E. The circuit breaker control switch will then be operated to close the ES circuit breaker feed to the transferred pressurizer heaters when it has been established that bus loading and emergency D/G loading permit doing so.

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 removable element. 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 The PORV is powered from the RED/YELLOW battery. The motor operated block valve (RC-V3) is powered from Valve Control Center 1C which may be connected to either of the two onsite AC power sources. Normally, it is aligned to operate from the RED ES bus. Since DC power is required to open the PORV and no power is required to close it, it is preferable to have the PORV and the block valve supplied from the same power train in order to increase the likelihood that if there is power available to open the PORV, there will also be power available to close the block valve. If, however, the PORV sticks open and the RED AC power source is disabled, Valve Control Center 1C can be transferred to the GREEN bus by means of a switch in the control room, allowing the block valve to be closed.

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 Conrol 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.

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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 recombiner 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.

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 hydrogen recombiner 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, and their control consoles will be below at elevation 295 ft. 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, controlled from the control room.

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 from alternate dc power 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 and Technical Specification requirements.

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2.1.1.5 <u>Containment Isolation Modifications</u>

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, 1600 psig SFAS, or pipe break signal. These signals will assure that radioactive 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 for 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. 2.1-5), Deletion of 4 psig RB Isolation Signal Logic):

   a. Reactor coolant pump seal return valves MU-V25&26, 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, valves IC-V2, 3, 4, & 6, should be isolated in accordance with the logic of Figure 2.2.

   c. Normal fan cooler coils will be isolated by 4 psig reactor building pressure signal and 1600 psig SFAS. Emergency cooling will be initiated by the 1500 psig signal.

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. The override will be to the isolation signal which will not automatically reopen the isolation valves. Operator action to reopen selected containment isolation valves will be required after the signal override has been accomplished. 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 2.1-3.

A common High Radiation alarm shall be provided in the control room for those radiation monitors that provide a high radiation alarm or closure signal.

4. Specific requirements for each containment isolation valve are tabulated in attached Table 2.1-2. 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 2.1-5, to demonstrate that containment integrity will be maintained.

6. Containment isolation signal override capability will be provided in accordance with attached Table 2.1-1 which lists the following types of overrides:

   a. Individual Isolation Signal Override - This override shall be capable of overriding only the specific isolation signal to the appropriate valves associated with only the penetration which it is desired to reopen. This type of override is noted by an "I" on Table 2.1-1. The initiating isolation condition may still exist when utilizing this override.

   b. Common Isolation Signal Override - 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 reopen by the operator. The common isolation signal override shall also provide the override for the individual isolation signal override. This type of override is noted by a "C" on Table 2.1-1. The initiating isolation condition may still exist when utilizing this override.

   c. Individual Isolation Signal Bypass - This bypass shall be capable of bypassing only the specific isolation signal to the appropriate valves associated with only the penetration which it is desired to be maintained open although an isolation signal is initiated. This type of bypass is noted by an "IB" on Table 2.1-1. The initiating isolation signal may exist when utilizing this bypass.

   d. Automatic Isolation Signal Override - 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 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 2.1-1.

e. No Override or 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.

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 1600 psig SFAS signal would not isolate containment for items a, b, c, f, g, h and i. Isolation on 1600 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 now immediately closed by operator action after reactor tripper existing operating procedures.

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 either of these signals does exist, however, the operator can only reopen the letdown valve by overriding or bypassing the closure signal to the valve.

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 tank vent
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 (alarm only)

Intermediate closed cooling water will be alarmed 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 since normally functioning seal water injection provides adequate cooling for the seals. Plant operating procedures will be revised in order to address reinitiation of ICCW cooling of the seals.

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 2.1-1 provides a list of override capability for each of the lines receiving either: reactor trip, high radiation, 4 psig or 30 psig building pressure, 1600 psig R.C. pressure (HPI) or line break isolation signals. Overriding the isolation signal shall not open the containment isolation valves, deliberate operator action shall be required to reopen selected individual valves.

Plant procedures will govern the conditions under which any of these overrides are utilized. In general, the prerequisite for override is a determinatior 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

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condition is achieved, the operator can override 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. The probability of containment isolation occurring on demand is increased.

2. Spurious initiation of an isolation signal will not introduce new accidents into the plant design. Spurious initiation of any of the above signals would not isolate any components that would also be isolated by a spurious initiation of the existing 4 psig building pressure signal.

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.

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2.1.1.7 Auxiliary Feedwater Modifications

2.1.1.7.1 System Description

The TMI Unit #1 Emergency Feedwater System is being modified so that:

1. Both of the motor driven Auxiliary Feedwater (AFW) pumps automatically start upon loss of both main feedwater pumps or loss of four (4) Reactor Coolant Pumps.

2. The motor driven AFW pumps are automatically loaded on the diesel generator during loss of offsite power.

3. Indication is available in the control room of AFW flow to each steam generator.

4. Manual control of the AFW flow to each steam generator independent of the Integrated Control System (ICS) is available to the operator in the control room.

5. Control room annunciation for all auto start conditions of the AFW system is available.

2.1.1.7.2 Design Bases

The TMI-1 Auxiliary Feedwater System (AFW) is being modified so that a single failure will not result in the loss of auxiliary feedwater system function during a Loss of Coolant Accident. To accomplish this the requirements of NUREG-0578 Section 2.1.7a and 2.1.7b will be met. In addition, the emergency feedwater control valves are being modified such that they fail open on loss of instrument air in order to meet the single failure criteria.

2.1.1.7.3 System Design

As indicated in Chapter 10 of TMI Unit #1 FSAR, the Emergency Feedwater System was designed to operate: 1) on loss of all four Reactor Coolant pumps or 2) if both main feedwater pumps fail.

The original system design was based on use of three auxiliary feedwater pumps. One of the three pumps is turbine driven and has a capacity of 920 gpm. The remaining two pumps are motor driven and have a capacity of 460 gpm each. The three pumps are located in the Intermediate Building which is designed to withstand seismic events, tornado, missiles and a hypothetical aircraft incident. The turbine driven pump is physically separated from the motor driven units. One of the motor driven pumps is powered from the class 1E 4160 volt bus 1D while the other motor driven pump is powered from the redundant class 1E 4160 volt bus 1E. The design of the 1D and 1E Bus has been changed so that they continue to supply power to the motor-driven pumps during all loss of off-site

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power conditions with or without ESAS actuation. To limit voltage dip on the diesel generator during loss of off-site power and coincident ESAS actuation condition, the motor driven pumps will be loaded as a block 5 load (i.e. will be loaded 5 seconds after block 4 loading). For a loss of offsite power only motor driven pumps will be loaded 5 seconds after the diesel generator has started. Power to the turbine driven pumps remains unchanged and is described in Chapter 10 of the FSAR.

Both of the motor driven and turbine-driven emergency feedwater pumps receive an auto-start signal on loss of all four reactor coolant pumps or loss of both main feedwater pumps. This is accomplished by utilizing contacts from the Reactor Coolant Pump power monitors and by sensing the differential pressure across the main feedwater pumps. The RC pump power monitors are a safety grade system and are described in chapter 7 of the TMI-1 FSAR. The main feed pump differential pressure sensing equipment is control grade. Both of the above initiation signals and circuits are designed so that a single failure will not result in the auxiliary feedwater system not functioning.

To accomplish this, the actuation system is arranged into two trains. Each train contains two differential pressure switches (one for each main feedwater pump), and four contacts from the RC pump power monitors (one for each pump).

Power for the "A" train is from the 120 V. AC Vital Distribution Panel VBA. Panel VBA can receive power either from the "A" station battery through the 1A inverter or from the "A" diesel generator. The "B" actuation train utilizes redundant pressure switches and RC pump power monitors and is powered from the 120 V. A.C. Vital Distribution Panel VBB. Panel VBB can receive power from either the "B" station battery through the 1B inverter or from the "B" diesel generator.

In addition to the above actuation signals, the turbine driven pump also receives an automatic start signal from the main feed pump trip circuitry. The details of this actuation signal are discussed in chapter 10 of the TMI-1 FSAR.

All three emergency feedwater pumps discharge into a common header. From this common header a separate six inch line delivers water to each steam generator. Each of the two supply lines contains an air operated control valve (EF-V30 A/B).

Under normal operations air for the control of these valves is supplied from the instrument air system. The instrument air system is described in chapter 5 of the TMI-1 FSAR. In the event the main source of instrument air is not available, a back-up source of instrument air has been provided. The back-up air supply is received from a 80 gal. reservoir which is supplied by an 18 SCFM air compressor. Transfer to the back-up air supply is automatic and no operator action is required. The back-up air compressor is powered from the 1A 480V Engineered Safeguards Control Center.

To provide further assurance that emergency feedwater can be delivered when required, the failure mode of control valves EF-V30 A/B is being changed. Currently these valves fail half open on loss of electrical control signal and fail "as-is" on loss of instrument air. The change consists of modification to the operator such that on loss of air, the valves will fail in the open position and remain in this position.

Control valves EF-V30 A/B are controlled by the Integrated Control System. The design of this sytem is described in chapter 7 of the TMI-1 FSAR. Upon loss of all reactor coolant pumps, and/or both feedwater pumps, the IC positions the control valves to maintain steam generator water level. If reactor coolant pumps are available, the ICS controls are set to maintain a 30 inch water level on the start-up range level indicator. If reactor coolant pumps are not available, the ICS maintains steam generator water level at 50% on the operating range level indicator.

The Integrated Control System is a control grade system. It does, however, receive power from the Class 1E power system. Specifically the ICS is supplied from Distribution Panal ATA. This panel can be powered from the station batteries thru inverter 1A and Panel VBA or from ES Control Center 1A through Panel TRA.

Manual Control of the emergency feedwater control valves can be taken from the control room. When manual control is selected all active components of the ICS are bypassed except for the raise/lower voltage circuit. As further assurance that control of the emergency feedwater control valves are available to the operator, an additional manual control station is being provided for each valve. The controls will be located in the control room and will be totally separate from the ICS. Power from the redundant portion of Class 1E power system will be provided to the back-up controls. A functional diagram of the new manual controls is shown in Figure 2.1-3. A new manual loader station for each control valve will be mounted on the control room console. This will allow the operator to manually set a ± 10 volt control signal into the voltage/pneumatic converter in order to control the position of the EFW control valve. An adjacent selector switch connects the signal from the manual loader station to the voltage/pneumatic converter and also replaces the ICS "EL" power supply with an independent 115 volt, 60 hz supply. Thus, if the EFW controls are disabled due to a failure in the ICS or failure of the "EL" power supply, the operator will have the ability to control flow to either steam generator entirely independent of the ICS.

Each of the emergency feedwater supply lines has also been provided with two redundant flow sensing devices. These devices are a sonic flow device as manufactured by Controltron and will be installed downstream of the control valves before the lines enter the containment building. The flow devices are safety grade and have been seismically qualified. The output of the flow devices will transmit the signals to the main control room where meters will be installed to read flow directly. The equipment to be installed will be safety related. Cabling will be routed as

described in Section 7 of the TMI-1 FSAR. The power supply for the instruments will be derived from the vital 120 V power system. Redundant Power supplies will be used for redundant instruments.

A diverse means of monitoring emergency feedwater flow is provided by the steam generator level indicators. These measurements are derived from Bailey type "BY" transmitters which, subsequent to their installation at TMI-1, have been seismically qualified and qualified for operation in a post-LOCA containment environment. One start-up range and one operating range transmitter have been raised higher above the reactor building floor to avoid flooding in a post-accident situation and have had their electrical connections protected to prevent degradation due to moisture. The level instruments are supplied from 1E on-site power sources and their wiring is run in raceways which have been analyzed to assume heat. They will withstand a seismic event.

2.1.1.7.4 System Operation

The TMI-1 Auxiliary Feedwater System is a stand-by plant system which is not used during normal plant start-ups, shutdowns or operation. The system is maintained in stand-by during plant operations and is automatically actuated upon loss of both main feedwater pumps or loss of all four RC pumps. The following table gives actuation time for the system:

Event	Turbine-Driven	Motor-Driven
a) Loss of Feedwater or Loss of RC Pumps	Immediate	5 Sec.
b) Above with loss of off-site power (LOP)	Immediate	15 Sec.
c) Above with ESAS but no LOP	Immediate	20 Sec
d) Above with ESAS and LOP	Immediate	30 Sec

Start-up and test data indicates that the turbine driven pump requires 18 seconds to reach full flow. The motor-driven pumps should be capable of accelerating to full speed in less than 10 seconds. Therefore under worst case conditions emergency feedwater flow should be established within approximately 40 seconds.

Control of auxiliary feedwater flow following initiation is accomplished by the ICS. The ICS controls the injection of auxiliary feedwater to maintain water level in each steam generator to one of two setpoints depending on whether RC pumps are or are not available. Under forced cooling conditions, the ICS controls level to 30 inches on the start-up range since this is sufficient to provide core cooling. However upon loss of forced RCS cooling the ICS controls steam generator level to 50% on the operating range to promote natural circulation with the Reactor Coolant System.

Manual controls in the control room are available for the operator to take control of the EFW flow to either steam generator when needed or in the event of ICS failure.

2.1.1.7.5 Design Evaluations

Table 3-1 (supplement 1 to this RESTART REPORT) indicates that the heaviest loading on one diesel generator during an ESAS actuation would be 2913 Kw and during a loss of offsite power only, the load would be 2817 Kw. The total load in either case is below the 2000 hour rating of 3000 Kw. Since no credit has been taken for the reduction in pumping requirements following a LOCA and since the diesels 2000 hr rating is not exceeded, the diesel operability will not be affected. A detailed loading study has also verified this fact and testing will be performed to further verify this fact.

2.1.1.7.6 Safety Evaluation

Safety analyses performed on the 177 Fuel Assembly B&W plants have determined that the emergency feedwater systems for a 2772 Mw plant must be capable of delivering 550 gpm (total to both generators). The basis for this criteria is contained in Volume 1; Section 6 - Supplement 3 of B&W's report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant". The analysis submitted by B&W is applicable to TMI-1. Several studies have also been performed by B&W for the 177 FA plants on loss of main feedwater transients. These analysis have demonstrated that 500 gpm or lower auxiliary feedwater flow is adequate following upset transients such as loss of power and the loss of normal feedwater flow. Therefore, the small break LOCA conditions with a 20 minute delay in auxiliary feedwater initiation sets the minimum emergency feedwater capacity requirements. Considering that TMI-1 is only a 2535 Mw, a minimum emergency feedwater capacity requirement of 550 gpm is very conservative.

As discussed in paragraph 2.1.1.7.3 above, the TMI-1 emergency feedwater system is comprised of two 460 gpm capacity electric pumps and one 920 gpm capacity steam driven (turbine) pump. The addition of the motor driven pumps (automatically) to the diesel block loading sequence and the turbine-driven pump start circuit ensures that a single failure will not result in less than the minimum required pump capacity being available under all conditions including loss-of-off site power. That is at least two motor driven or one motor driven and the turbine pump will be available under all single failure conditions.

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The addition of the motor driven EFW pumps auto-start circuits and addition of these motors to the diesel block loading sequence ensures that a single failure will not result in less than the minimum required pump capacity being available under all conditions including loss-of-off site power.

The TMI-1 AFW design provides an emergency feed line with control provisions in line to each steam generator. The design is such that the required quantity of water can be provided to at least one steam generator during all single failure conditions involving a Loss of Coolant Accident or loss of normal feed. Under steam line or feed line break conditions, when both main and auxiliary feedwater is isolated to the affected steam generator, a single failure of the unaffected auxiliary feed line control valve will produce unacceptable results. To counteract this situation several short term design improvements have been implemented. A Back-up instrument air systems have been added, the failure mode of the control valves have been changed, and an additional manual control station has been added. All of these changes provide additional assurance that the TMI-1 control valves will be operable when required or at least will fail in the open position. In the long term, the system will be redesigned to account for the extremely unlikely condition where a control valve sticks closed during a steam or feedline break accident.

As noted above, the failure mode of the feedwater control valves, EF-V30A/B, have been changed from a fail-as-is to a fail open position on loss of instrument air. This failure mode is considered best because it gives priority to reliability of feedwater delivery for decay heat removal. Prevention of overfill is a second priority and a condition which should be prevented but without compromising decay heat removal

Several changes have been made to ensure that the operator can prevent an overfill and overcooling condiiton. These changes consist of addition of the back-up class IE powered manual control stations for EF-V30A and EF-V30B in the control room. These changes were made to back-up the existing automatic level control system and plant instrumentation systems which in themselves are highly reliable. In addition, plant procedures are being modified to provide guidance to the operator in recognizing overcooling incidents and taking prompt corrective action. The operators will be trained in the requirements of these procedures as part of the Operator Accelerated Retraining Program. These changes ensure that control of the emergency feedwater system will be available to the operator in the control room for prevention of steam generator overfilling conditions.

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e) The automatic initiating signals and circuits are designed so that failure will not result in the loss of manual capability to initiate AFW from the control room.

f) Safety-grade indication of auxiliary feedwater flow to each steam generator is being provided in the control room. This design is consistent with the existing system design (i.e., two indicators per line is provided).

g) The Flow instruments are to be powered from Class 1E power systems.

Manual capability to initiate the auxiliary feedwater system from the control room has been retained and is such that a single failure in the manual circuits will not result in the loss of system funciton. In addition provisions for testing of the initiating circuits, although not currently included in the design, will be provided. Control room annunciation for all auto start conditions will also be provided.

2.1.1.7.7 Startup Testing and Inservice Testing/Inspection Requirements

During the initial TMI-1 start-up testing, hot functional testing was performed to:

1. Verify the Integrated Control System (ICS) controls the OTSG to the minimum level set point of 30 inches during HFT heat-up.

2. Verify the ICS controls the emergency feedwater system and OTSG level for the following simulated conditions:

   a. Both main feedwater pumps tripped.

   b. AC hand power to the ICS lost.

   c. All four RC pumps tripped.

   d. All four RC pump & both main F.W. pumps tripped.

3. Verify the auto start capability of the steam driven emergency feedwater pumps.

4. Verify operability of the Emergency Feedwater System to supply feedater when OTSG pressure is 1015 psig.

These tests are documented in Test procedure TP 600/11. Acceptable test results were obtained and therefore no need exists to re-perform the above tests. However prior to re-start of TMI-1 the following test will be conducted:

1. Functional tests shall be performed to verify the emergency feed pumps start on loss of feedwater or loss of four reactor coolant pumps.

2.1.2 Long Term Modifications

2.1.2.1 Post Accident Monitoring

2.1.2.1.1 System Description

Post accident monitoring capability will be provided in compliance with Reg. Guide 1.97, Revision 3. Pending the availability of appropriately qualified instrumentation and equipment, the following modifications can be completed by January 1, 1981. The conceptual design will be provided for NRC review by January 1, 1980.

Containment Pressure - Continuous containment pressure indication will be provided in the control room using a range from -5 psig to three times the design pressure of the containment. The pressure indication will be safety grade and will meet the design and qualification requirements of Reg. Guide 1.97. Redundant indication of pressure will be provided.

Containment Water Level - Continuous containment water level indication shall be provided in the control room. A safety grade wide range indicator from the bottom of containment to a level of 10 feet will be installed in accordance with the requirements of Reg. Guide 1.97. In addition, a narrow range indicator from the bottom to the top of the sump with continuous indication in the control room shall be installed which meets the requirements of Reg. Guide 1.89 and is capable of being periodically tested.

Containment Hydrogen Indication -Safety grade continuous indication of containment hydrogen will be provided in the control room. The range of indication will be 0-10% concentration assuming commercial availability over this range.

High Range Containment Radiation Monitor - Two safety grade containment radiation monitor for photon radiation shall be provided with continuous and recording display in the control room. The range of this monitor shall be 10^7 R/hr and shall detect photon radiation down to 60 Kev. Testability of the radiation monitor will be provided in accordance with Reg. Guide 1.118. To our knowledge, manufacture of appropriately qualified equipment to satisfy these requirements will commence by July, 1980.

High Range Effluent Monitor - One high range effluent monitor shall be installed for each normal noble gas release point. The range of these monitors shall be as follows:

Undiluted Containment Exhaust - 10^5 μCi/cc
Diluted Containment Exhaust - 10^4 μCi/cc
Auxiliary & Fuel Handling Building Exhaust - 10^3 μCi/cc
Condenser Off Gas - 10^2 μCi/cc

The design shall be seismically qualified in accordance with Reg. Guide 1.97 and the power supply shall be non-interruptible. The display shall be continuous and recording in the control room Testability will be provided in accordance with Reg. Guide 1.118.

2.1.2.6 Auxiliary Feedwater System

Auto start of the emergency-feedwater (EFW) System is being implemented in two phases: 1. Control Grade Auto Start - This is a non-safety related initiation as described in paragraph 2.1.1.7 and it is a short-term approach, 2. Safety Grade Auto Start - This will be a long-term modification where the initiation will meet the requirements for Class 1E system and the system is functionally described below.

1. The safety grade EFW auto start when implemented will automatically initiate the system on presence of the following conditions with or without the availability of the off-site power:

   - Loss of both normal feedwater pumps, or
   - Loss of all four reactor coolant pumps, or
   - Low differential pressure between the normal feedwater with main steam lines at each steam generator,

   The system initiation on low steam generator level will eventually be added. This will be done after the necessary analysis and engineering has been completed to insure that this signal will give a satisfactory actuation and will not interact with other plant functions. Loss of normal feedwater pumps is detected by differential pressure switches across each pump (two switches per pump, i.e., one switch per train). Since they are installed in a non-seismic building, these switches are not safety grade instruments. However, they will be tied into the EFW initiating circuits (Train A & B) through buffer devices. The other components of the initiating circuits will be safety grade items.

2. All cables associated with the initiating logic will be qualified for Class 1E application and the initiations will be designed to meet single failure criteria. All circuits will meet the regulatory criteria for separation of Class 1E circuits.

3. The initiating logic will include hardware for the following purposes:

   - Latching mechanism to seal-in the actuation
   - Manual Reset Capability
   - Testability of the initiating circuit

4. Indication will be provided in the control room to identify the source of the initiation.

5. Annunciation will be provided in the control room to alarm:

   - Auto start of the EFW system. This will be a common alarm for both the trains.

THREE MILE ISLAND UNIT NO. 1

TABLE 2.1-1
List of Isolation Signal Override Capability

	Penetration No.	Isolation Signal: Reactor Trip	High Radiation	4 psig Building	30 psig Building	500 psig (SFAS)	Line Break	Loss of Off-Site Power
Containment Air Sample	108	N/A	N/A	C	N/A	C	N/A	N/A
R.B. Sump	353	C	IB	C	N/A	N/A	N/A	N/A
RCDT	330,331	C	IB	C	N/A	N/A	N/A	N/A
RCS Sample	328	C	IB	C	N/A	N/A	N/A	N/A
R.B. Purge	336,423	C	NO	NO	N/A	N/A	N/A	N/A
RCS Letdown	309	A	IB	C	N/A	N/A	N/A	N/A
Demin Water	307	C	N/A	C	N/A	N/A	N/A	N/A
OTSG Sample	213, 214	C	IB	C	N/A	N/A	N/A	N/A
NSCCW	346, 347	N/A	N/A	N/A	NO	N/A	NO	N/A
ICCW	302, 333, 334	N/A	N/A	N/A	NO	N/A	NO	N/A
R.B. Air Coolers	421, 422	N/A	N/A	C	N/A	C	N/A	N/A
R.C. Pump Seal Return	329	N/A	N/A	N/A	NO	N/A	N/A	NO

Legend
- C - Common Signal Override; initiating isolation condition may still exist.
- I - Individual isolation signal override capability; procedures governing override to be developed.
- IB - Individual isolation signal bypass capability
- A - Automatic isolation signal override.
- NO - No override or 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.

