system pressure shall not exceed code allowable limits of 2750 psig.

3. Mitigation

i. Reactor coolant system trip on high pressure.

ii. The secondary system heat sink is restored by initiation of emergency feedwater to full flow within 40 seconds. Heat removal is through the turbine bypass valves or main steam relief valves.

4. Conclusions

Results of the TMI-2 feedwater line break accident have become bounding for Unit 1 with the addition of a feedwater line break initiating signal. The addition of reactor trip or loss of feedwater increases the safety margin over the TMI-2 analysis. Lowering of the high pressure trip setpoint also increases safety margins. PORV operation was not assumed in the feed line break analysis, so that the increase in the valve setpoint does not affect analysis results. The PORV would actuate for the worst case feedline break accident analyzed in the TMI-2 FSAR.

8.3.14 Waste Gas Decay Tank Rupture (FSAR Section 14.2.2.5)

1. Description

The rupture of a waste gas decay tank would result in radiological releases via the plant ventilation system. The tank contents as calculated assuming the activity evolved from degassing the primary coolant system after operation with 1% failed fuel.

2. Acceptance Criteria

Doses shall not exceed the limits of 10CFR100.

3. Mitigation

Elevated release of activity from the unit vent.

4. Conclusions

This analysis has not been changed as a result of any plant modifications.

8.3.15 Small Break Loss of Coolant Accidents (LOCA)

1. Description

Small break LOCA's are piping ruptures whose break areas range from as small as 0.01 ft.^{2} to as large as 0.5 ft.^{2}. These LOCA's may or may not involve depressurization of the Reactor Coolant System (RCS).

2. Acceptance Criteria

i. Fuel cladding oxidation (metal water reaction) shall not exceed 0.17 times the total cladding thickness.

ii. Peak Cladding Temperature (PCT) shall not exceed 2200°F.

iii. A coolable geometry shall be maintained.

iv. Long term cooling shall be assured.

3. Conclusion

Pursuant to NRC regulations (10CFR50.46) and 10CFR50 Appendix K) B&W performed generic LCOA analyses of their 177 fuel assembly lowered loop plants. Initially this work was performed to meet the Interim Acceptance Criteria (IAC) and documented in BAW-10052. Later, the analyses were revised to the Final Acceptance Criteria (FAC) using the approved Appendix K model (BAW-10104). The FAC analysis results were documented in BAW-10103.

The work performed for BAW-10052 was used as the basis for the small break LOCA lcoation and size sensitivity study and therefore no new work was performed for BAW-10103 other than analysis of three specific break sizes and locations (0.04 ft.^{2}, 0.44 ft.^{2} and 0.5 ft.^{2} break sizes).

In April 1978, B&W identified an error in their ECCS model. The error was also evident in the model used for the BAW-10052 sensitivity studies and therefore the basis for the acceptability of the small break analysis was eliminated. B&W performed additional small break studies using the corrected model. The revised analyses are documented in a letter from J. H. Taylor, B&W to S. A. Varga, NRC dated July 18, 1978. These analyses cover break sizes 0.04, .055, .07, .085, 0.1, 0.15, 0.2, 0.3, 0.13, and 0.17 ft.^{2}.

Key assumptions for the small break LOCA analyses versus the TMI-1 plant specific information are given below:

	BAW-10103 Generic	TMI-1
Reactor Power (MWt)	2772	2335
Reactor Trip (psig)	1900	1900*
RC Pumps (LOOP)	Coastdown	Coastdown
AFW Available**	Yes-40 sec.	Yes-30 sec.
ESFAS HPI (psig)	1600	1500
Operator Action	Yes X-coun	none***
HPI Distribution	70% to Core after 10 min.	70% to core from time zero***
HPI Flow (gpm)	450 at 600 psig	500 at 600 psig

* Variable low pressure at full power.
** Amount assumed for generic analyses 550 gpm which is less than the minimum 900 gpm available for TMI-1.
*** Prior to startup TMI-1 will install HPI injection leg cross connects and flow control devices to eliminate operator action to cross connect HPI and equalize flow in all four injection legs.

In all cases, TMI-1 plant specific information is as conservative or more conservative than the generic assumption.

Since the TMI-2 accident, greater focus has been placed on small break LOCA's and the capability of the ECCS to mitigate them. Problems such as those discussed in the "Michelson Report" where the pressurizer stays full due to the loop seal arrangement despite loss of RCS inventory have been addressed. These studies are documented in B&W's "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" May 7, 1979 (Reference 2). Breaks of 0.01, 0.02, and 0.07 ft.^{2} are analyzed utilizing varying assumptions on the availability and timing of AFW and HPI. These analyses use the same initial assumptions as used in BAW-10103 except that ESFAS is assumed to occur at 1350 psig. Therefore, they are also bounding analyses for TMI-1 except regarding the distribution of HPI flow as discussed below.