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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 B & W Proposed	Valve Actuation Signal Source Modified	Notes
108	Containment Air Sample	RM	CM-V1	Ball	1	Air	Open	Closed	Closed	Yes	1,10	1,10	1,2,6,10	
			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	1,3,10	8,9,10	3,7,8,9,10	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	1,2,10	1,5,10	
309	Letdown Line to Purification Demineralizers	MU	MU-V2A	Globe	2-1/2	EMO	Open	Closed	Closed	Yes	1,10	1,2,10	1,4,5,6,10	
			MU-V2B	Globe	2-1/2	EMO	Open	Closed	Closed	Yes	1,10	1,2,10	1,4,5,6,10	
			MU-V3	Gate	2-1/2	Air	Open	Closed	Closed	Yes	1,10	1,2,10	1,6,10	
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1	Globe	3/8	EMO	Closed	Closed	Closed	Yes	1,10	1,2,10	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	7,10	3,7,8,9,10,11	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	1,2,10	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	1,2,10	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	1,3,10	8,9,10	3,7,8,9,10	B&W does not discuss 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	1,3,10	1,2,9,10	3,7,8,9,10	See Note (1) below.

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THREE MILE ISLAND UNIT NO. 1

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: B&W Proposed	Valve Actuation Signal Source: Modified	Notes
336	Reactor Building Outlet Purge Line	AH	AH-V1A	Butterfly	48	Air	Closed	Closed	Closed	Yes	1,10	1,7,10	1,4,5,10	
			AH-V1B	Butterfly	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	8,9,10	7,8,9,10,11	See Note (1) below
347	Reactor Coolant Pump Motor Cooling Water Return	NS	N. /4	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	8,9,10	7,8,9,10	See Note (1) below
			NS-V35	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	8,9,10	7,8,9,10,11	
353	Reactor Building Sump Drain	WDL	WDL-V534	Gate	6	Air	Closed	Closed	Closed	Yes	1,10	1,2,10	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	8,9	1,2,10;	
422	Reactor Building Normal Air Coolers Return Line	RB	RB-V7	Gate	8	Air	Open	Closed	Open	Yes	1,10	8,9	1,2,10;	Add auto initiation of Emerg. R.B. cooling on 4 psig R.B. and 1600 psig R.C. pressure isolation signals.
423	Reactor Building Inlet Purge Line	AH	AH-V1C	Butterfly	48	EMO	Closed	Closed	Closed	Yes	1,4,10	1,2,10	1,4,5,10	
			AH-V1D	Butterfly	48	Air	Closed	Closed	Closed	Yes				
-	R.C.P. Seal Return Cooling, etc.		NS-V32	Gate	6	EMO	Open				1,2,10	-	8,10,11	This is not a containment isolation valve. This valve is isolated to prevent one NS pump runout.

Valve Actuation Signal Source

1) 4 psig reactor building pressure isolation
2) 1600 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
11) Loss of off-site power isolation for NS pump runout protection; this is not for containment isolation function.

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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THREE MILE ISLAND UNIT NO. 1
TABLE 2.1-3
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. near the sampling line downstream of the containment isolation valve and upstream of connection for Turb. Plant sampling	Area Gamma Detectors (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.	Area Gamma Detector (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.	Area Gamma Detector (New)
330	Reactor Coolant Drain Tank	WDG	WDG-V3	Locate the monitor on the outside of the tank.	Area Gamma Detector (Existing)
and	Vent		-V4		
331	Reactor Coolant Drain Tank Pump Discharge	WDL	WDL-V303 -V304		
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 an area radiation monitor in the R.B. Sump mounted inside a seismically supported pipe.	Sump Area 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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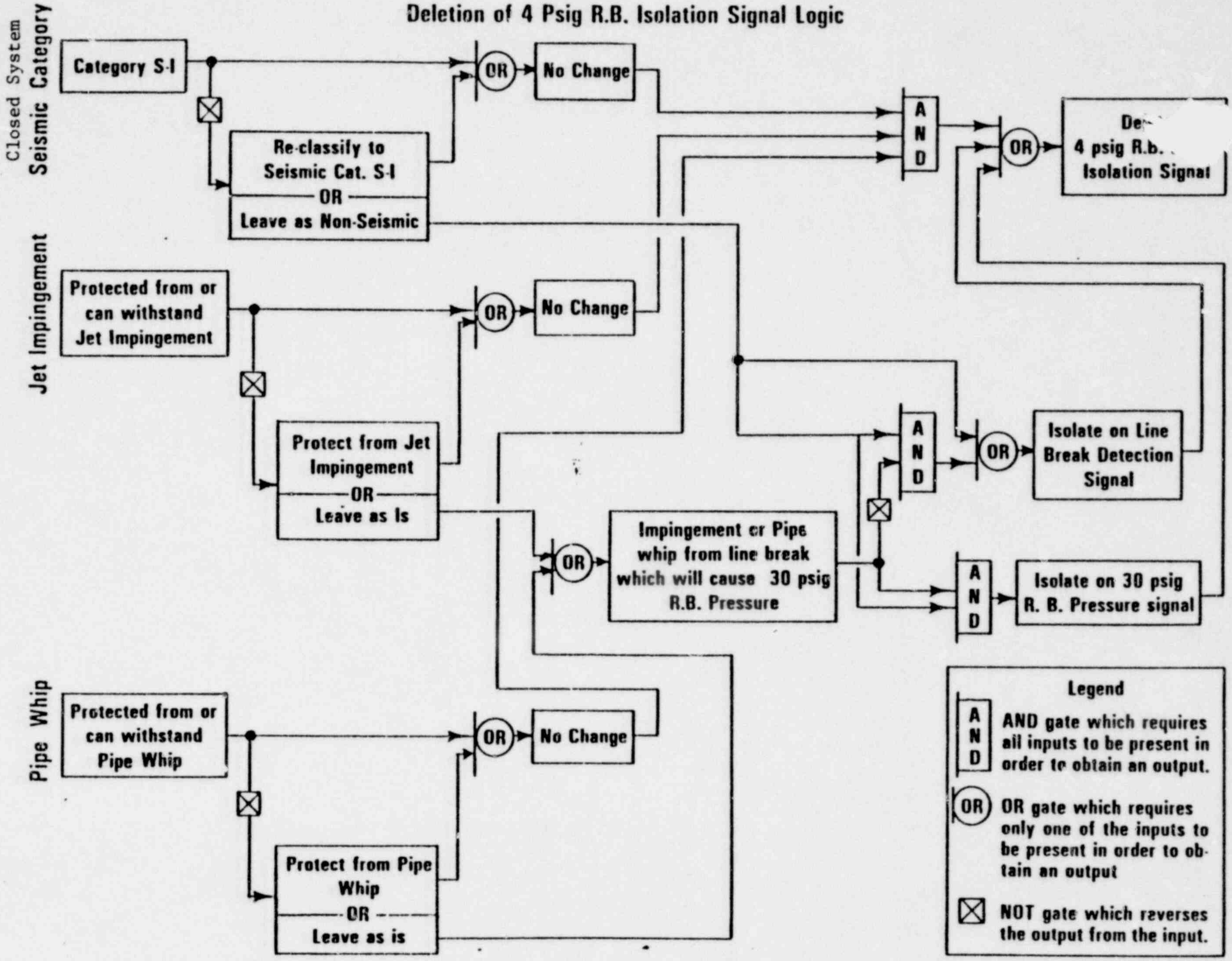
Figure 2.1-5

Deletion of 4 Psig R.B. Isolation Signal Logic



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5.0 THREE MILE ISLAND NUCLEAR STATION ORGANIZATION

5.1 GENERAL

Following the TMI-2 accident, Metropolitan Edison Company recognized through its own and other investigations of the accident that major organizational changes were desirable for more effective management control. These changes indicate Met-Ed's commitment to operational safety and provide significant improvement in the control of operational activities, and the technical and management resources directing and supporting facility operations.

The first step taken was to combine the technical and management resources of Met-Ed and GPU Service Corporation Generation Divisions into a single organizational entity identified as the TMI Generation Group.

The TMI Generation Group was formed on July 30, 1979, to strengthen the overall management and provide greatly increased technical resources for the restart of TMI Unit 1 and the recovery of TMI Unit 2. The Group is headed by R. C. Arnold. To effect this new organization, Mr. Arnold was elected to the position of Senior Vice President of Met-Ed, and he continues to serve as a Vice President of GPU Service Corporation. In this position, Mr. Arnold reports to Herman M. Dieckamp, President of GPU and GPUSC, and acting president of Met-Ed. This reporting structure provides a direct link from the Chief Operating Officer of these three companies to the activities at TMI. A primary objective of the TMI Generation Group is to insure that NRC Regulations, Technical Specifications and established procedures are adhered to.

This group was formed to take advantage of the wealth of nuclear experience represented by management and technical staff from within the GPU Service Corporation and Metropolitan Edison Company. This reallignment more than tripled the number of professionals that have TMI as their primary responsibility.

There are senior management personnel with an average technical experience well over 20 years reporting to the head of the TMI Generation Group in the areas of:

. TMI-1 Operations

. TMI-2 Recovery

. Environment, Health and Safety

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. Reliability Engineering

. Radiological Controls

. Engineering and Design

Various steps have been taken in this reorganization to strengthen key functions in the operation and support for Unit 1. Examples of this are:

. The line management responsibilities for TMI Units 1 and 2 are completely separated.

. Each TMI unit is to the maximum extent feasible, to have direct control of the resources necessary for effective and safe conduct of plant activities.

. The head of the TMI-1 Operations, Mr. J.G. Herbein, Vice President-Nuclear Operations is serving full time at TMI ahd his responsibilities and functions are described in Section 5.2.1.

. The organization formed under Mr. Herbein's direction specifically gives the Unit 1 Superintendent only the responsibility for operations and maintenance and relieves him of the direction of administration, training, engineering, radiation protection and chemistry functions.

. The radiological control function for Unit 1 has been elevated so that it reports directly to the Vice President-Nuclear Operations.

. The GPU Service Company and Metropolitan Edison Company Quality Assurance and Control organizations were merged, and Operating Quality Assurance for TMI is their major function.

The following sections describe the pertinent details of the TMI Generation Group.

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5.4 <u>Quality Assurance Program and Procedural Control the TMI-1 Restart</u>

5.4.1 The TMI-2 accident has required major readjustments in the organization and management of the TMI Nuclear Station. The organizational structure of the General Public Utilities Service Corporation (GPUSC) and the Quality Assurance Program for controlling the operational activities, at TMI Nuclear Staton are contained in the Operational Quality Assurance Plan for TMI Units 1 and 2. This Plan establishes the organization and the management controls and Quality Assurance Program necessary to assure that the operational phase activities at the Nuclear Station are performed and controlled in a manner that will not endanger the health and safety of the public or the employees or contractors of Metropolitan Edison Company (Met-Ed) or the GPUSC. Inherent in the oeprations of the Nuclear station are those day-to-day activities which are direclty associated with keeping Unit 1 on the line and Unit 2 in a safe configuration while the decontamination and/or restoration activities are being performed until such time as Unit 2 becomes operational. These activities are performed by the Operations personnel and those supporting activities such as radiation protection, surveillance testing, environmental monitoring, refueling, inservice inspection, modification, etc. which are required to assure continued operation in a safe and economical manner. Inherent also in the operations of the Nuclear Station are those activities associated with the verification of the completeness and adequacy of the work perfromed and the provision of independent safety review and operational advice.

This Operational Quality Assurance Plan is applicable to (1) the operation of TMI Unit 1; (2) the maintenance of those TMI Unit 2 items which have not been affected by the accident; (3) the maintenance of those TMI Unit 2 items which upon satisfactory completion of decontamination and recovery are turned over to and accepted by Nuclear Operations; and, finally, (4) the operation of TMI Unit 2.

The overall responsibility for the establishment and implementation of the Operational Quality Assurance Program and for assuring proper and complete interfacing of the organizations having responsibilities for performing the work rests with the TMI Generation Group. This group is managed by the Senior VP Met-Ed/VP GPUSC. The TMI Generation Group consists of one Vice President and five Directors, as illustrated on Figure 5.4-1. The responsibilities of these departments relative to operations will be detailed separately in the Operational Quality Assurance Plan.

5.4.2 <u>Quality Assurance Department</u>

The TMI Generation Group Quality Assurance Department under the direction of the Manager of Quality Assurance reports to the

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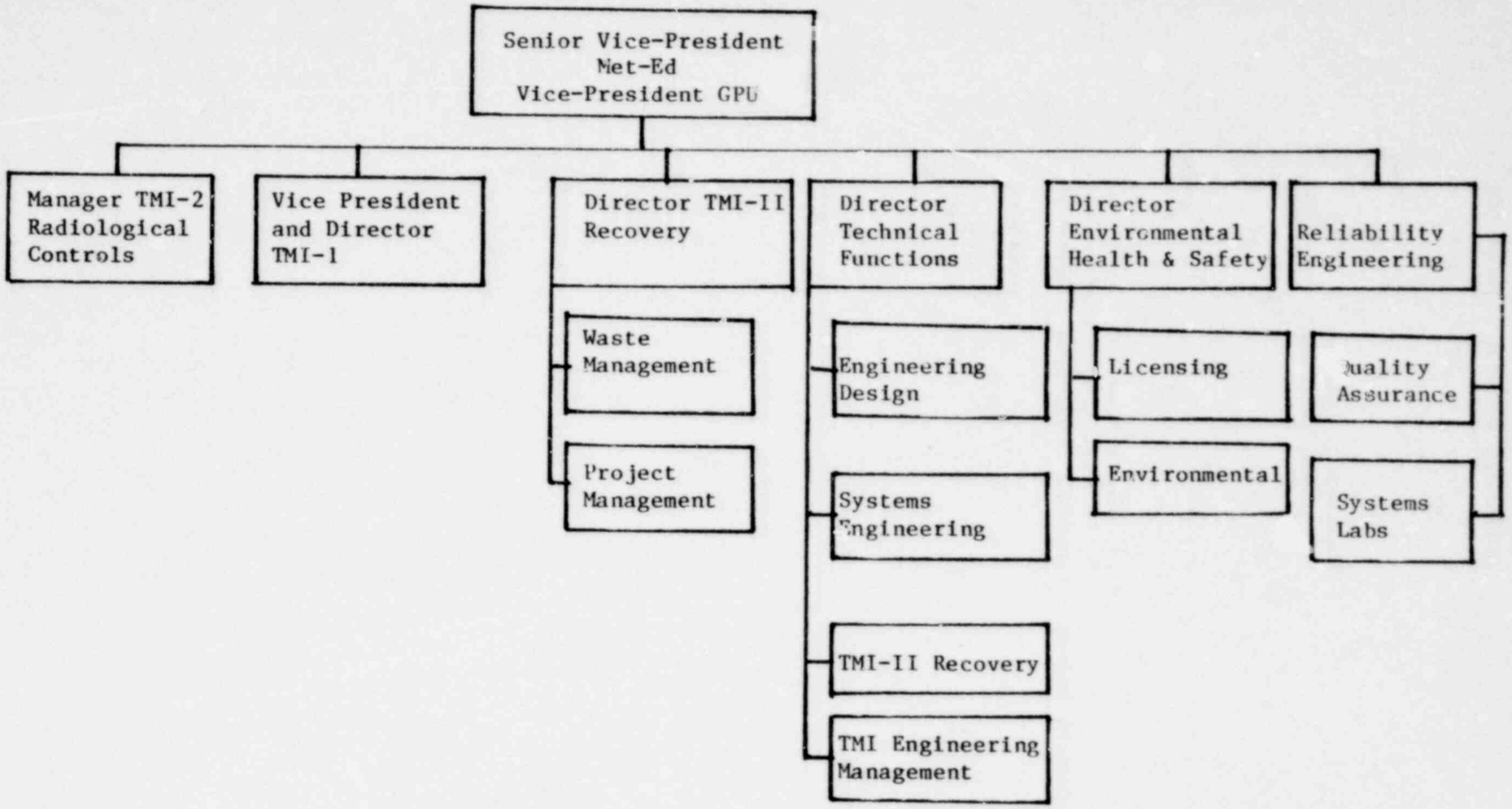

Figure 5.3-1
Station Support Organization

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TMI GENERATION GROUP

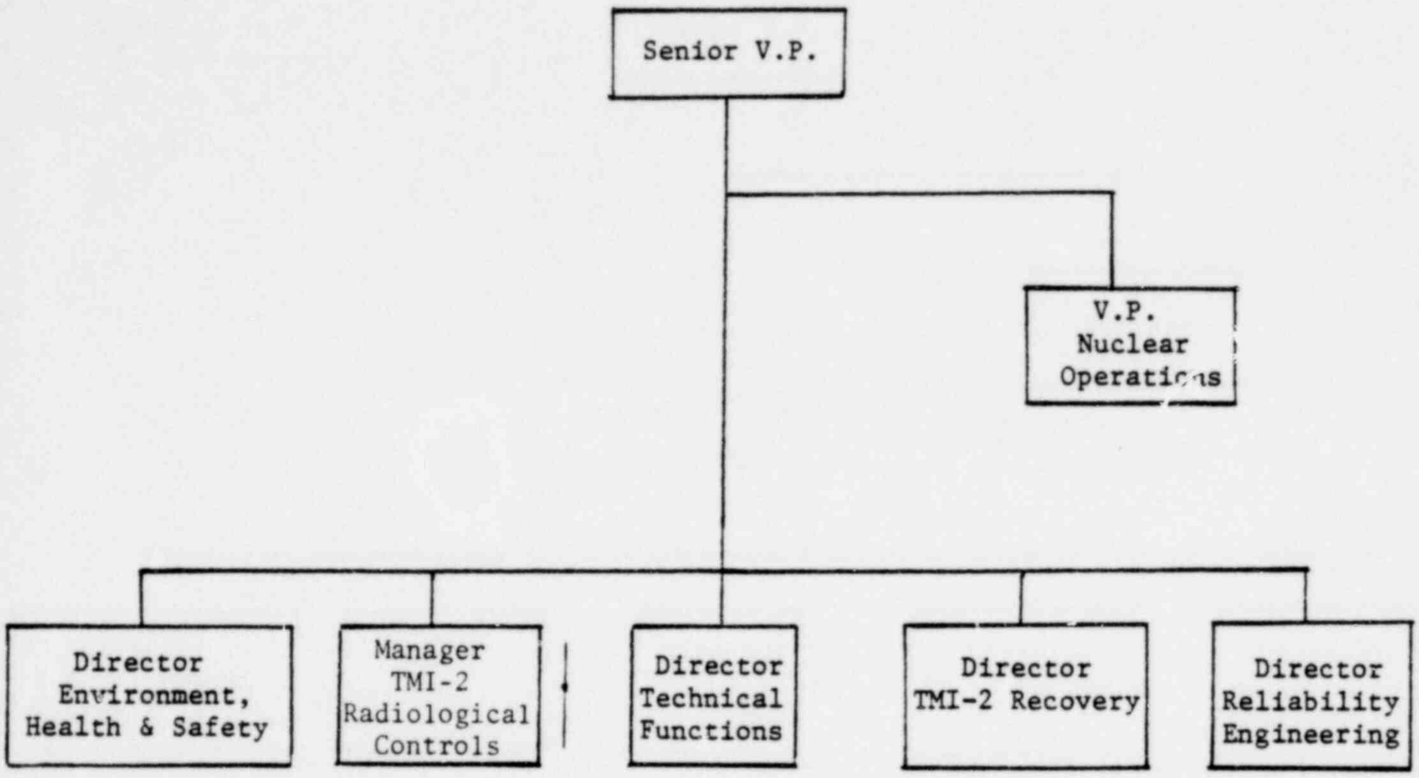

FIGURE 5.4-1

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8.0 SAFETY ANALYSIS

8.1 INTRODUCTION

Design 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 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 would improve plant performance. Some of these modifications were being evaluated prior to the TMI-2 accident on March 28, 1979.

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. The reactor protection high pressure trip setpoint has been changed 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. A complete loss of feedwater flow will initiate a reactor trip.

3. A turbine trip will initiate a reactor trip.

4. The emergency feedwater system will be modified before restart to allow:

   a. control grade automatic initiation of the steam and master driver EFW pumps upon loss of all 4 reactor coolant pumps or a loss of both main feedwater pumps.

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

   c. alternate manual control for the EFW control valves.

5. A long-term modification will provide safety grade actuation of the EFW pumps on the following signals:

   a. low steam generator level. This is a long-term item since further engineering is required. Plant safety therefore will be discussed with and without this feature.

b. negative feed to steam differential pressure.

c. loss of all four reactor coolant pumps.

d. loss of both main FW pumps.

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. Post accident instrument and valve operator availability will be improved by the addition of heat shrink tubing.

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

3. The reactor building spray system will be modified to delete sodium thiosulfate. Sodium hydroxide will be retained. This change will provide equal drawdown of the BWST and NaOH tanks for a large spectrum of single failures.

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%)

ii. RCS pressure not 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 has been 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 psig. 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

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. Mitigation

   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/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; as demonstrated by the natural circulation tests and events discussed in Reference 2.

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. The 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.

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8.3.7 Loss of Electric Power (FSAR Section 14.1.2.8)

1. Description

Separation of the unit from the transmission network can result in the trip of the turbine and reactor. A more 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).

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 of 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. Control of the steam driven emergency feedwater by the EFW level control system.

ii. Steam relief through the atmospheric dump and main steam relief valves either by the ICS or in accordance with Emergency procedure 1202-2 and 2a.

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 Abnormal Transient Operating Guideline (ATOG) 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 steam lines 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.

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

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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 change the results of this analysis. Results are still valid and acceptable.

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8.3.11 Fuel Handling Accident (FSAR Section 14.2.2.1 and References 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 initiating 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.

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. Releases to the environment result both from releases via the secondary system and from leakage from containment.

2. Acceptance Criteria

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

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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, Section 15.1.8, S3-21.49, S2-21.43, Reference 2, Q.3 of Supplement 1, Part 2)

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, increased pressurizer level and pressure, and reactor trip on high RCS pressure. The TMI-2 analysis assumes a complete loss of feedwater due to a break upstream of the first feedwater line check valves. No analysis of loss of feedwater due to pump trip or valve closures were analyzed. The loss of feedwater flow due to the postulated break is analyzed as an immediate loss of flow, which results in a bounding analysis for loss of feedwater events. 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).

Reference 2 and Question 3, Supplement 1, Part 2 provide results for a loss of normal Feedwater event. Table 8-2 compares the analysis assumptions to the plant design.

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.

iii. Pressurizer does not become water solid during a loss of Feedwater transient.

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 since reactor trip will be initiated sooner. The RCS heatup is thus reduced. 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.

   As demonstrated by Table 8-2 and Q3 Supp 1, Part 2, TMI-1 meets the acceptance criteria for a loss of Feedwater transient and the analysis bounds the TMI-1 plant design.

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 0.005 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. Local fuel cladding oxidation (metal water reaction) shall not exceed 0.17 times the total cladding thickness, or .05 the overall cladding mass.

   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. Mitigation

   i. Inventory will be maintained by the high pressure injection system.

   ii. Emergency Feedwater flow within 20 minutes of very small break LOCA's allows depressurization of the RCS and allows sufficient inventory addition by the HPI system to maintain core cooling.

4. Conclusion

Pursuant to NRC regulations (10CFR50.46) and 10CFR50 Appendix K) B&W performed generic LOCA 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 $ft.^2$.

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Key assumptions for the small break LOCA analyses versus the TMI-1 plant design 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****
ESFAS HPI (psig)	1600	1500
Operator Action	Yes-cross-connect	none***
HPI Distribution	70% to Core within 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. Results of Reference 2 demonstrate that EFW is not required before 20 minutes.

*** 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.

**** For worst case LOCA in which offsite power is lost, EFW is initiated by the control grade loss of feedwater signal.

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 Reference 21 (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 assumptions for TMI-1 except for the distribution of HPI flow as discussed below. The analysis in Reference 2 also established that EFW flow is not required less than 20 minutes before any steam line break accident.

In Reference 2, credit is taken for operator action to initiate HPI or EFW. No mention is made as to whether whether operator action includes the 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 on the broken HPI leg would be essentially zero and therefore the HPI loss out the break could be high resulting in inadequate injection to the core.

The criterion established by B&W for the small break analysis requires that 70% of the total flow for one HPI pump be injected into the broken legs of the reactor coolant system. This criteria applies to a 2772 MW thermal 177 fuel-assembly plant. For TMI-1 with a licensed core power of 2535 MWt, the 70% - 30% criterion can be relaxed in direct proportion to the power reduction. This is justified based on the fact that the decay heat load following a small break LOCA is proportional to power and therefore cooling requirements will be directly proportional to the power at which the plant has operated. Therefore, for TMI-1, the acceptable flow split can be relaxed to 64% - 36%. 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 the broken leg.

The venturis have been sized to limit flow in each leg to 137.5 gpm when only one high pressure injection pump is operating and Reactor Coolant System is at atmospheric pressure. The venturi design ensures that for the worst case HPI line break condition, the 64%/36% flow split can be achieved when Reactor System Pressure is less than 1500 psig. At RCS pressure conditions greater than 1500 psig, a flow split beyond the 64%/36% acceptance criteria will occur. B&W has reviewed this situation and judged the cavitating venturi performance is acceptable. This conclusion is based on the fact that under HPI line break conditions, the Reactor Coolant System will not expend significant time above 1500 psig and that during the time the RCS is above 1500 psig the cavitating venturi ensures that there is significant flow of high pressure injection into the RC system. B&W also notes that a much larger small break than a HPI line break sets the generic flow split criteria and therefore for a HPI line break the flow split criteria can be relaxed.

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In addition to the benefits discussed above, the venturis provide two added benefits. First, they balance flow of the injection legs under all other small break conditions such that TMI-1 flow split will be within the bounds of the generic analysis (i.e., 70%/30% flow split). Secondly, the cavitating can be relaxed.

8.3.16 Large Break Loss of Coolant Accidents (Reference FSAR Section 14.2.2.3)

1. Description

Break sizes in the reactor coolant system (RCS) greater than 0.5 ft.^{2} are classified as large break loss of coolant accidents (LOCA's). These breaks involve rapid depressurization of the RCS and are accompanied by rapid increases in containment pressure. Offsite doses are calculated from the design basis radioactivity release to containment, and the design basis containment leak rate.

2. Acceptance Criteria

i. Peak fuel clad temperature does not exceed 2200°F.

ii. The core is maintained in a coolable geometry.

iii. Local fuel cladding oxidation (metal water reaction) shall not exceed 0.17 times the total cladding thickness of .05 times the total cladding mass.

iv. Offsite doses are within the limits specified by 10 CFR 100.

3. Mitigation

i. Core flood tank actuation at 600 psig to establish water inventory.

ii. Low pressure injection system flow below 200 psig to establish core cooling for the remainder of the accident.

iii. Building spray addition to put iodine in solution with the containment water volume thus preventing release to the environment.

iv. Containment leak tightness to limit radioactivity releases.

v. Switchover of the decay heat removal system suction source to the containment building sump on low-low BWST level.

4. Conclusion

The calculated offsite dose resulting from the design basis LOCA will increase as a result of the deletion of sodium thiosulfate from the building spray system. Doses will still

be within the limits of 10 CFR 100. Dose calculations performed for TMI-2 (see TMI-2 FSAR, Section 15 and Reference 5) demonstrate that design basis LOCA doses are within the limits of 10 CFR 100. The TMI-2 dose calculations were performed taking no credit for sodium thiosulfate. Since Unit 2 has a slightly large thermal power level and allowable containment leak rate, then Unit 2 dose calculations conservatively bound the worst case LOCA dose for TMI-1.

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.

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 assurances 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 improved 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 FSAR analysis results.

   The combined effect of the PORV and RPS setpoint changes are to decrease the probability of PORV operation. The 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 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. This event has additional safety margins beyond the TMI-2 analysis since both turbine and feedwater trips result in a

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reactor trip. This earlier reactor trip will result in a smaller heatup of the primary system.

5. 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.

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

7. 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.

8. Dose calculation performed for TMI-2 demonstrate that the requirements of 10CFR100 are met even after sodium thiosulfate is deleted.

9. The transition to natural circulation following a complete loss of feedwater will be demonstrated by a startup test. Reference 2 documents natural circulation tests and natural circulation events at B&W designed reactors. These tests and events demonstrate that natural circulation is a reliable and effective means of core cooling.

10. An analysis of the station blackout will be performed as part of the B&W Owners Group ATOG program to determine what specific actions would be required to bring the plant to a safe shutdown condition.

11. A PORV setpoint of 2450 psig does not result in unacceptable interactions between the PORV and the pressurizer safety valves, whose setpoint is 2500 psig.

1438 049

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.

21. Michelson, C. "Decay Heat Removal During a Very Small Break LOCA for a B&W 205-Fuel Assembly PWR". January, 1978.

22. Winks, Robert W., Analysis of the Requirements for Loss-of-Electrical-Load Transient at TMI-1. Babcock & Wilcox. October 28, 1975.

23. Winks, Robert W., Final Report on Phase 1 (Turbine Trip Test) and Evaluation for Phase 2 Testing for the Loss of Electrical Load Capability at TMI-1. Babcock and Wilcox, May 20, 1975.

24. McFadden, J. F., R. D. Hentzen, J. F. Harrison, N. S. Burrell. RETRAN-A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems: Volume 4: Applications, pp 72 thru 83 EPRIC-CCM-5. Palo Alto, California, December, 1978.

25. Harrison, J. F., N. G. Trikouros, A. A. Irani. "RETRAN Simulation of the Three Mile Island Unit 1 Turbine Trip Test." American Nuclear Society Transactions, Vol. 32, pp 455-459.

26. Denver, D. J., J. F. Harrison, N. G. Trikouros. "RETRAN Natural Circulation Analyses During the Three Mile Island Unit 2 Accident." General Public Utilities, Parsippany, N.J. November 1, 1979.

1438 051

TABLE 8-2

KEY INPUT PARAMETERS FOR LOFW ANALYSIS

Parameter	B&W Generic Study (Ref. 2)	B&W Revised (Q3, Supp. 1, Part 2)	TMI-1
Reactor Coolant System			
Reactor power, MWt	2772 (100% FP)	2772	2619 (102%)
Decay heat	1.0 ANS 5.1	1.2 ANS 5.1	
System pressure, psig	2185 (core outlet)		2192
RCS coolant flow, % design flow	108.0		106.5
Reactor temperature, F			
Inlet	557		555
Average	582		579
Outlet	607		603
Configuration	Lowered Loop	Lowered Loop	Lowered Loop
Pressurizer			
Operating level, in.	220		220
Volume, ft^3			
Steam	708	708	1529 (total)
Water	800	800	
Operating pressure, psig	2155		2157
Code safety valves			
Number	2		2
Flow area, $in.^2$	3.34		5.09
Valve capacity, lb/s-ft^2			
Liquid			
Steam			8,130,000 lbm/hr. @ 2600 psig
Setpoints, psig			
Open	2500		2500
Close	2460		2475
PORVs			
Number	1		1
Flow area, $in.^2$	1.05		1.05
Valve capacity, lb/s-ft^2			
Liquid			
Steam			112,000 lbm/hr @ 2300 psig
Setpoints, psig[(a)]			
Open	2450		2450
Close	2400		2400

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Am. 6

TABLE 8-2 (Cont.)

Parameter	B&W Generic Study (Ref. 2)	B&W Revised (Q3, Supp. 1, Part 2)	TMI-1	
Reactivity Feedback				
Doppler coefficient, $\Delta k/k°F$	-1.56×10^{-5}		-1.47×10^{-5} (BOC)	
Moderator coefficient, $\Delta k/k°F$	-1.05×10^{-4}	0	-0.77×10^{-4} (BOC)	
Reactor Protection				
High RC pressure trip setpoint, psig[(a)]	2300	2300	2300	
High RC pressure trip delay, s	0.4	1.0	~0.3	
Auxiliary Feedwater				
Flow, gpm	1000	370	740	1160[b]
Pressure, psig	N/A	10[illegible]0	1030	700[b]
Time delay, sec.	40		35	35
Steam Generators				
Inventory, lbm per OTSG		18,400		18,400

(a) As measured at hot leg tap.

(b) Assumes failure of the turbine-driven EFW pump (100% capacity) and open failure of the motor-driven recirculation line.

APPENDIX 8A

RETRAN/GPU-01

ANALYSIS OF TMI-1

1438 054

Am. 6

APPENDIX 8A

Introduction

As discussed in response to Question 39, Supplement 2, Part 2, GPU has embarked on an analysis of the TMI-1 plant response using RETRAN/GPU-01. This code is a modification of the RETRAN (see Reference 24) one-dimensional thermal hydraulic analysis code developed for the Electric Power Research Institute.

Model Description

Two basic models have been developed for TMI-1. The first is the one-loop model shown on Figure 8A-1. This nodalization scheme has been developed to provide relatively detailed analysis results for cases when non-symetric secondary effects are not important. The second model is shown on Figure 8A-2. This model provides a representation of the RCS and secondary system as a two loop model. A nodalization of this type allows the modeling of non-symetric effects in either the RCS or secondary systems.

The control systems for both models are the same, with the exception that secondary system controls are modeled separately for each secondary system. RETRAN/GPU-01 models all of the reactor protective system and SFAS trips and initiating signals. The model also initates SFAS on loss of all four reactor coolant pumps and 20% voiding at the pump inlet. Secondary system pressure control is explicitly modeled to separate the effects of the turbine bypass, atmospheric exhaust, and safety valves. Feedwater and emergency feedwater are terminated upon initiation of a feedwater latch signal (600 psig in the steam generator). The emergency feedwater system is modeled to separate the motor driven and steam driven pumps, with diesel generator load sequencing and mechanical flow coast up accounted for in each system, separately. Emergency feedwater is initiated by the RETRAN trips when an initiating condition is sensed. OTSG level is controlled at a low or high level, depending upon the availability of the RCS pumps.

The HPI/makeup system is also modeled explicitly. The normal makeup pump controls pressurizer level at the normal set point of 220 inches via the makeup control valve (MU-V17). Upon SFAS initiation, normal makeup is isolated and HPI is initiated, with flow varying with RCS system backpr -sure.

Proposed Analyses

Table 8A-1 lists the analyses which we intend to undertake using RETRAN/GPU-01. There are four basic accidents/transients of interest: loss of feedwater, loss of offsite power, station blackout, and feedwater line break. We may also perform steam line break analyses if a suitable model can be developed. None of the above analyses are intended for use in support of the TMI-1 restart effort. However, other analyses may be performed as docket analyses in the future.

1438 055

The scope of analyses is intended to provide a more detailed understanding of the overall plant response to a broad spectrum of likely design basis transients.

Analysis Results

To date, several analyses have been completed. Results are discussed briefly below.

Figure 8A-5 provides results of the base case loss of feedwater transient. No equipment failures are assumed and no operator action is taken. The reactor trips as a result of the loss of feedwater trip. The analysis was terminated after 10 minutes. Hot and cold leg temperatures have converged, and pressurizer level is being restored by the normal makeup pump. Pressurizer spray is actuated very briefly, while the RCS pressure never approaches the PORV setpoint.

Emergency feedwater initiates as a result of the loss of feedwater pumps. Both the motor driven and the steam driven pumps achieve full flow, and flow is automatically controlled to maintain OTSG level at 30 inches in the startup range. Secondary system pressure is limited by the safety valves, atmospheric exhaust valves, and turbine bypass in the first 50 seconds, and by the turbine bypass valves thereafter.

Figure 8A-6 provides results of a loss of feedwater event with an EFW system which has flow limiting devices. Two motor driven pumps run, but flow is limited to 740 gpm. The analysis was carried out until the plant achieved stable conditions: RCS temperature and pressure were constant, pressurizer level was restored, the secondary side heat sink was provided via the turbine bypass system. As can be seen from the "PSAT meter graph", the subcooled condition of the RCS was never challanged.

Figure 8A-7 provides results of the base case station blackout. After one hour, RCS conditions have still not converged, i.e., hot leg temperature and RCS pressure are still decreasing and pressurizer level is about to go off scale (although the pressurizer would not be empty). As in the other analyses, no operator actions were assumed, and no additional failures were imposed. The analysis does provide some indication of the time frame for operator actions to restore onsite AC power.

TABLE 8A-1

PROPOSED RETRAN/GPU-01 ANALYSES OF TMI-1

I. Loss of Offsite Power (LOOP)

Case 1:	Base	Show plant response to LOOP and transition to natural circulation.
Case 2:	1940 gpm EFW and 1% decay heat (max cooldown case)	Examine overcooling potential with 200% EFW and minimum decay heat.
Case 3:	Stuck open OTSG safety valves (17% of design flow)	Examine plant performance for first 10 minutes of secondary side depressurization using one OTSG model.
Case 4:	EFW flow limitation	Examine long-term plant response with EFW flow limitation and LOOP from 100% power.
Case 5:	1% decay heat with EFW flow limitation	Examine effect of flow limiters in EFW system on max cooldown case.
Case 6:	2 OTSG model with stuck open relief valves	Evaluate effect of non-symetric cooldown on secondary side.
Case 7:	2 OTSG model & no EFW to 1 OTSG	Evaluate non-symetric loss of heat sink.
Case 8:	EFW in superheat region	Evaluate effect on transition to natural circulation of putting EFW into superheat region rather than downcomer. Shows more realistic plant response.

II. Station Blackout

Case 1:	Base	Look at long-term plant response to event, including voiding in the RCS LOOP.
Case 2:	1 gpm pressurizer leakage and min. decay heat	Effect on plant response due to cool-down of pressurizer steam space.
Case 3:	Raise atmospheric exhaust setpoint	Evaluate efficacy of blackout procedure, which calls for this action.

1438 057

TABLE 8A-1 (Cont.)

III. Loss of Feedwater

Case 1: 460 gpm EFW Flow	Examine plant response to operation of only one motor-driven EFW pump.
Case 2: EFW flow limitation	Look at plant transition to stable shutdown with EFW flow limited.
Case 3: 1940 gpm	Contrast performance against EFW flow limitation.
Case 4: Failure of EFW level control	Evaluate effect of continuous EFW flow causing overfill of OTSG.
Case 5: Trip from low power and min. decay heat	Look at plant response below turbine trip threshold.
Case 6: Case 2 or TMI-2 in 2 OTSG model	Compare 1 & 2 LOOP results.
Case 7: Partial LOFW	Compare response to complete LOFW & evaluate ICS response.
Case 8: No EFW	Look at time available before HPI must be initiated.
Case 9: Turbine stop valve fails open	Limiting overcooling case following a LOFW.
Case 10: Manual actuation of HPI	Look at effect on pressurizer level of operator initiating a second HPI pump immediately after reactor trip.

IV. Feed Line Break Accident

Case 1: EFW initiation on feed/steam ΔP	Demonstrate plant can tolerate the design basis feedline break accident.
Case 2: EFW initiation on low OTSG level	Evaluate timeliness of this signal for feedline break.
Case 3: Partial break which causes gradual loss of OTSG level and no feed/steam ΔP signal	Determine if operator response is as can be reasonably expected.

1438 058

TABLE 8A-1 (Cont.)

V. Steam Line Break Accident

Case 1:	Benchmark of TMI-2 docket analysis	Establish benchmarked code for use on TMI-1.
Case 2:	TMI-1 design basis analysis	Demonstrate means required to establish long-term safe shutdown.
Case 3:	Break from startup condition with flooded nozzles	Determine if startup from low OTSG level is necessitated by safety concerns.

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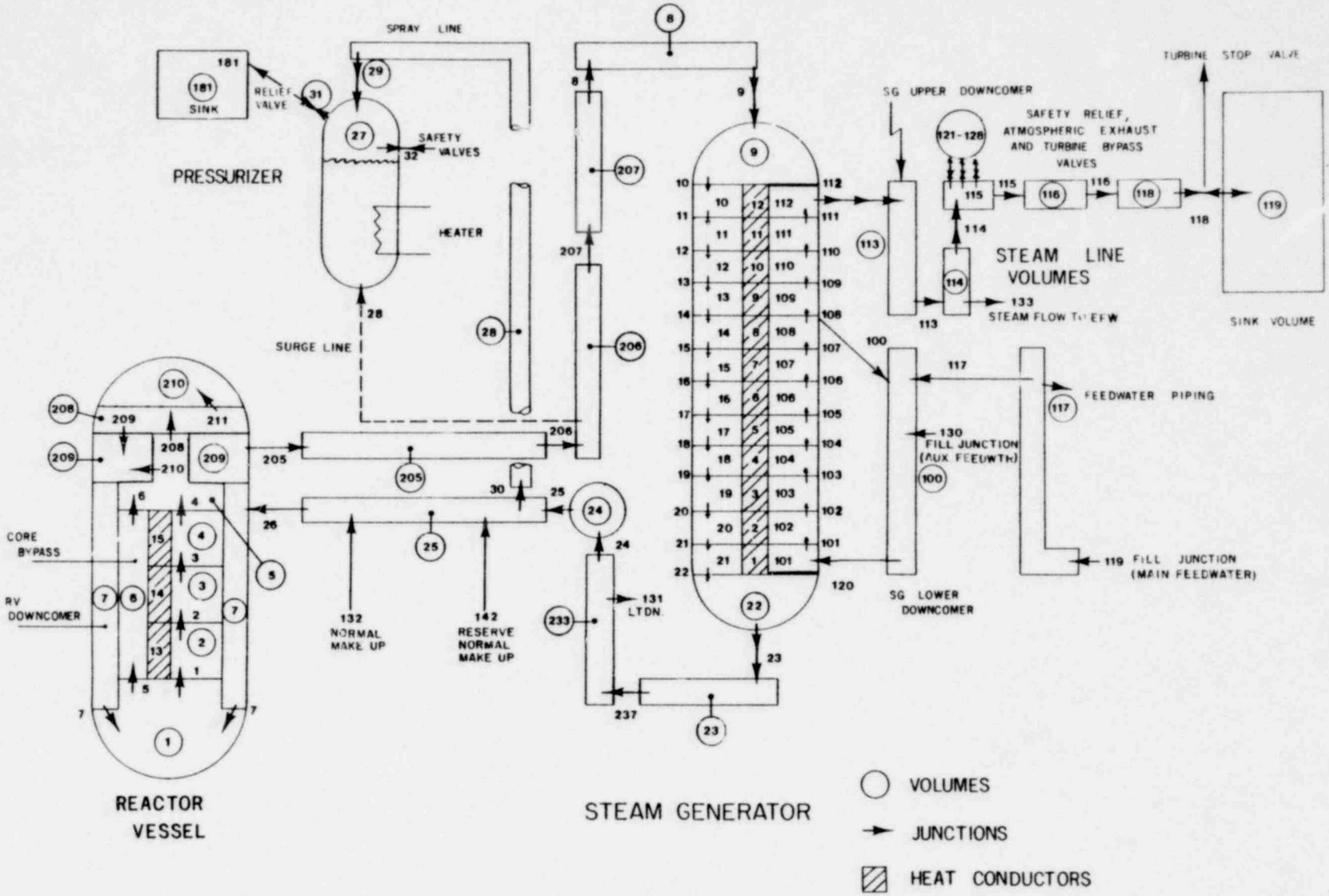