In Reference 2, credit is taken for operator action to initiate HPI or EFW. No mention is made as to whether operator action includes time necessary to cross connect HPI as required in B&W's other small break accident analyses. TMI-1 will complete the installation of permanent cross connection of the HPI prior to startup, therefore, operator action will not be necessary. All of the B&W small break LOCA analyses assume essentially equal backpressure for all four HPI injection points. This assumption is the basis for the 70%/30% flow split of HPI (assuming a single failure of one HPI train) between the core and the break respectively, after cross connection is accomplished. Such an equal backpressure would not exist given an HPI line rupture. The back pressure of the broken HPI leg would be essentially zero and therefore the HPI loss out the break would be high resulting in inadequate injection to the core.

At TMI-1 the distribution of HPI flow between the injection ponts can be as poor as 64%/36% compared to the generic assumption of 70%/30%. TMI-1 is rated at a lower thermal power 2535 Mwt versus the 2772 MWt for the generic analysis, therefore, the 64%/36% flow split could likely be shown to be acceptable. The 64%/36% flow split would not be obtained for an HPI line break as explained above, therefore, operator action would be required to isolate the ruptured HPI line. The need to isolate could be determined by observing the individual flow indicators for the HPI legs. The high flow leg would then be isolated. This action would be contrary to the operators instinct and would require considerable judgment since the initial flow imbalance may not be dramatic. Since too great a chance for operator error (error of omission) exists, cavitating venturis will be added to the injection legs to limit flow in any leg to a maximum of 250 gpm. These venturis prevent HPI pump runout on HPI line break and should eliminate the need for oeprator action. The fact that 250 gpm is adequate for core makeup for a break the size of the HPI line (1.5 inches for the thermal sleeve which limits the break size) will be confirmed by analysis. The venturis have an added benefit of balancing flow of the injection legs under other small break conditions, thus TMI-1 is within the bounds of the generic analyses which assume the 70%/30% flow split is achieved.

8.4 SUMMARY AND CONCLUSIONS

Plant modifications to TMI-1 allow the plant analyses to bound the expected plant behavior (see below). In some cases, analysis for TMI-2 have been referenced because they either analyze events that are not in the TMI-1 FSAR (feedline break) or provide additional assura ces of safety margins (steam line break).

1. Raising the PORV setpoint and lowering the high pressure trip setpoint affects all of the pressurization transients in the FSAR. Safety margins are imprrved since the high pressure trip setpoint has been lowered. No credit was taken for operation of the PORV, so that raising the valve setpoint has no effect on the analysis results.

   The combined effect of the PORV and RPS setpoint changes are to decrease the probability of PORV operation. The

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integrity of the primary coolant system will be challenged less frequently, so that this change is in the conservative direction. It should be noted that this modification could result in more frequent plant trips.

2. Reactor trip resulting from loss of feedwater results in improved safety margins for loss of feedwater events and does not degrade plant response for any accidents/transients.

3. Reactor trip as a result of turbine trip increases safety margins for the loss of feedwater or feed line break analyses. The effect of retaining or deleting plant features that permitted this event to occur without a reactor trip is being analyzed.

4. The feedwater line break analysis has additional safety margins since both turbine and feedwater trips result in a reactor trip. This earlier reactor trip will result in a smaller heatup of the plant.

5. The addition of emergency feedwater initiating signals for the feedline break accident makes the TMI-2 feedwater line break accident analysis bounding and conservative for TMI-1.

6. Modifications to the high pressure injection system will allow adequate HPI flow for the spectrum of LOCA's. System performance is not degraded for any other accidents/transients in which HPI flow is initiated.

7. Upgrading of instrumentation inside containment assures that instrumentation will be functional in the postulated accident environments.

8. Automated switchover of the BWST to the recirculation mode provides additional assurance that switchover will occur within the correct level band. Correct operator action had always been assumed in previous LOCA analyses. The automated switchover achieves the same function requirement by means of a safety grade control system.

9. Does calculation will be provided to demonstrate that the requirements of 10CFR100 can be met after sodium thiosulfate is deleted.

10. Partitioning of the fuel handling building does not degrade the capability of the building HVAC to mitigate fuel handling accidents.