FIGURE 8A-1

THREE MILE ISLAND UNIT 1

RETRAN ONE LOOP MODEL

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Am. 6

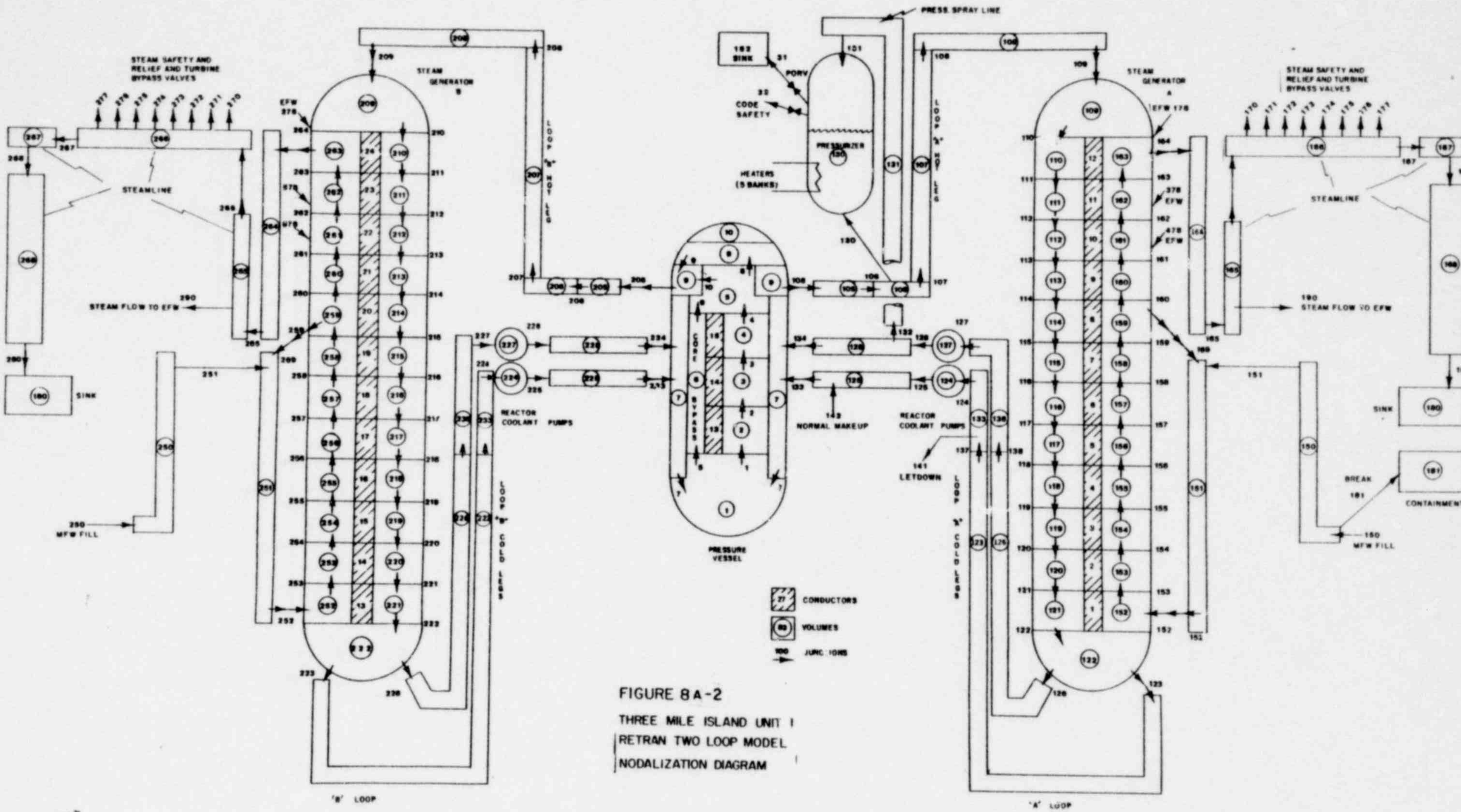

FIGURE 8A-2

THREE MILE ISLAND UNIT I

RETRAN TWO LOOP MODEL

NODALIZATION DIAGRAM

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Am. 6

ANALYSIS ASSUMPTIONS

```
*****************************************************
************** LOFW BASE CASE  **********************
*****************************************************
***                MAKEUP: AVAILABLE                *
***              LETDOWN: ISOLATED AT T=0           *
***              EFW MECH: AVAILABLE                *
***             EFW STEAM: AVAILABLE                *
***         EFW CAPACITY: 1940 GPM                  *
***       TURBINE BYPASS: AVAILABLE SETPOINT=1025PSIG*
***    ATMOSPHERIC DUMP: AVAILAPLE                  *
***       SMALL SAFETIES: 3% RESET POINT            *
***              BANK 1 : 3% RESET                  *
***            BANKS 2&3: 3% RESET                  *
***           PRESS HTRS: 5 BANKS                   *
***          PRESS SPRAY: AVAILABLE 2220/2170 PSIA  *
***          RC PUMPS   : AVAILABLE                 *
***          RPS TRIP   : TURBINE                   *
***     RPS DEFEATED    : NONE                      *
***         SFAS TRIP   : 1500 PSIG RCS             *
***                     : 4 PSIG CONTAIN PRESS DEFEAT*
***   DIESEL GENERATORS : 2                         *
***        OFFSITE POWER:   AVAILABLE               *
***                 PORV: 2450/2400 PSIG            *
***        PZR SAFETIES : 2500/2475 PSIG            *
*****************************************************
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Figure 8A-5

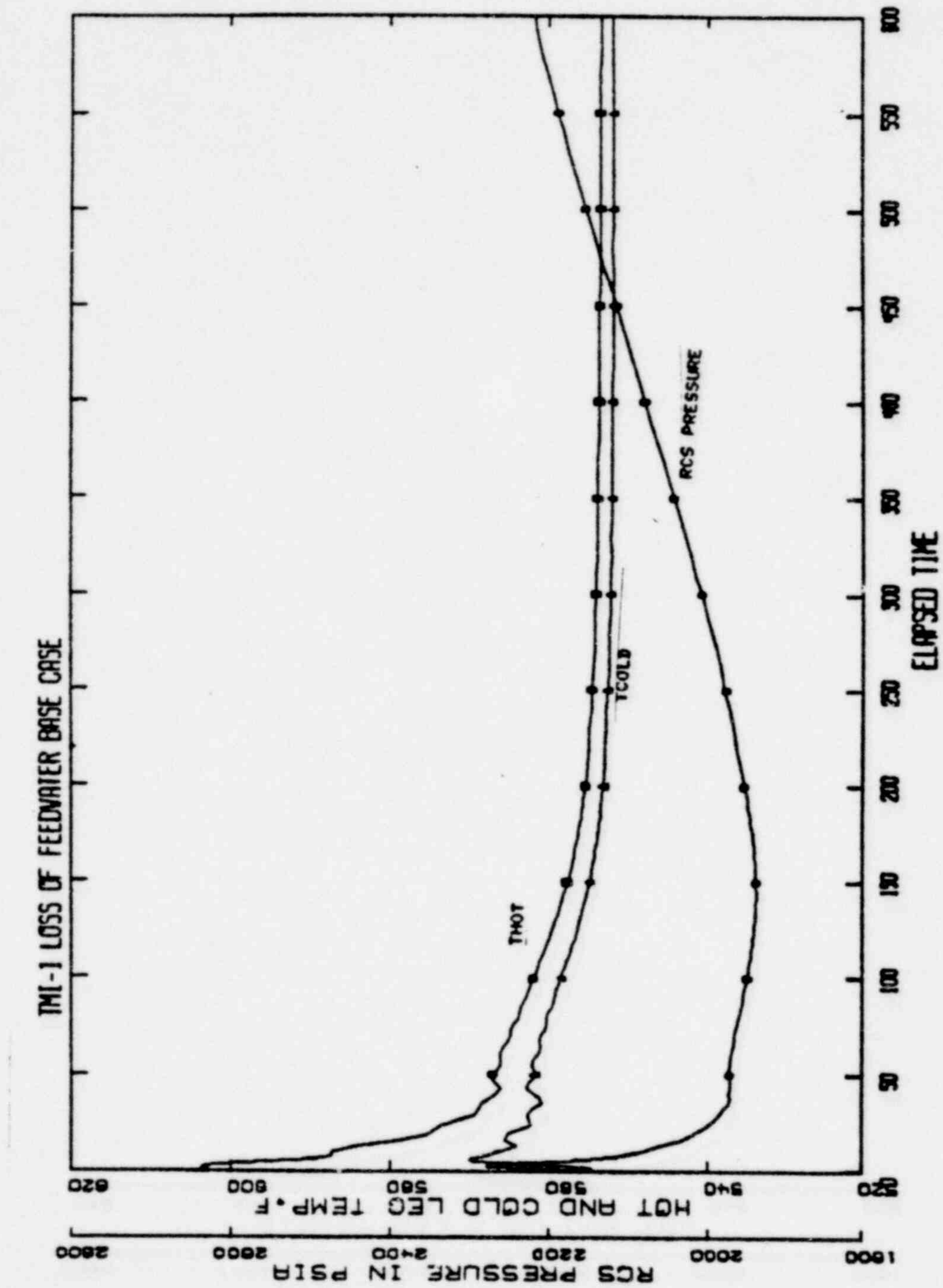

Figure 8A-5
Sheet 1

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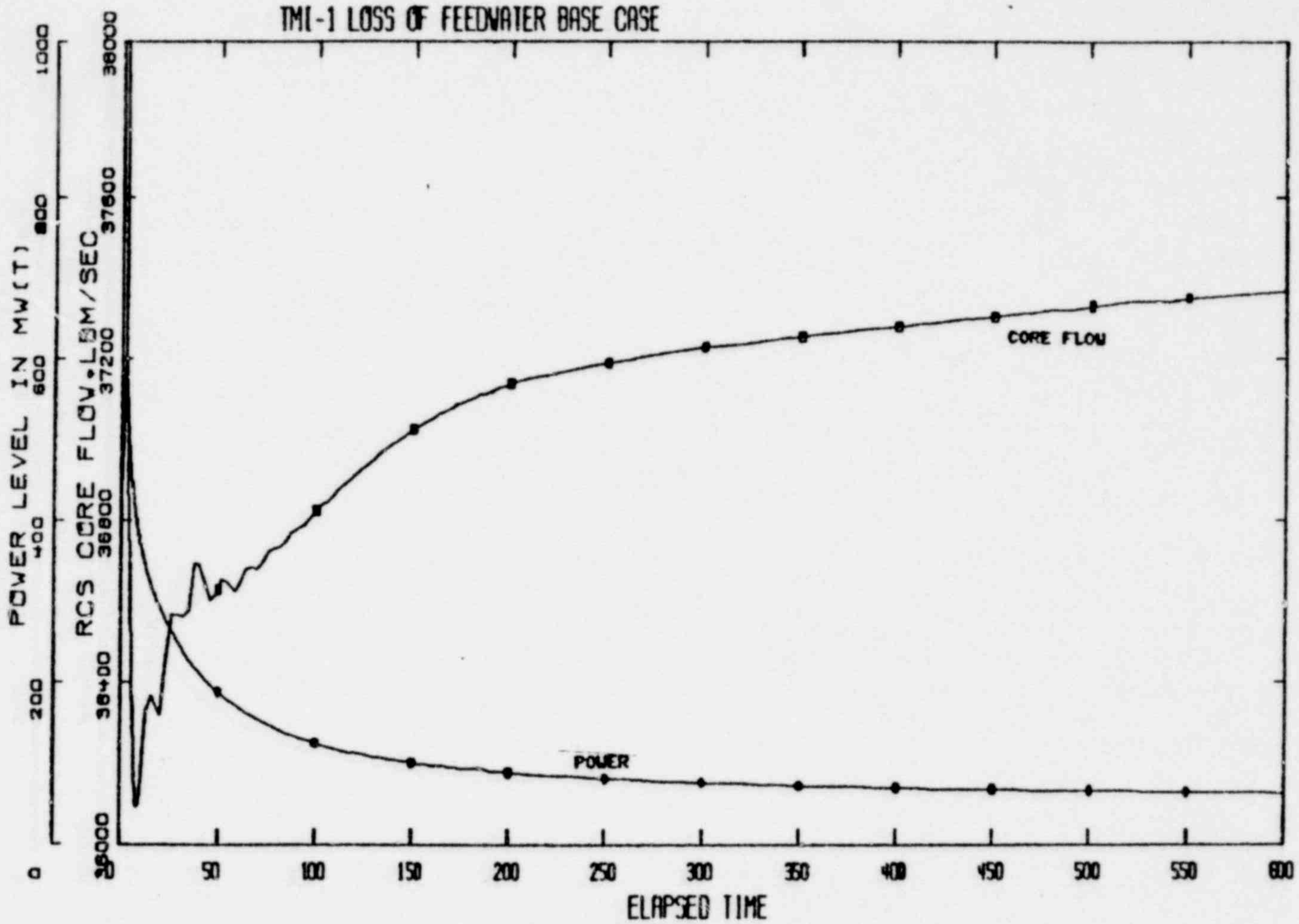

Figure 8A-5
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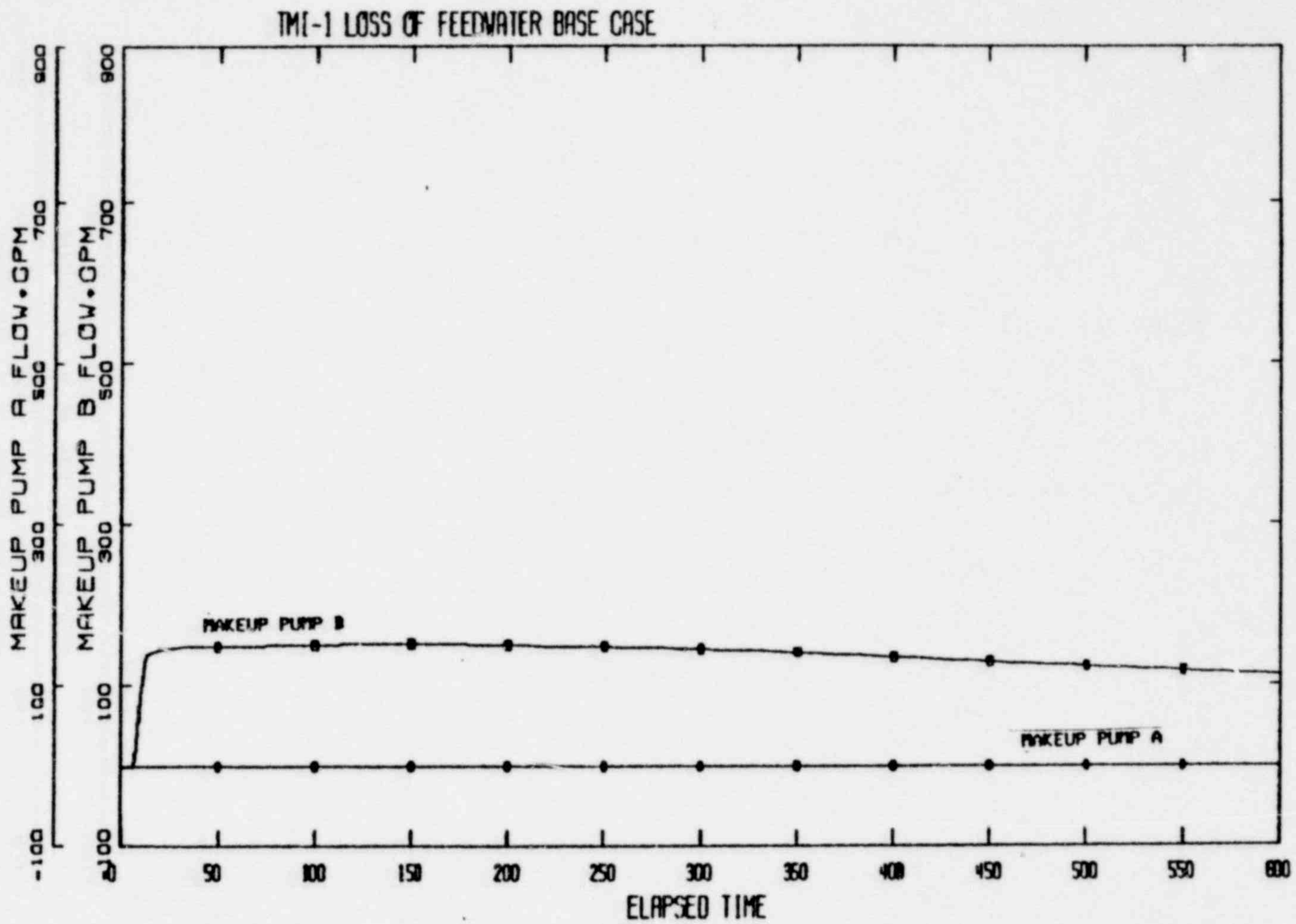

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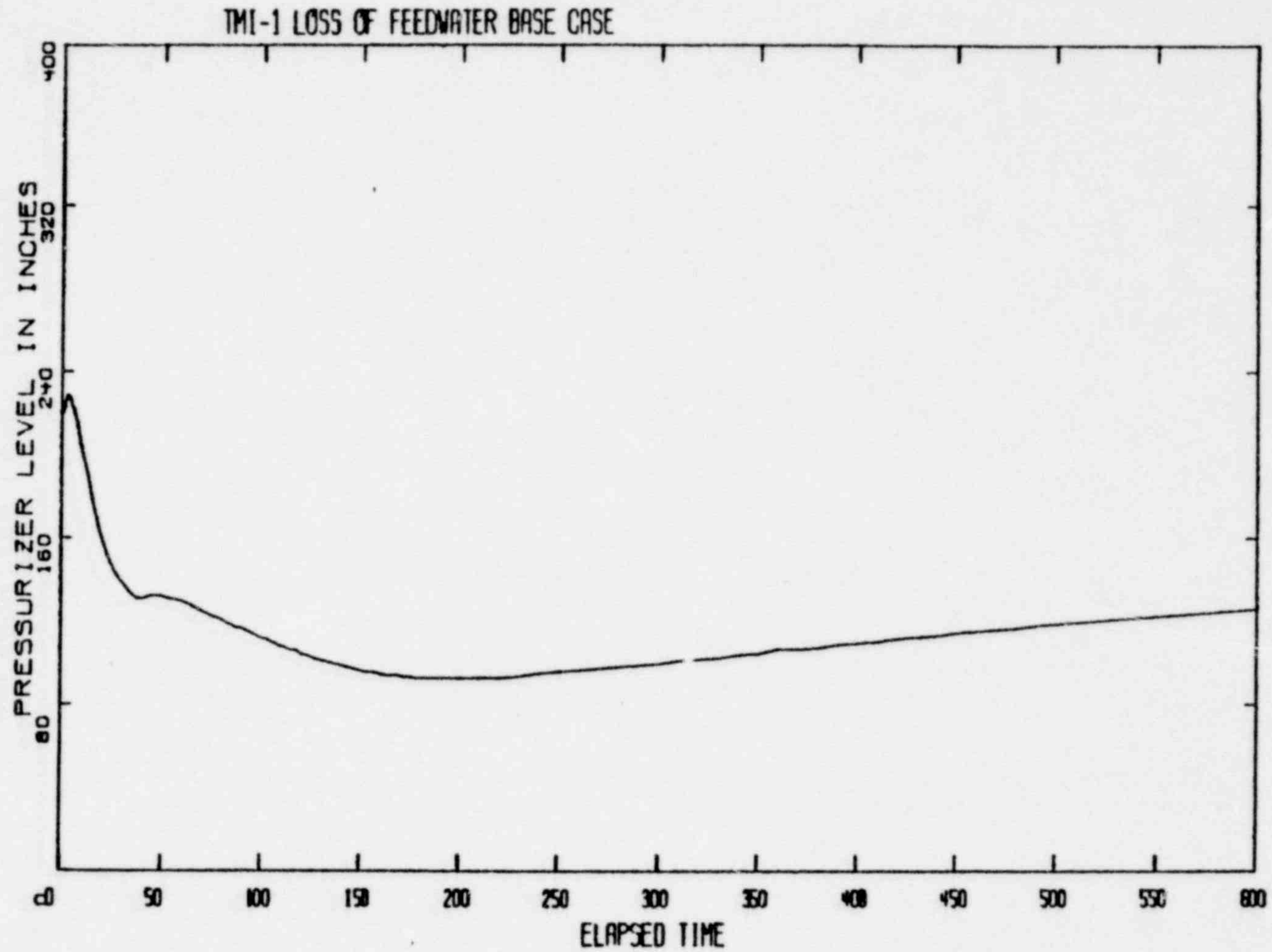

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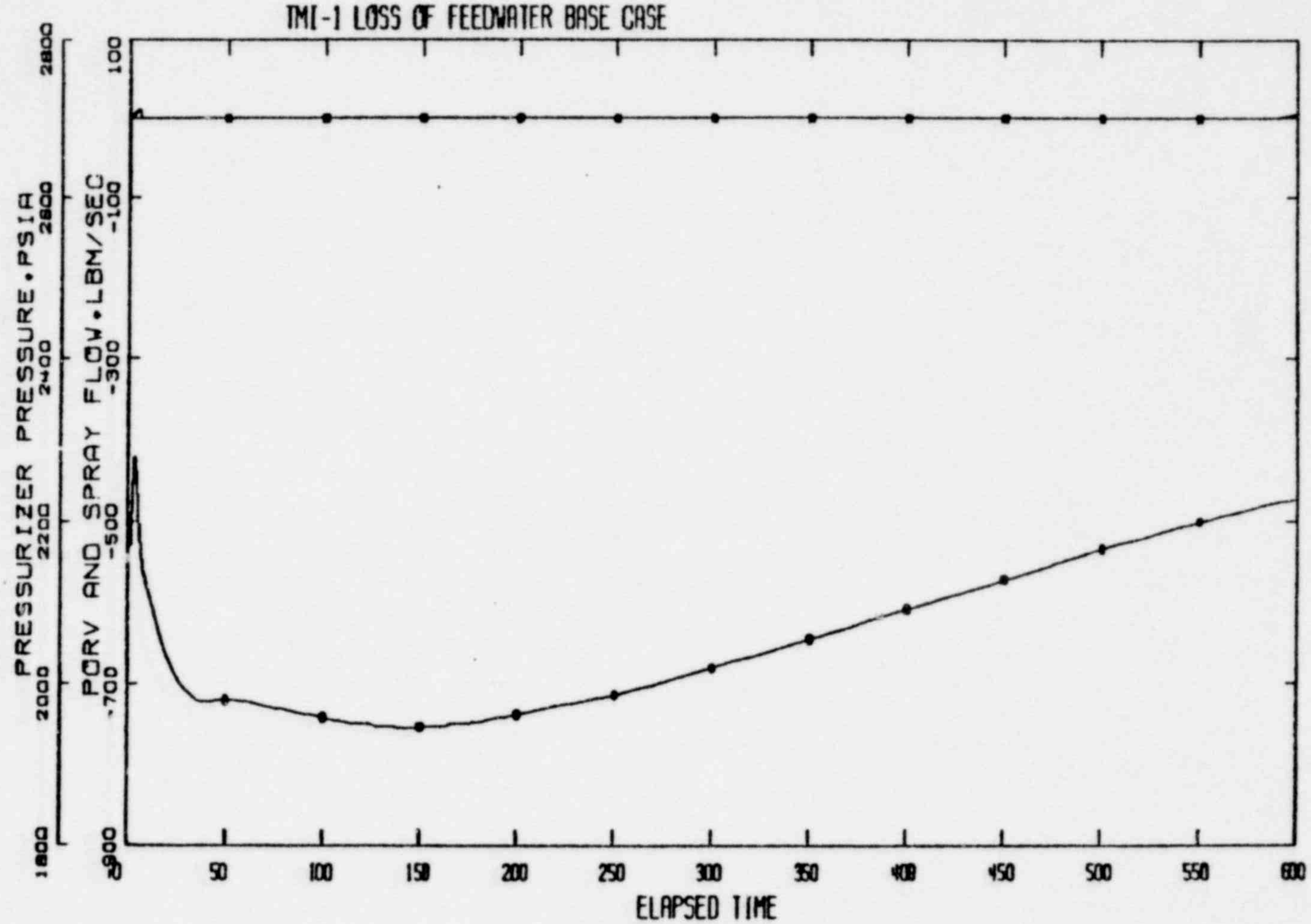

Figure 8A-5
Sheet 5

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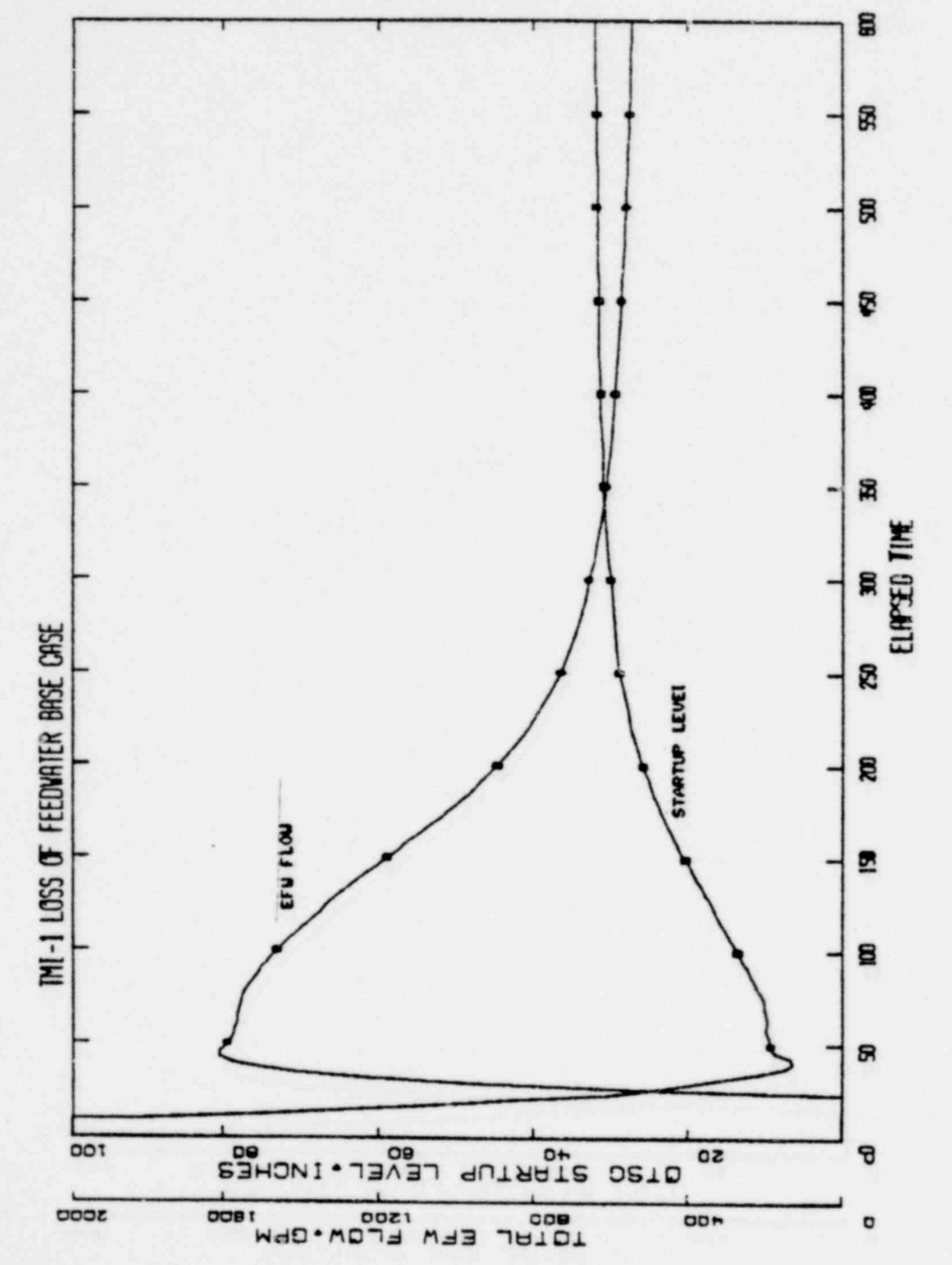