11. The transition to natural circulation following a complete loss of feedwater will be demonstrated by a startup test.

12. An analysis of the station blackout will be performed to determine what specific actions would be required to bring the plant to a safe shutdown condition.

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REFERENCES

1. Three Mile Island Unit 1 Nuclear Station, Final Safety Analysis Report, USNRC Docket No. 50-289.

2. "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," Volumes I & II, Babcock and Wilcox, May 7, 1979.

3. "GPUSC Safety Evaluation Report for Three Mile Island Unit 1 Cycle 5 Reload," dated March 1979.

4. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid) on "High Pressure Trip and Pressurizer Code Safety Valve Settings," GQL-0669, April 17, 1978.

5. "Supplement No. 2 to the Safety Evaluation Report by the office of Nuclear Reactor Regulation, Three Mile Island Nuclear Station Unit No. 2, Docket Number 50-320," USNRC, NUREG 0107, dated February, 1978.

6. Letter, Met-Ed (J. G. Herbein) to USNRC (S. A. Varga), on "Analysis of Fuel Performance During a Steamline Break for TMI-2," License No. CPPR-66, Docket No. 50-320, dated November 18, 1977.

7. Letter, Met-Ed (J. G. Herbein) to USNRC (S. A. Varga), on "Response to Staff Questions on Analysis of Fuel Performance During a Steamline Break," dated December 9, 1977.

8. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid) on "TMI-1 Fuel Handling Accident Inside Containment," GQL-0460, dated April 20, 1977.

9. Letter, USNRC (R. W. Reid) to Met-Ed (J. G. Herbein), dated February 4, 1979.

10. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid) on "TMI-1 Fuel Handling Accident Inside Containment, "GQL-0460, dated May 8, 1979.

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

12. USNRC to Met-Ed "Order for Modification of License," Docket No. 50-289, May 19, 1978.

13. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid), on "Small Break LOCA," GQL-0809, May 3, 1978.

14. Safety Evaluation and Environmental Impact Appraisal by the Office Nuclear Reactor Regulation, Supporting Amendment No. 65 to Facility Operating License No. DPR-47, Amendment No. 62 to Facility Operating License No. DPR-55 Duke Power Company, Oconee Nuclear Station, Units Nos. 1, 2 and 3, Docket Nos. 50-269, 50-270, and 50-287, October 23, 1978.

15. TMI-1 Fuel Densification Report, BAW-1389, Babcock & Wilcox, June 1973.

REFERENCES - (Cont'd)

16. GPUSC Safety Evaluation Report of B&W's TMI-1 Cycle 4 Reload Report, dated January 13, 1978.

17. GPUSC Safety Evaluation of B&W's TMI-1 Cycle 3 Reload Report, dated January 21, 1977.

18. Office of Standards Development, U.S. Nuclear Regulatory Commission, Regulatory Guide 1.70, "Standard Format and Content Guide, Rev. 3, LWR Edition.

19. "Order and Notice of Hearing, Docket 50-289," dated August 9, 1979, USNRC.

20. "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

ACCIDENTS/TRANSIENTS CONSIDERED FOR RE-ANALYSIS

TABLE 8.2-1

	Affected by Plant Changes	Depend on Operator Action	Non-Safety Equipment Used	Don't Consider Failure of Non-Safety Equipment	Analyses Need to be more Realistic
Startup Accident	X			X	
Dilution Accident	X		X	X	
Cold Water	X			X	
Loss of Coolant Flow				X	X
Dropped Rod			X		
Loss of AC	X	X	X		
Loss of Elec. Load	X		X	X	
Steam Line Failure			X	X	X
Steam Generator Tube Failure			X	X	
Fuel Handling Accident			X		
Rod Withdrawal at Power	X			X	
Rod Ejection Accident	X				
Small Break LOCA	X		X	X	X
EVENTS NOT ANALYZED IN FSAR					
EFW Inadvertent Initiation	X		X	X	X
Loss of Feedwater	X		X		
Feed Line Break	X				
HPI Line Break	X	X	X		
Loss of Offsite Power	X			X	X

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9.0 DRAWINGS

To be provided later.

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10.0 CROSS REFERENCE TO ORDER RECOMMENDATIONS

10.1 INTRODUCTION

The August 9, 1979 Order and Notice of Hearing issued by the Commission listed numerous actions recommended by the Director of Nuclear Reactor Regulation (NRR). These recommendations are listed in Section 10.2. The section of this report that covers the recommendations is referenced or a response is given.