Figure 8A-5
Sheet 6



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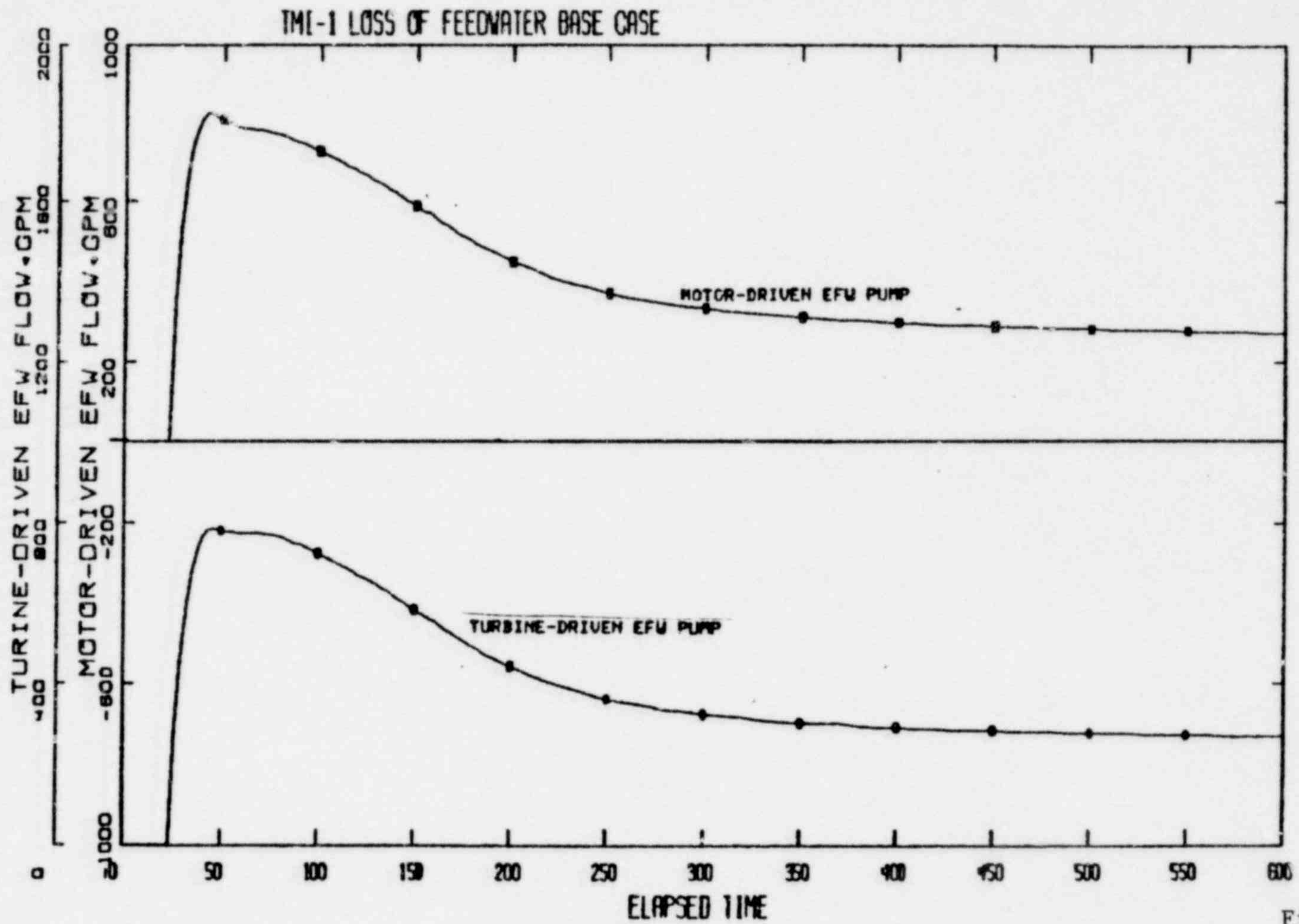

Figure 8A-5
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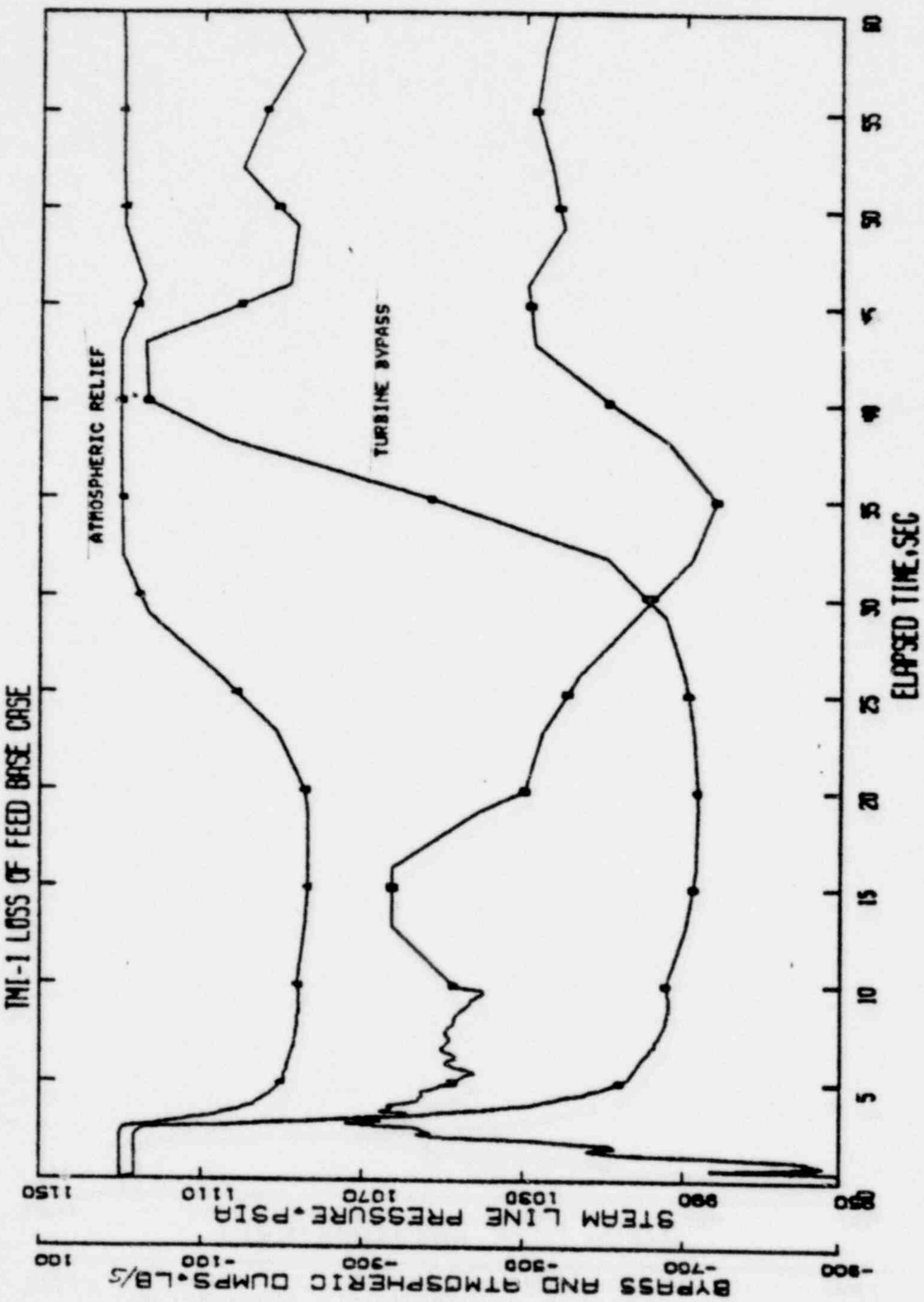

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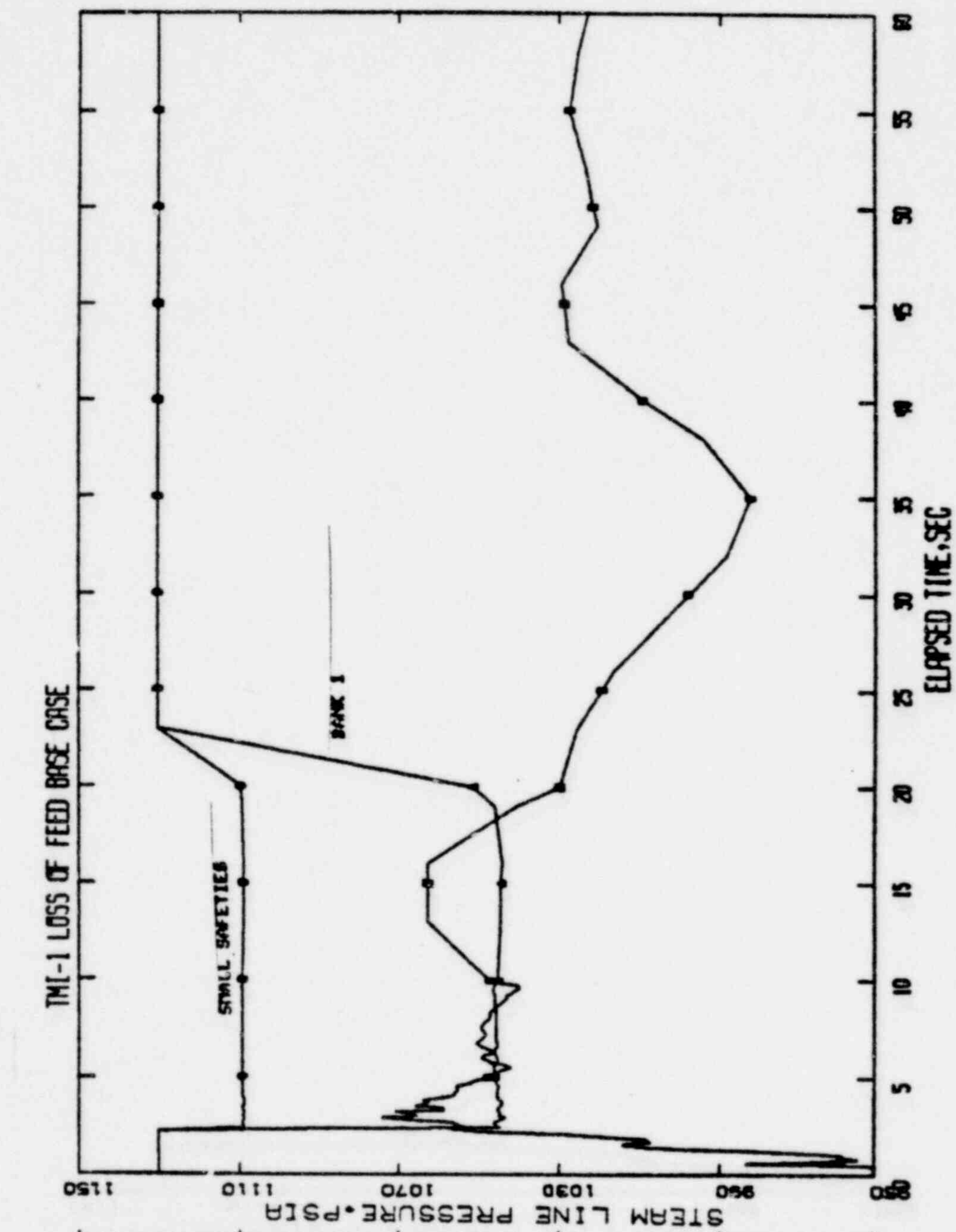

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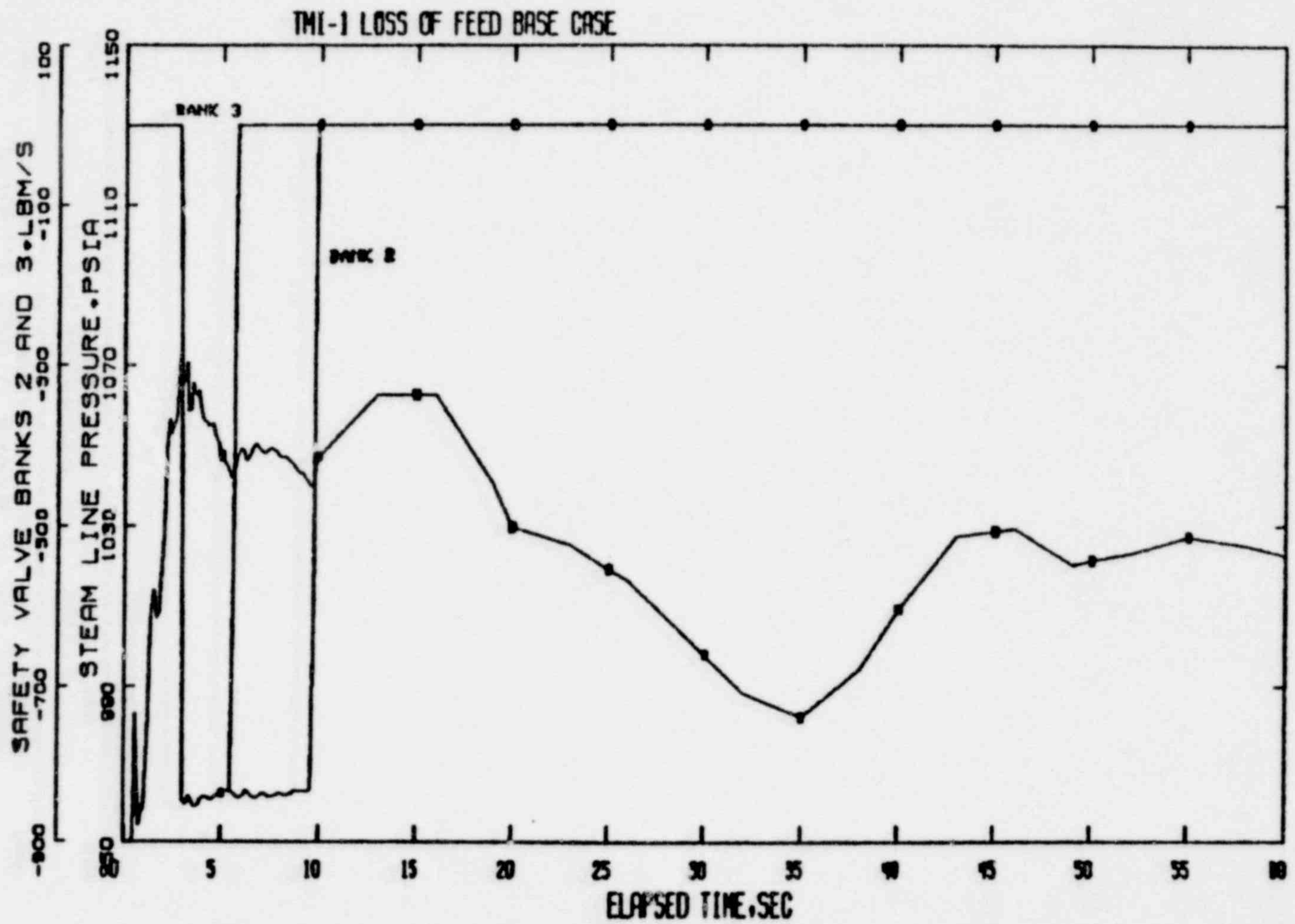

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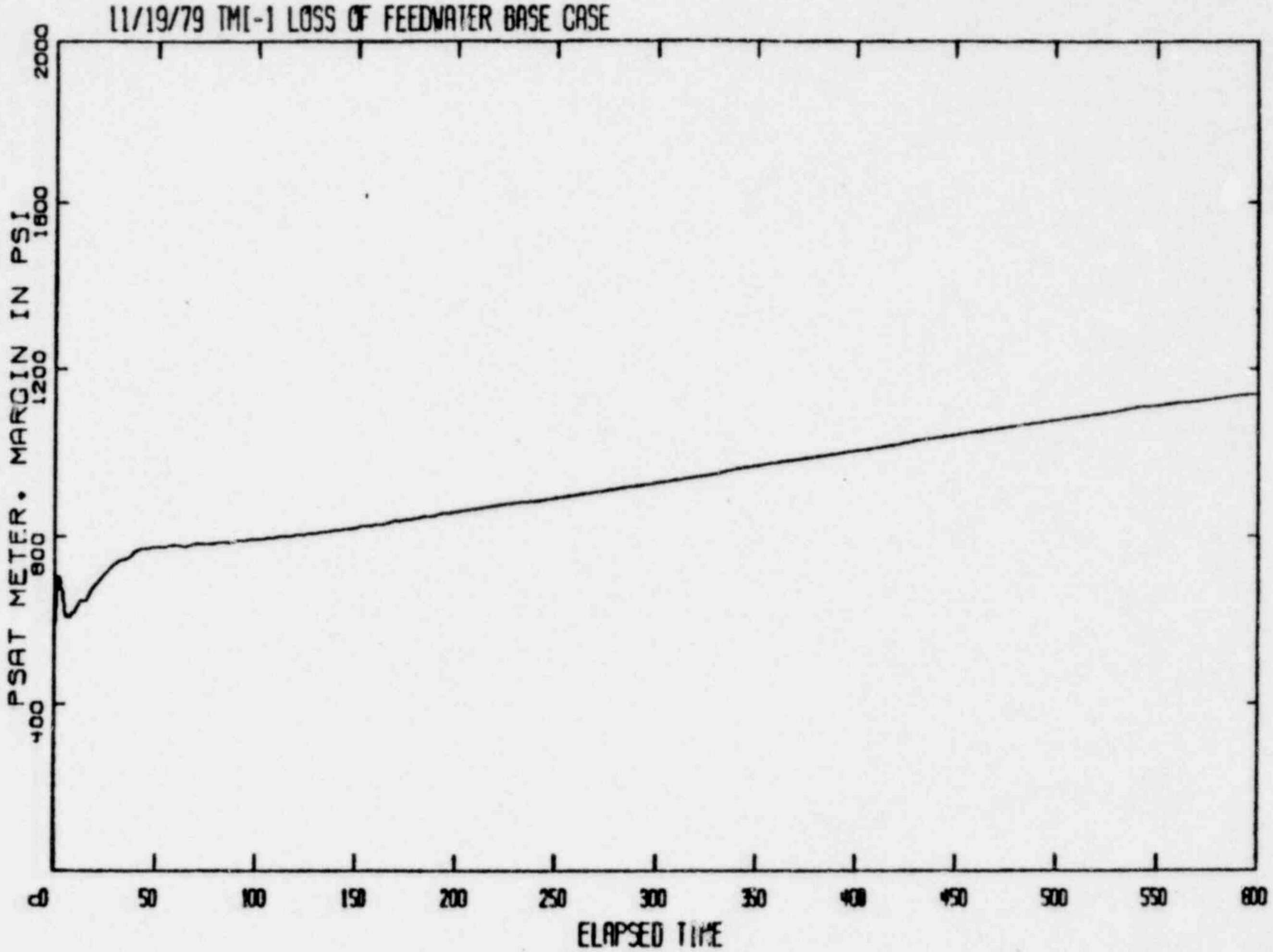

Figure 8A-5
Sheet 11

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LOFW ANALYSIS ASSUMPTIONS

CASE 2(EFW LIMIT 100%)

MAKEUP:	AVAILABLE
LETDOWN:	ISOLATED AT T=0
EFW MECH:	AVAILABLE
EFW STEAM:	AVAILABLE
EFW CAPACITY:	740 GPM 100%-FLOW LIMITER
TURBINE BYPASS:	AVAILABLE SETPOINT=1025PSIG
ATMOSPHERIC DUMP:	AVAILABLE
SMALL SAFETIES:	3% RESET POINT
BANK 1 :	3% RESET
BANKS 2&3:	3% RESET
PRESS HTRS:	5 BANKS
PRESS SPRAY:	AVAILABLE 2220/2170 PSIA
RC PUMPS :	AVAILABLE
RPS TRIP :	TURBINE
RPS TRIPS DEFEATED :	NONE
SFAS TRIP STATUS :	1500 PSIG RCS
:	4 PSIG BLDG PRESS DEFEATED
DIESEL GENERATORS :	2
OFFSITE POWER:	AVAILABLE
DECAY HEAT:	1.0 ANS
PORV:	2450/2400 PSIG
PZR SAFETIES :	2500/2475 PSIG

Figure 8A-6

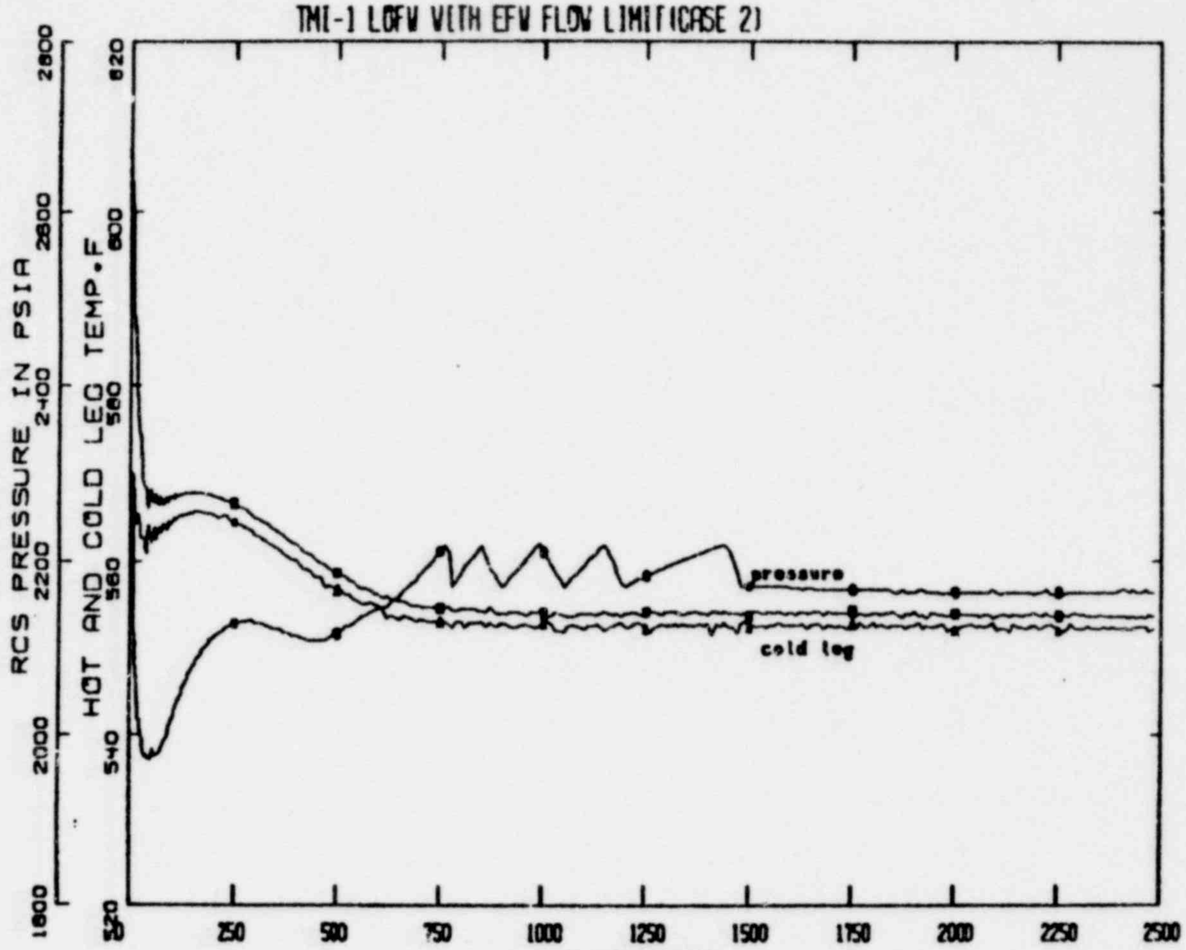

Figure 8A-6
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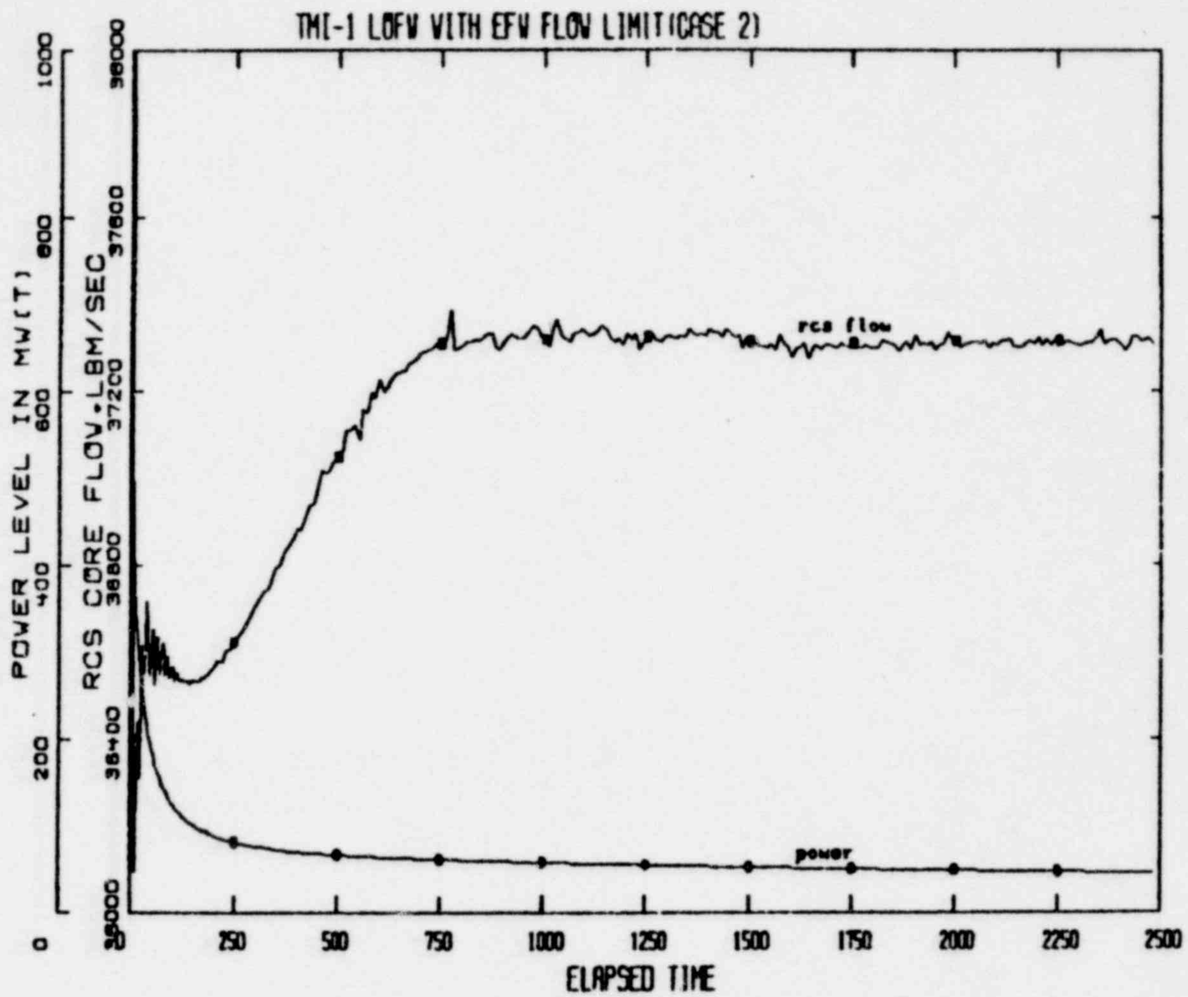
1438 075



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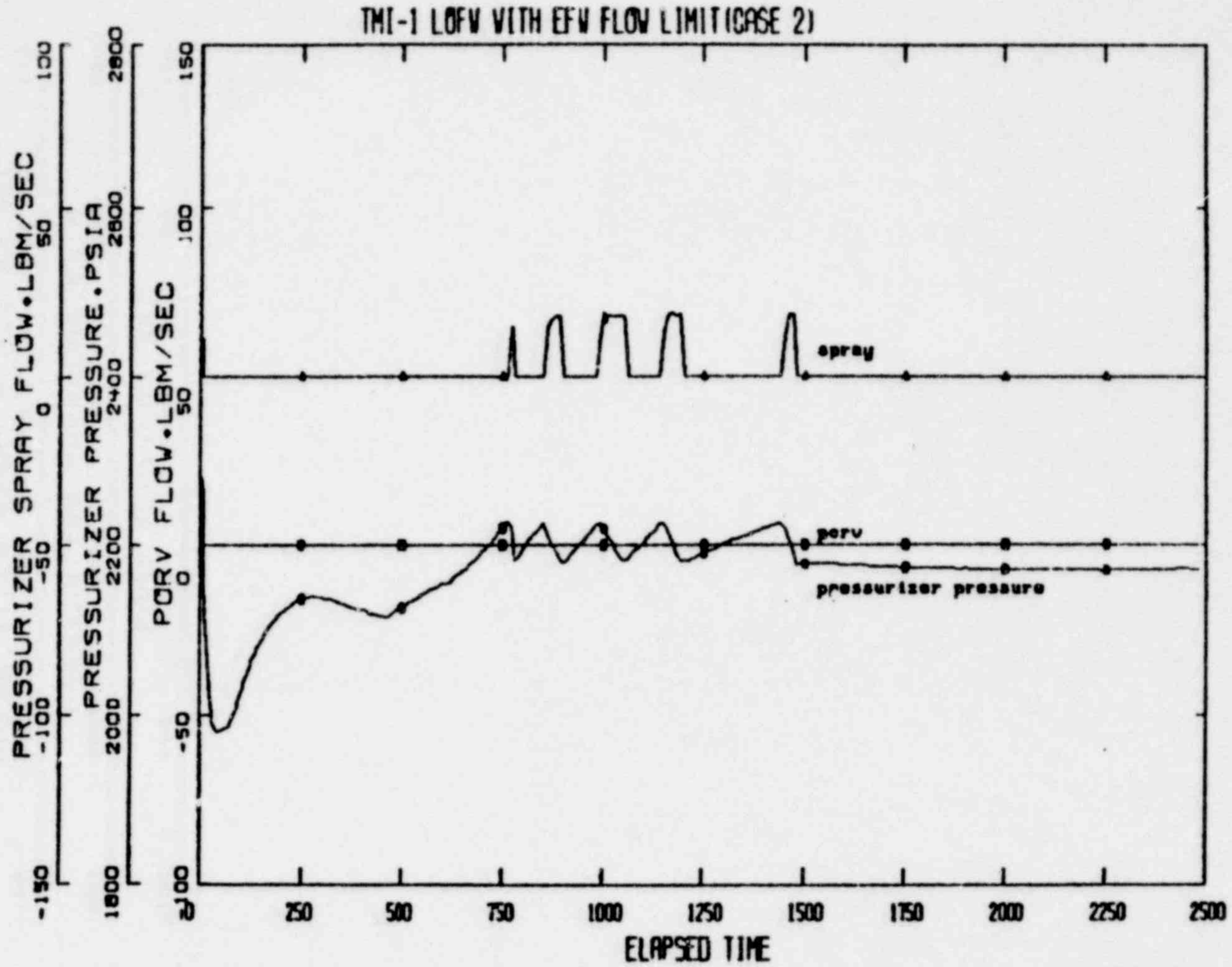

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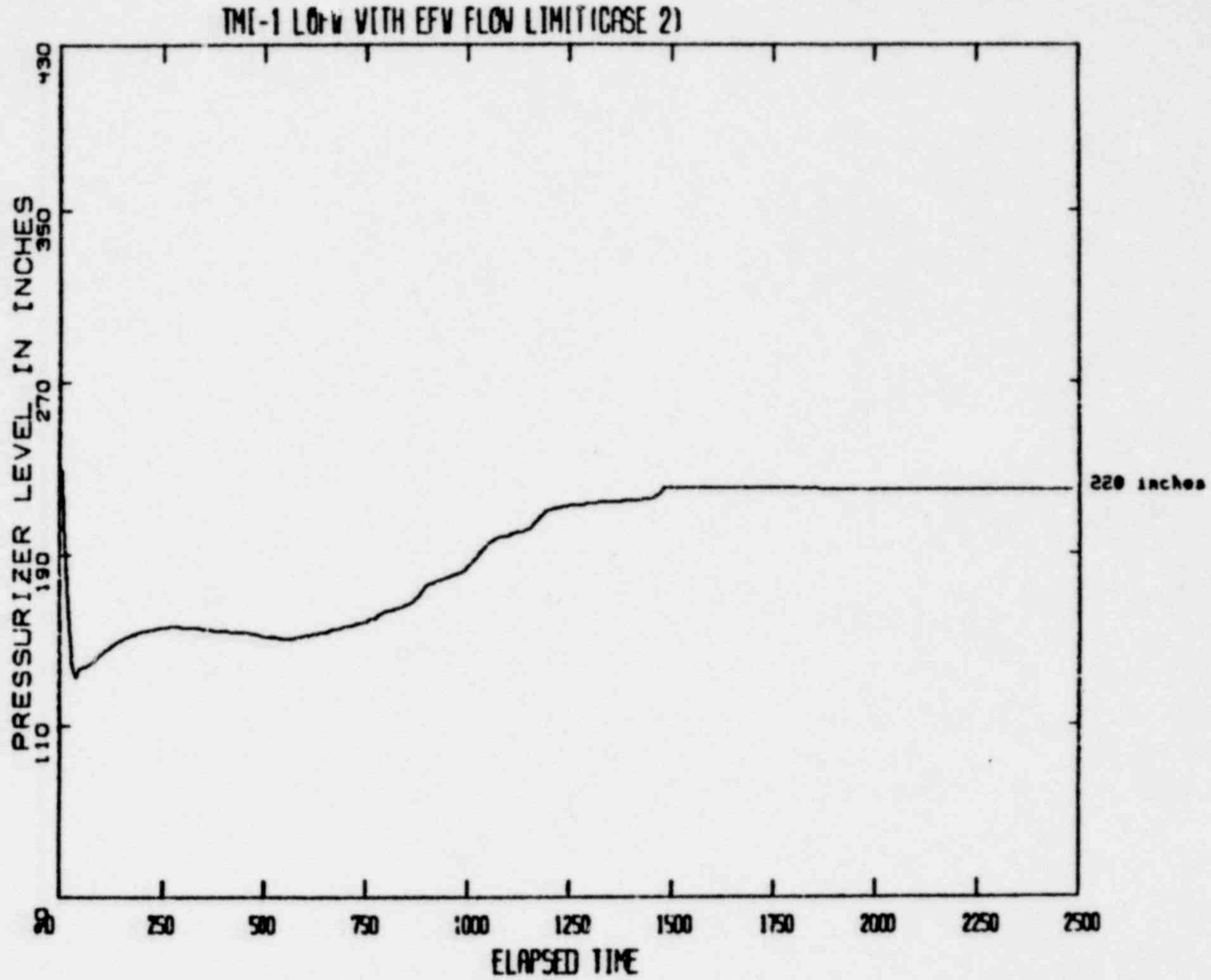

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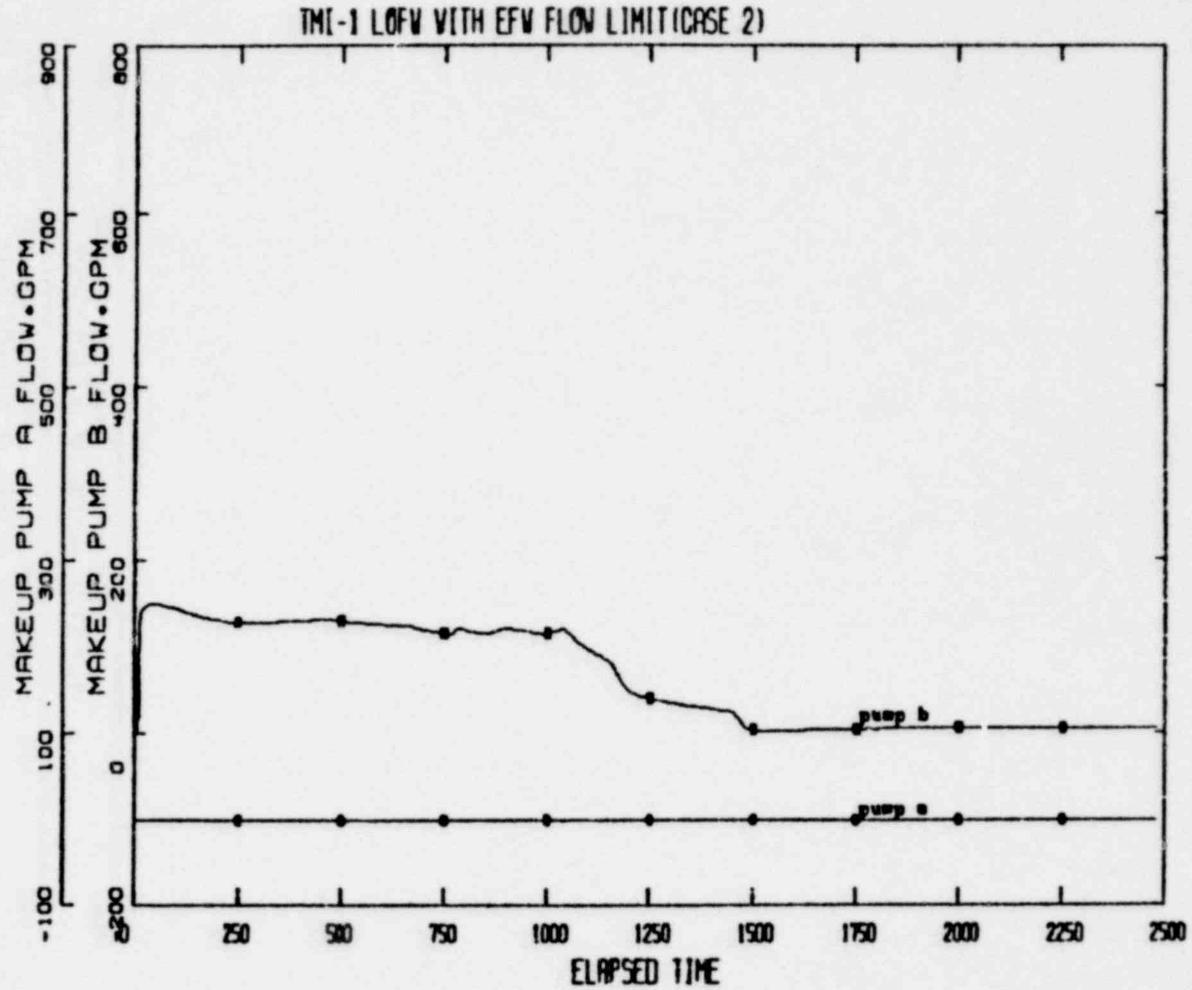

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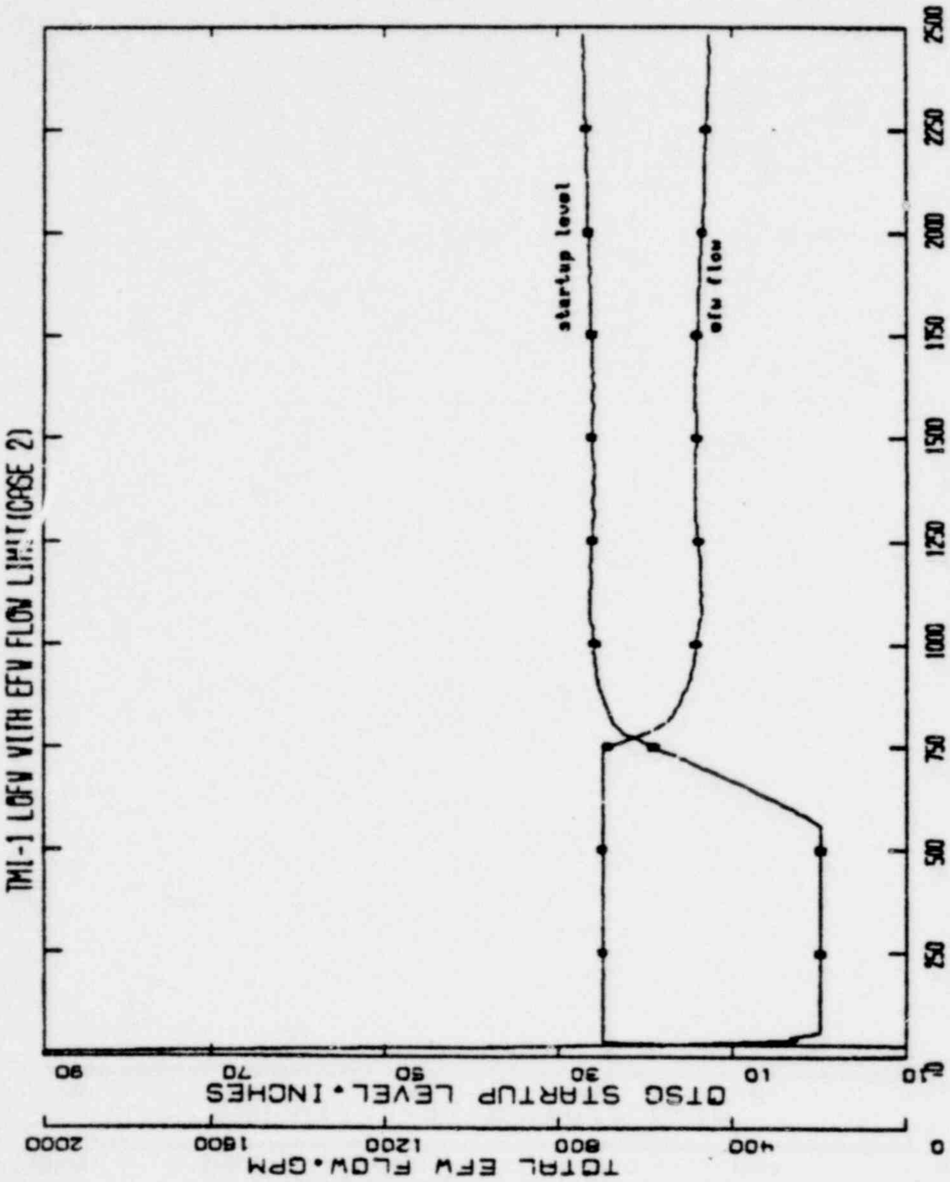

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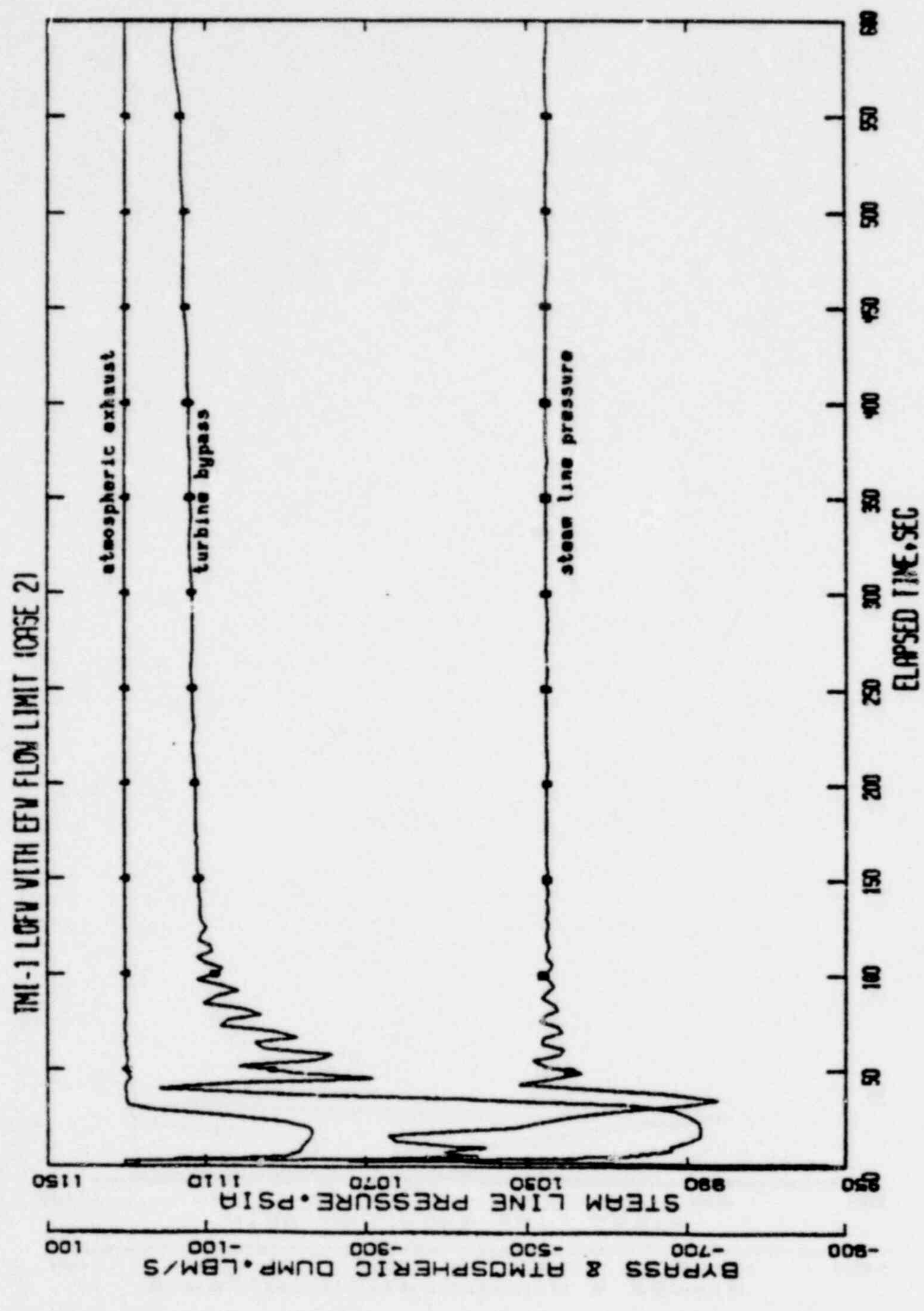

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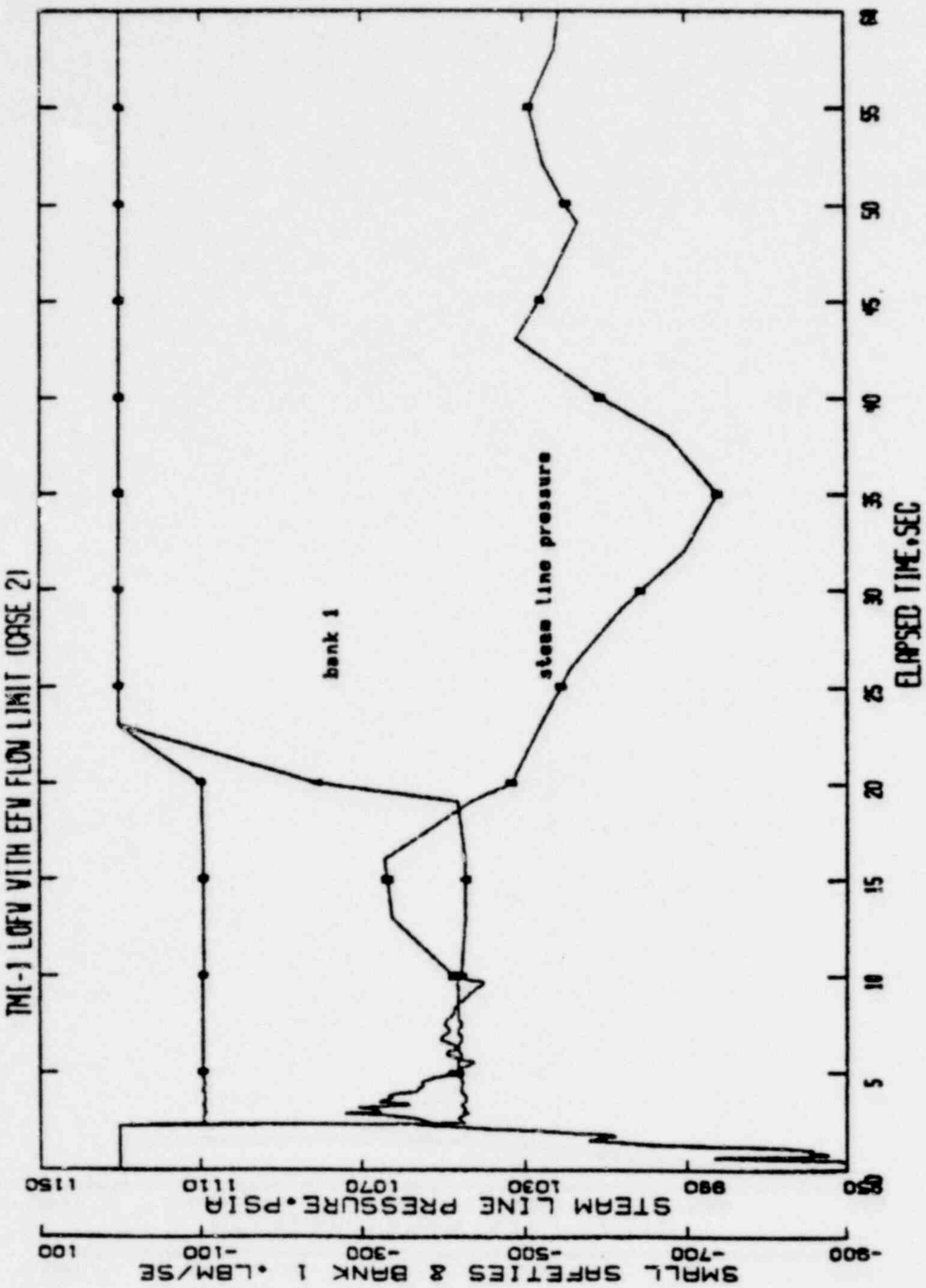

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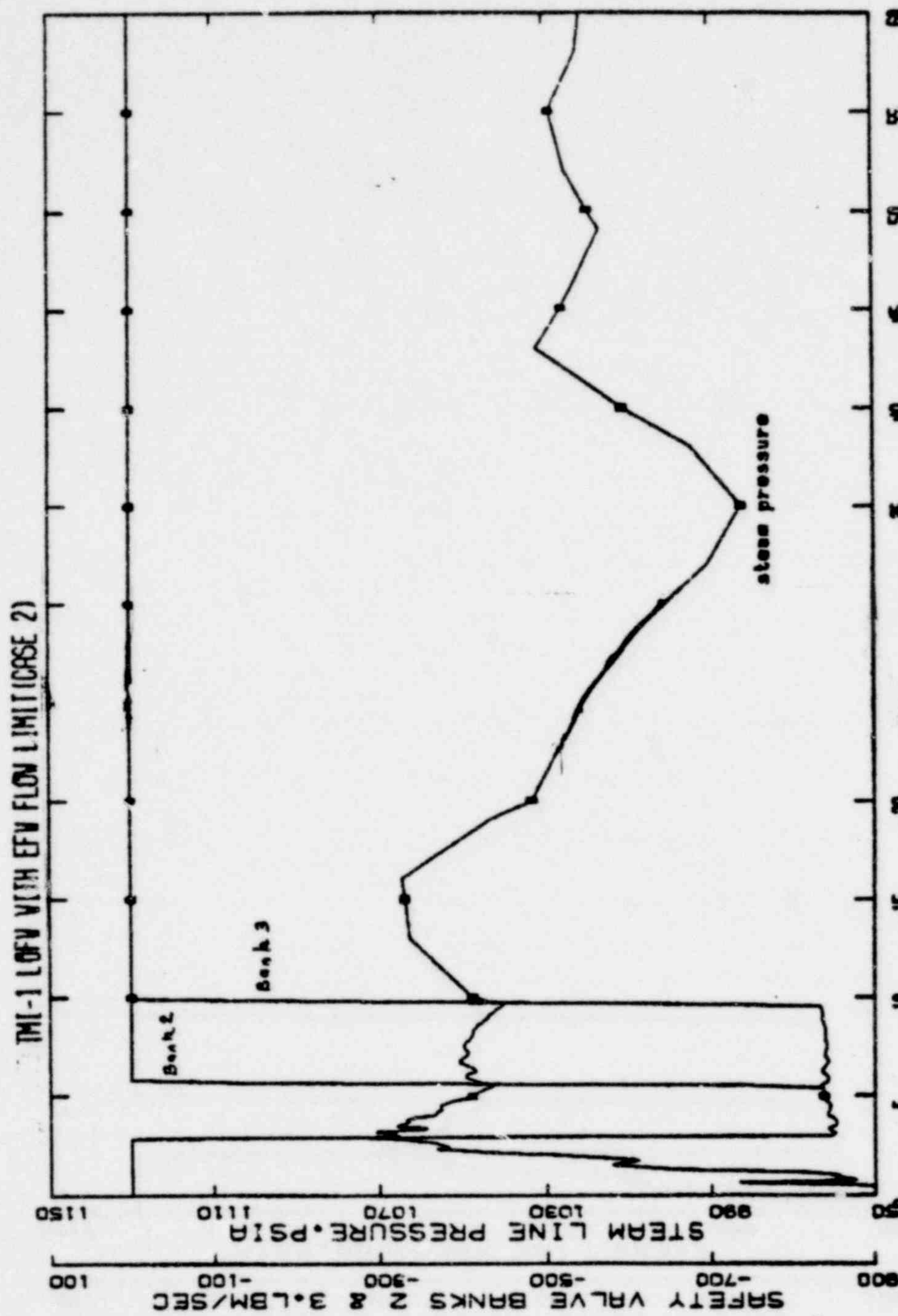

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Sheet 9

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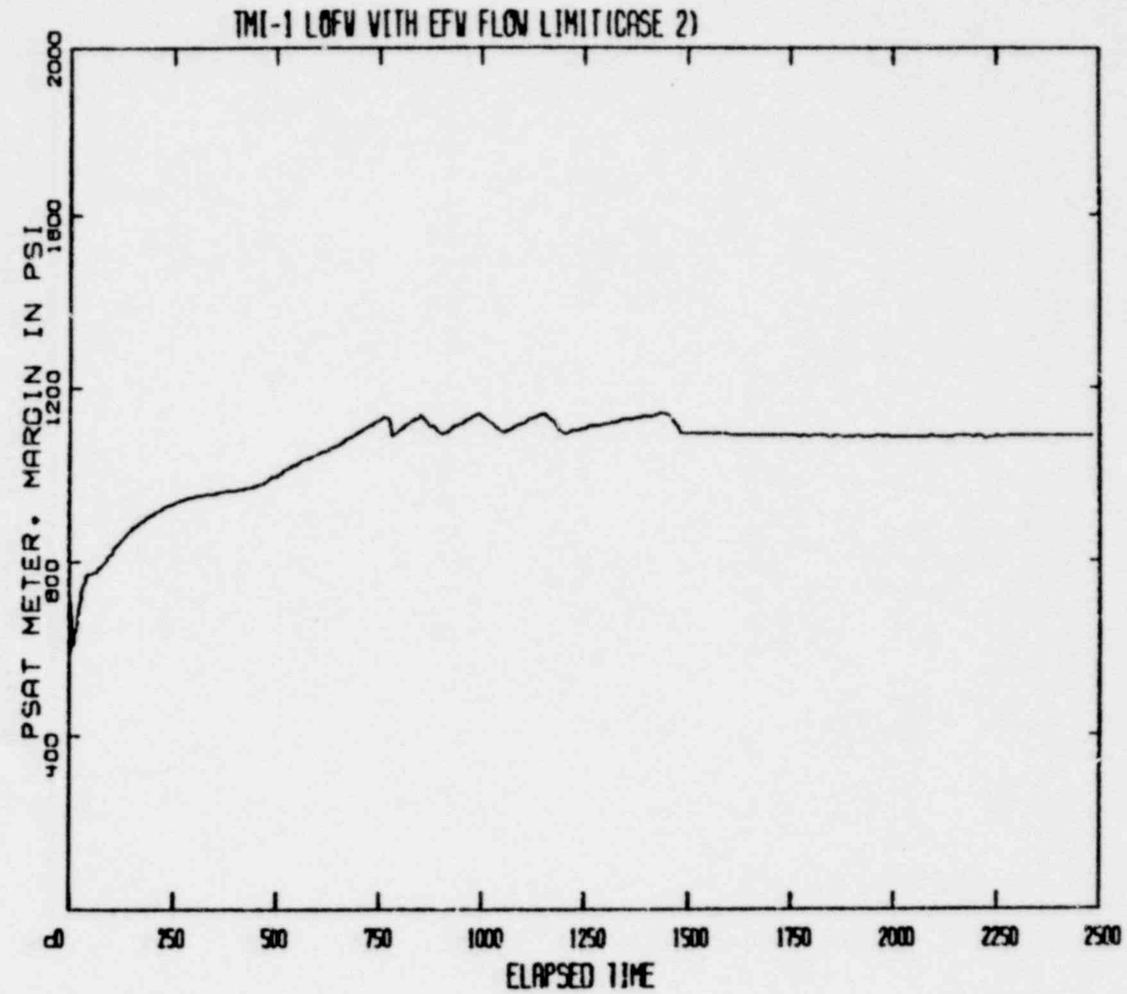


Figure 8A-6
Sheet 10

1438 084

STATION BLACKOUT ANALYSIS ASSUMPTIONS