A number of the recommendations require additional guidance or have been amended/modofied during the course of their development especially those related to I&E Bulletins 79-05A, 05B,a nd 05C and NUREG 0578. Met-Ed is therefore responding to these recommendations as they are currently understood.

10.2 SHORT-TERM RECOMMENDATIONS AND MET-ED RESPONSES

Recommendation	Response
1(a) Auxiliary Feedwater Upgrading	Section 2.1.1.7
1(b) Auxilary Feedwater Operating Procedures	Sections 3.1.1 and 3.1.4
1(c) Control Grade Reactor Trip on Loss of Turbine/FW	Section 2.1.1.1
1(d) Complete Analysis for Small Break LOCA's and Revise Procedures	Section 3.1
1(e) Retraining of all Reactor Operators	Section 6.0
2 I&E Bulletins	
IEB 79-05A	
Item 1	Not Applicable
Item 2	See Section 10.3.1
Item 3	Section 3.1.1
Item 4	Sections 3.1.1 and 6.2
Item 5	Sections 3.1.2 and 3.1.3
Item 6	Section 2.1.1.5
Item 7	Sections 3.1.2 and 3.1.3

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Recommendation	Response
Item 8	Section 11.2.1
Item 9	Section 2.1.1.5.3
Item 10	Section 3.1.3
Item 11	(Later)
Item 12	Section 3.1 Table 3.1-1 (AP 1044)
IEB 79-05B	
Items 1 & 2	Sections 3.1.4 and 6.2
Item 3	Sections 11.2.3 and 8.1 (Based on B&W analyses submitted May 7, 1979)
Item 4	Sections 3.1.1 and 2.1.1.1
Item 5	Section 2.1.1.1
Item 6	Section 3.1 Table 3.1-1 (AP 1044)
Item 7	Section 11.2
IEB 79-05C	
Item 1	Section 3.1.1
Item 2	(Later)
Item 3	Section 3.1.1
Item 4	Sections 3.1.1 and and 6.0
Item 5	See NUREG 0578 Item 2.1.9 below
Item 1 (Long Term)	Sections (later) and 8.2.4
3. Emergency Plan Upgrading	Section 4.0

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Recommendation		Response
4.	TMI-1/TMI-2 Radwaste Ventilation and Sampling Separation	Section 7.2
5.	TMI-1 Radwaste Management Capability	Section 7.3
6.	Organization and Resources	Section 5.0 to be supplemented separately
7.	Financial Qualifications	To be Submitted Separately
8.	TMI-2 Lessons Learned Recommendations - NUREG 0578	
	2.1.1	Section 2.1.1.3
	2.1.2	Met-Ed will participate in this Industry Program to test these valves.
	2.1.3.a	Section 2.1.1.2
	2.1.3.b	Section 2.1.1.6
	2.1.4	Section 2.1.1.5
	2.1.5	Section 2.1.1.4
	2.1.6	(Later)
	2.1.7	Section 2.1.1.7
	2.1.8	(Later)
	2.1.9	Sections 3.1.1, 6.0 8.1 and (Later)
	2.2.1.a	Section (Later)
	2.2.1.b	Section 5.0
	2.2.1.c	Section (Later)
	2.2.2	Section 4.0
	2.2.3	Not Applicable until NRC Regulations are Revised

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10.3 SPECIFIC RESPONSES TO RECOMMENDATIONS

10.3.1 Response to IEB 79-05A, Item 2

TMI Unit Transient Review

The review of previous TMI Unit 1 transients has been completed as requested by I&E Bulletin 79-05A, Item 2. An in-depth study of the reactor trips which were not an integral part of a Startup and Test procedure (TP 800 series) constituted the major portion of this review. Transients such as rod drops and turbine trips without an associated reactor trip were given a brief review although these transients were not similar to the Davis Besse Event (Enclosure 2 of I&E Bulletin 79-05).

There were no significant deviations from the expected performance during these past transients however there were two cases where the system response differed from it's "normal" response to the initiating event/ reactor trip. Pressurizer level exceeded 400 inches as indicated by the Control Room strip chart and the post trip review for Trip #11. Average reactor coolant temperature following Trip #12 dropped to approximately 536°F with a corresponding low level in the pressurizer of 16 inches. The data from both of these trips indicate that no safety limits were exceeded and that prompt post trip recovery followed in each case.

Reactor Trip #11 was initiated by an RPS High Pressure trip as a result of a turbine trip caused by high bearing vibration on August 30, 1974. The high pressurizer level occurred during the attempted run back from 75% FP when the RC average coolant temperature rose from 578° to 606.5°F. Actuation of the PORV in conjunction with the ICS runback kept the reactor on line while operator action to cutback feed flow caused the high rise in Tave. The RC expansion due to the temperature rise in the system resulted in the high level exceeding the nominal range of the pressurizer. The RCS Pressure transient caused by the turbine trip and ensuing action resulted in a variable pressure/ temperature trip. Recovery of RCS pressure, temperature and pressurizer control quickly followed the reactor trip.

In addition to the high pressurizer level during Trip #11, the RC Drain Tank rupture disc was ruptured. This was most likely due to the pre-trip leakage to RC Drain Tank during normal operation compounded by the high energy blowdown through the electromatic relief when the pressurizer was essentially solid. This is not considered to be a deviation from the expected performance since the rupture disc performed as designed to prevent overpressurization of the Drain Tank when the energy input exceeded the cooling capacity of the system.

Reactor Trip #12 on March 30, 1975, from 100% FP was initiated by an RPS High Pressure Trip following a turbine trip caused by a temporary loss of 125 VDC power to the EHC. Pressurizer level dropped approximately 219 inches and Tave dropped to 536°F following the trip; both of these related parameters exceeded their normal response to a turbine trip. The apparent cause was the failure of two main steam safety valves, MS-V21A and MS-V20B, to reseat. This allowed an initial drop in OTSG header pressure to 950 psi until the turbine bypass valves adjusted to compensate for this additional steam relief. Although this was a deviation from the expected performance it is not considered to be significant since no Limiting Conditions for Operation were violated, pressurizer level remained on scale, and the turbine bypass valves were more than adequate to control header pressure.

Summary of Corrective Action

Since the above review did not identify any significant deviations from expected performance, no major corrective actions were undertaken. Minor corrective action such as rechecking of instrument setpoints were performed. Details concerning each transient that was reviewed and the specific corrective actions taken are available on site.

10.4 LONG-TERM NRR RECOMMENDATIONS AND MET-ED RESPONSES

To be provided later.

11.0 TECHNICAL SPECIFICATIONS

11.1 INTRODUCTION

A considerable number of plant modifications are being accomplished in response to TMI-2 Lessons Learned (NUREG-0578), the TMI-1 Order and Notice of Hearing - August 9, 1979, IE Bulletins, and Met-Ed's review of the TMI-2 accident. The hardware modifications are described in Section 2.0 of this report. In some instances, Technical Specification changes are appropriate to account for systems and changes to systems not formerly discussed in the TMI-1 Technical Specifications. These new Technical Specifications to be provided are discussed in Section 11.2. Formal requests to modify the TMI-1 Technical Specifications will be forwarded separately for each area covered in Section 11.2 since certain committee review requirements of the existing Technical Specifications must be completed before final submittal.

11.2 TECHNICAL SPECIFICATION CHANGES

11.2.1 Auxiliary (Emergency) Feedwater (AFW)

The importance of the AFW System was demonstrated during the TMI-2 accident, therefore, Limiting Conditions for Operation and Surveillance requirements are appropriate.

A LCO will be provided requiring an AFW flow path to each Steam Generator (SG) be available at 100% capacity. If a flow path becomes unavailable or if the capacity drops below 100% to each SG, the plant shall be shutdown within 48 hours and placed in a condition not relying on SG's for cooling within 12 additional hours. If a flow path is unavailable to both SG's or if capacity drops below 100% to both SG's, the reactor will be shutdown within one hour and placed in a cooling mode not relying on SG's within an additional 12 hours.

Appropriate surveillance requirements will be provided to specify flow capacity and flow paths as well as appropriate surveillance of instruments.

11.2.2 Reactor Trip on Loss of Feedwater or Turbine Trip

New Technical Specifications will be provided to impose appropriate LCO's and surveillance requirements.

11.2.3 High Pressure Trip Setpoint Reduction

Technical Specification Changes will be proposed reducing the existing setpoint (2390 psig) to 2300 psig.

11.2.4 Containment Isolation Setpoints

The existing Technical Specifications will be changed to specify initiation of containment isolation on reactor trip.

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11.2.5 Hydrogen Recombiner

A new LCO will be provided for this new system (See Section 2.1.1.4) covering operability and surveillance requirements.

11.2.6 TMI-1/TMI-2 Separation

LCO's and surveillance requirements will be proposed to govern the interfaces (e.g., locked valves) between TMI-1 and TMI-2.

11.2.7 Administrative Controls

Section 6.0 of the TMI-1 Technical Specifications will be modified as necessary to account for any organizational and administrative changes (e.g., Shift Engineer).