BASE CASE

MAKEUP:	UNAVAILABLE
LETDOWN:	ISOLATED
EFW MECH:	UNAVAILABLE
EFW STEAM:	AVALIABLE
EFW CAPACITY:	920 GPM
ATMOS DUMP:	AVAILABLE
TURBINE BYPASS:	UNAVAILABLE
SMALL SAFETIES:	3% RESET POINT
BANK 1 :	3% RESET
BANKS 2&3:	3% RESET
PRESS HTRS:	UNAVAILABLE
PRESS SPRAY:	UNAVAILABLE
RC PUMPS :	UNAVAILABLE
RPS TRIP :	LOOP
DIESEL GENERATORS :	0
OFFSITE POWER:	UNAVAILABLE
DECAY HEAT:	1.0 X ANS
PORV:	2450/2400 PSIG
PRESSURIZER SAFETY:	2500/2475 PSIG

Figure 8A-7

1438 085

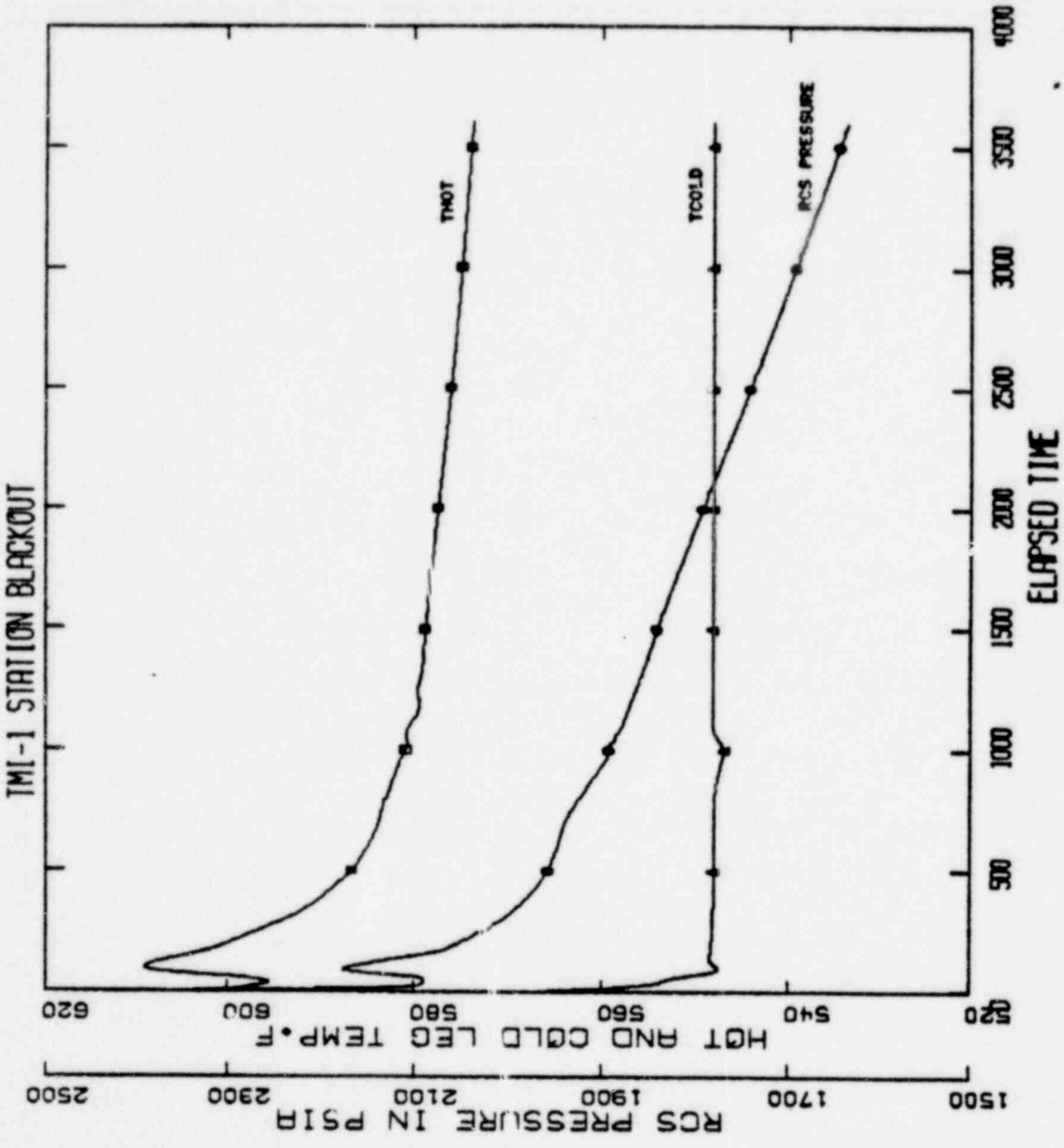


Figure 8A-7
Sheet 1

1438 086

Am. 6

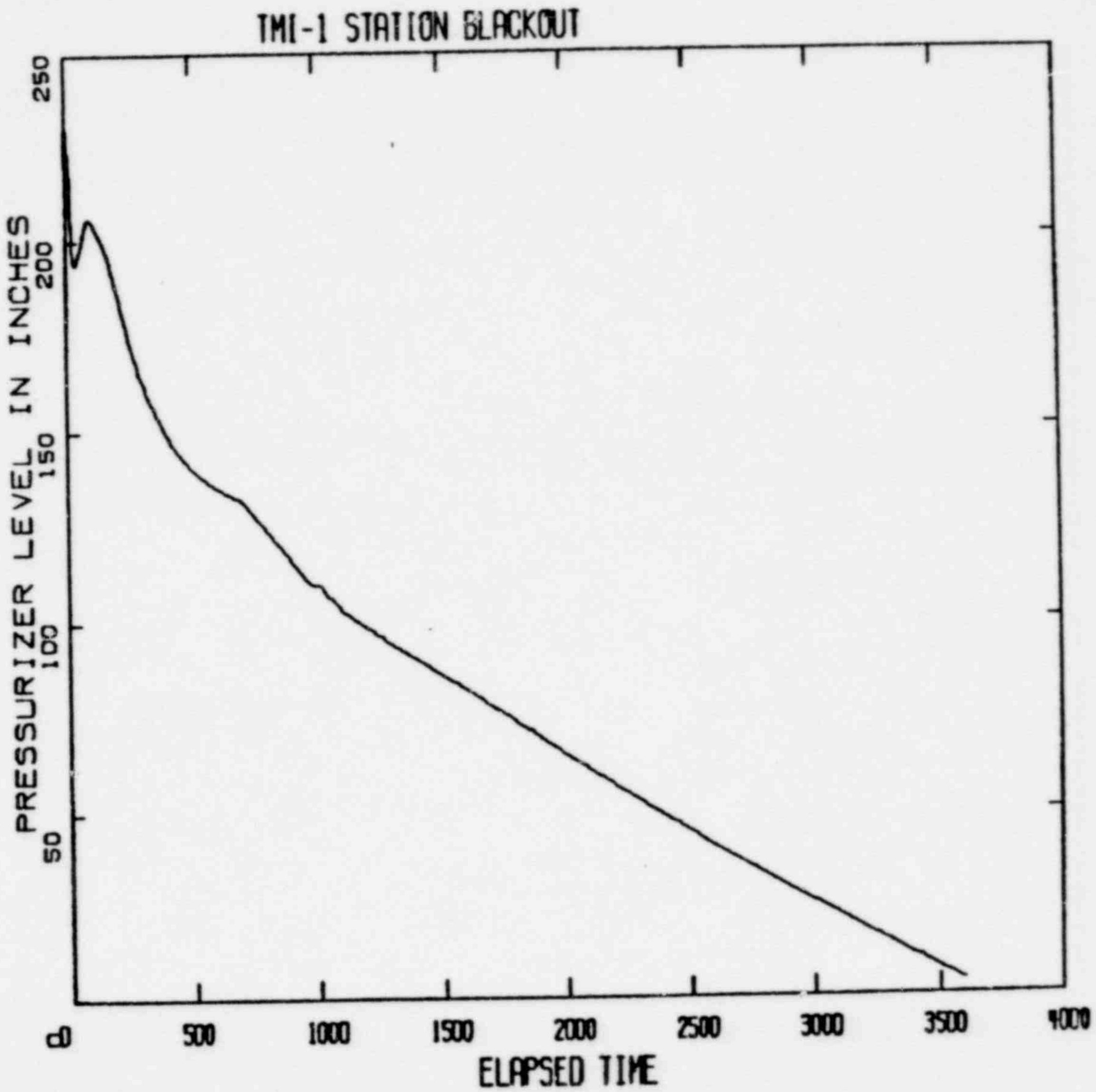
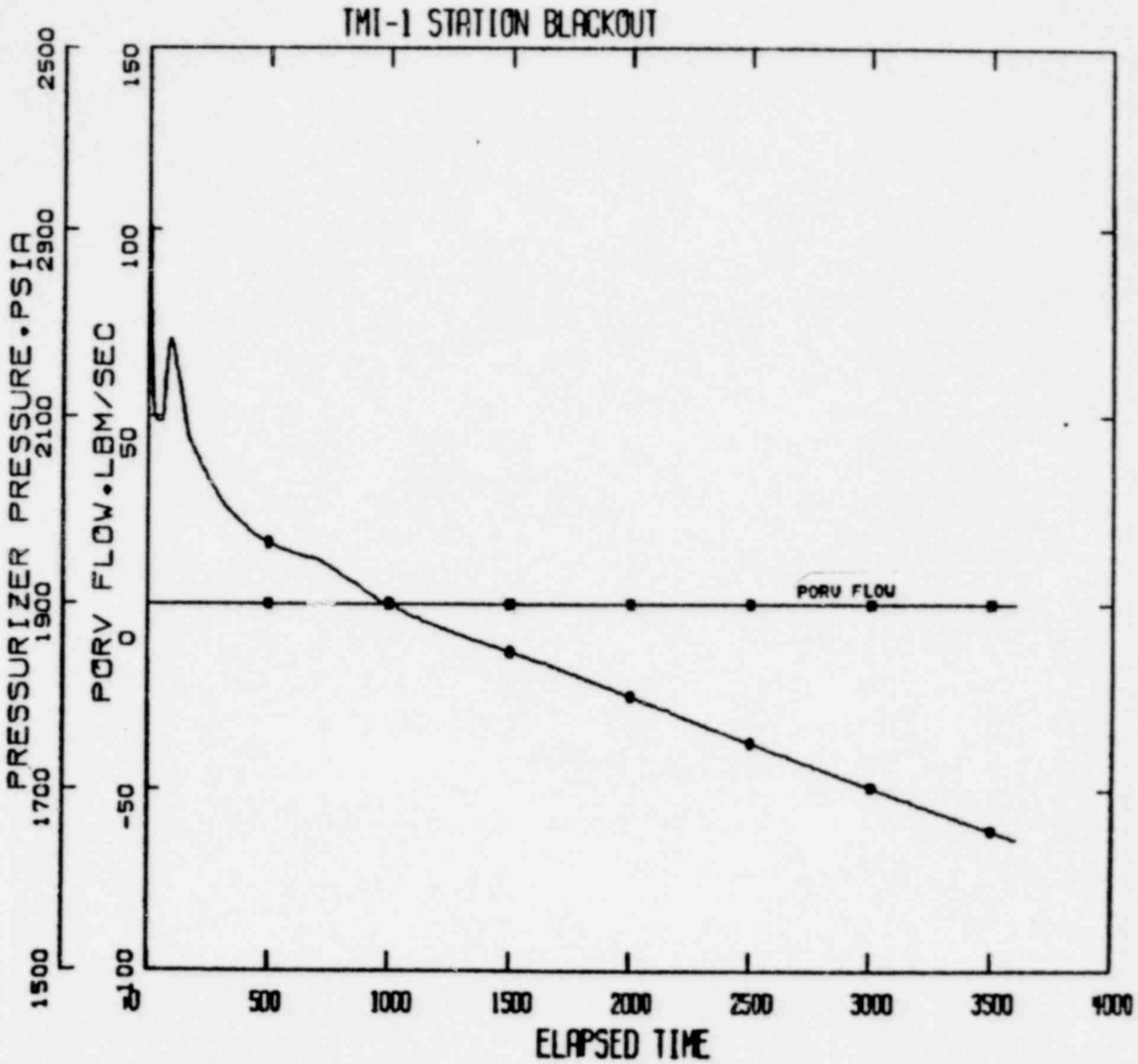


Figure 8A-7
Sheet 2

1438 087



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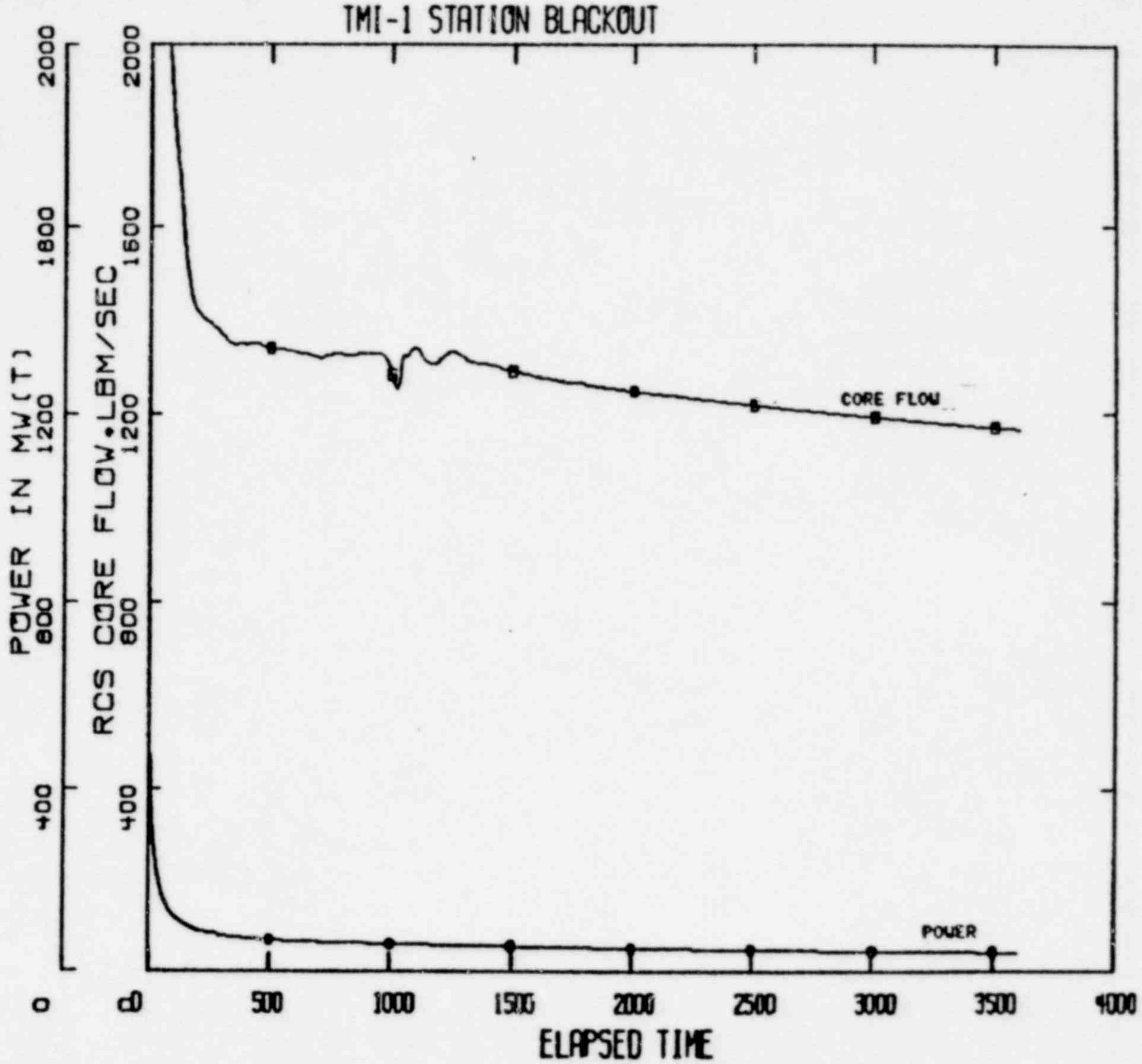
Am. 6

Figure 8A-7
Sheet 3



1438 089

Am. 6

Figure 8A-7
Sheet 4

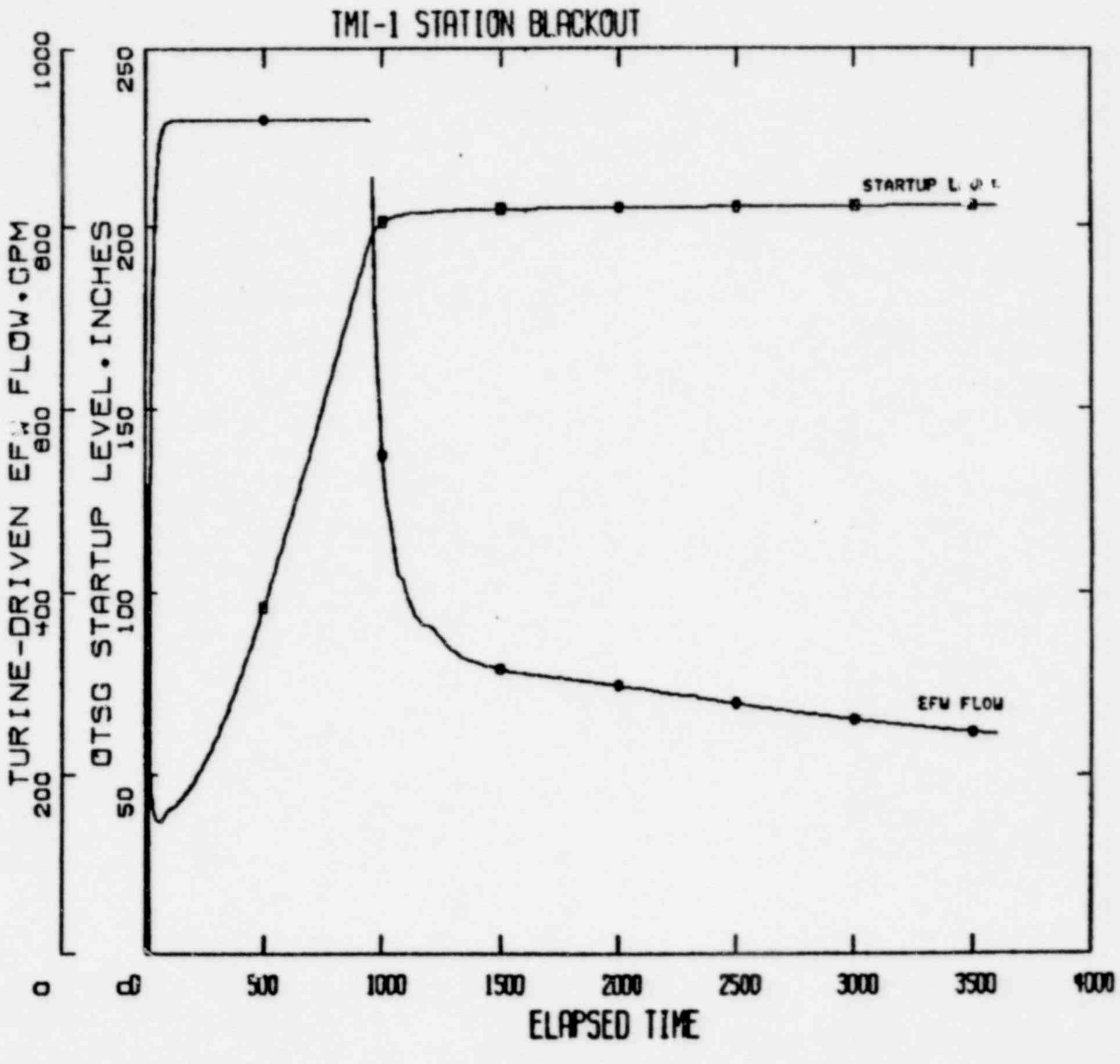


Figure 8A-7
Sheet 5

1438 090

Am. 6

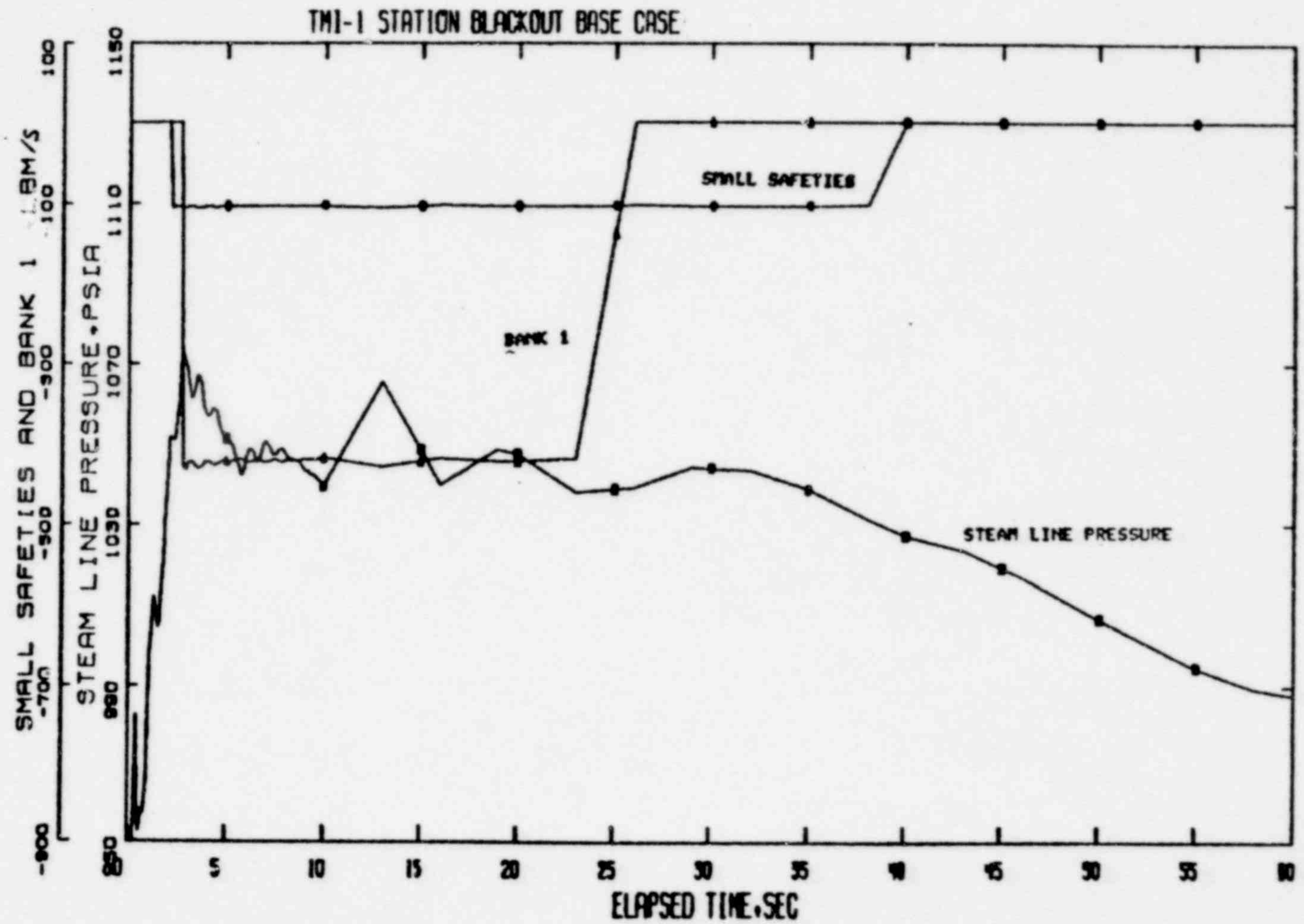
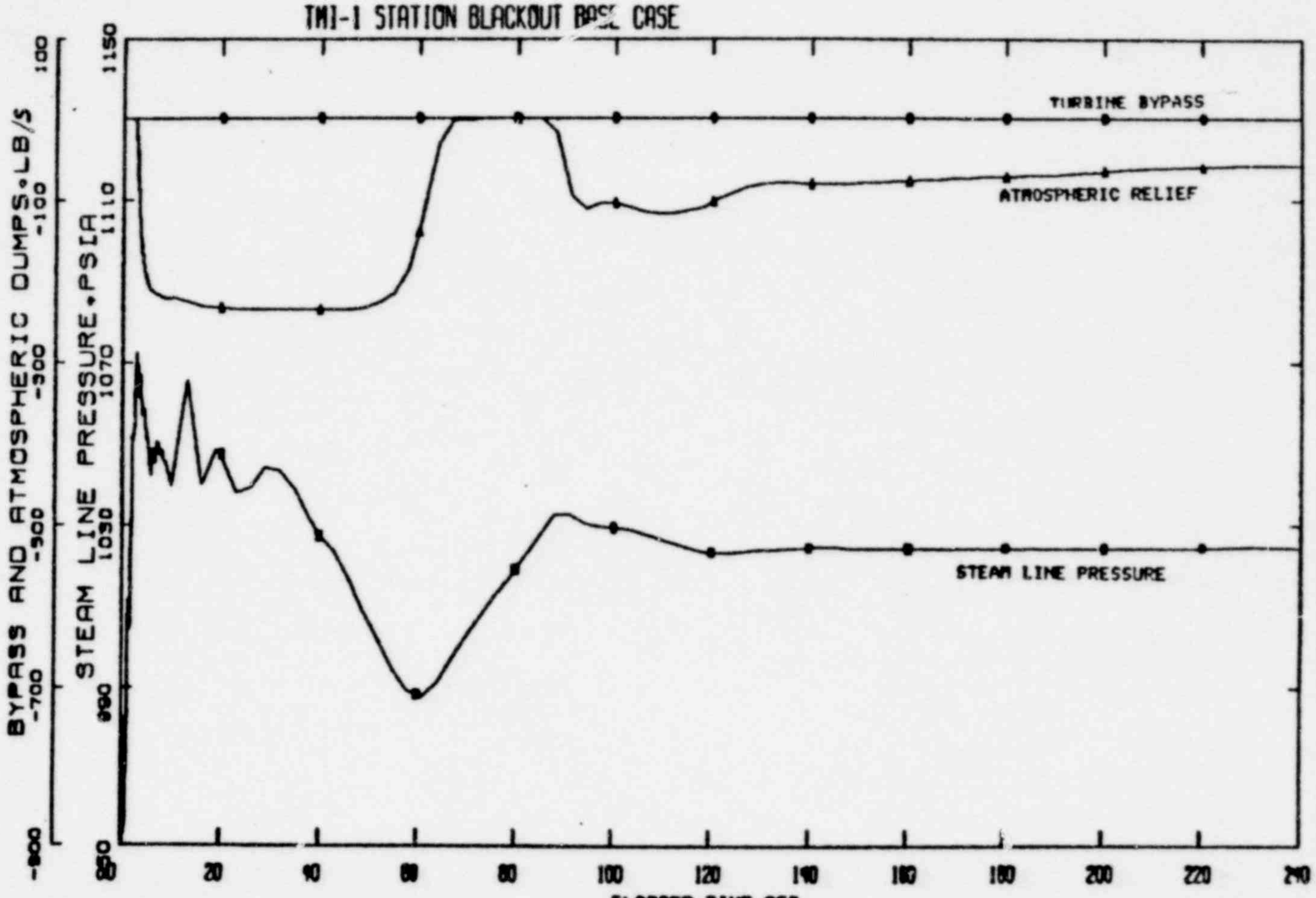


Figure 8A-7
Sheet 6

1438 091

Am. 6



1438 092

Figure 8A-7
Sheet 7

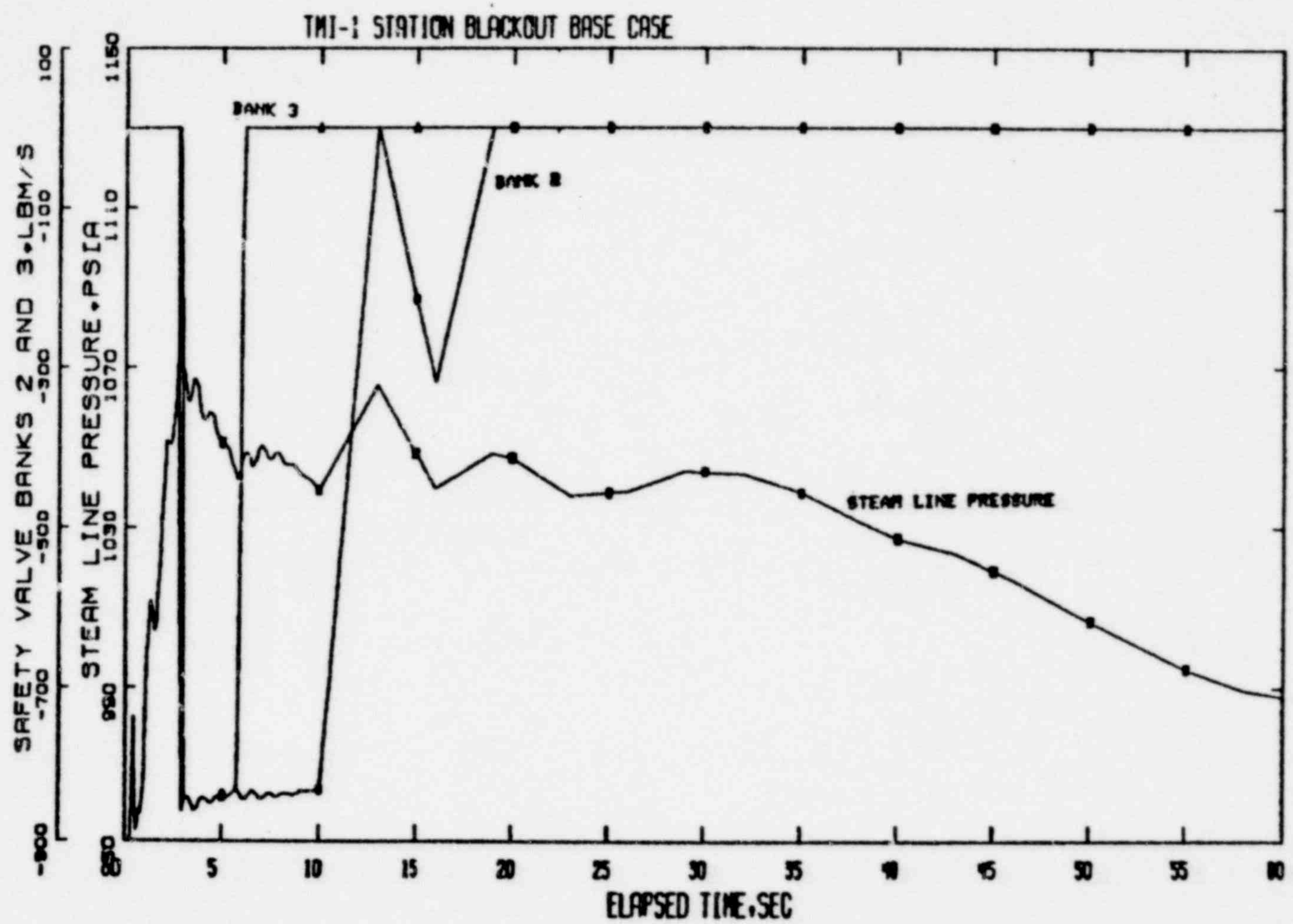


Figure 8A-7
Sheet 8

1438 093

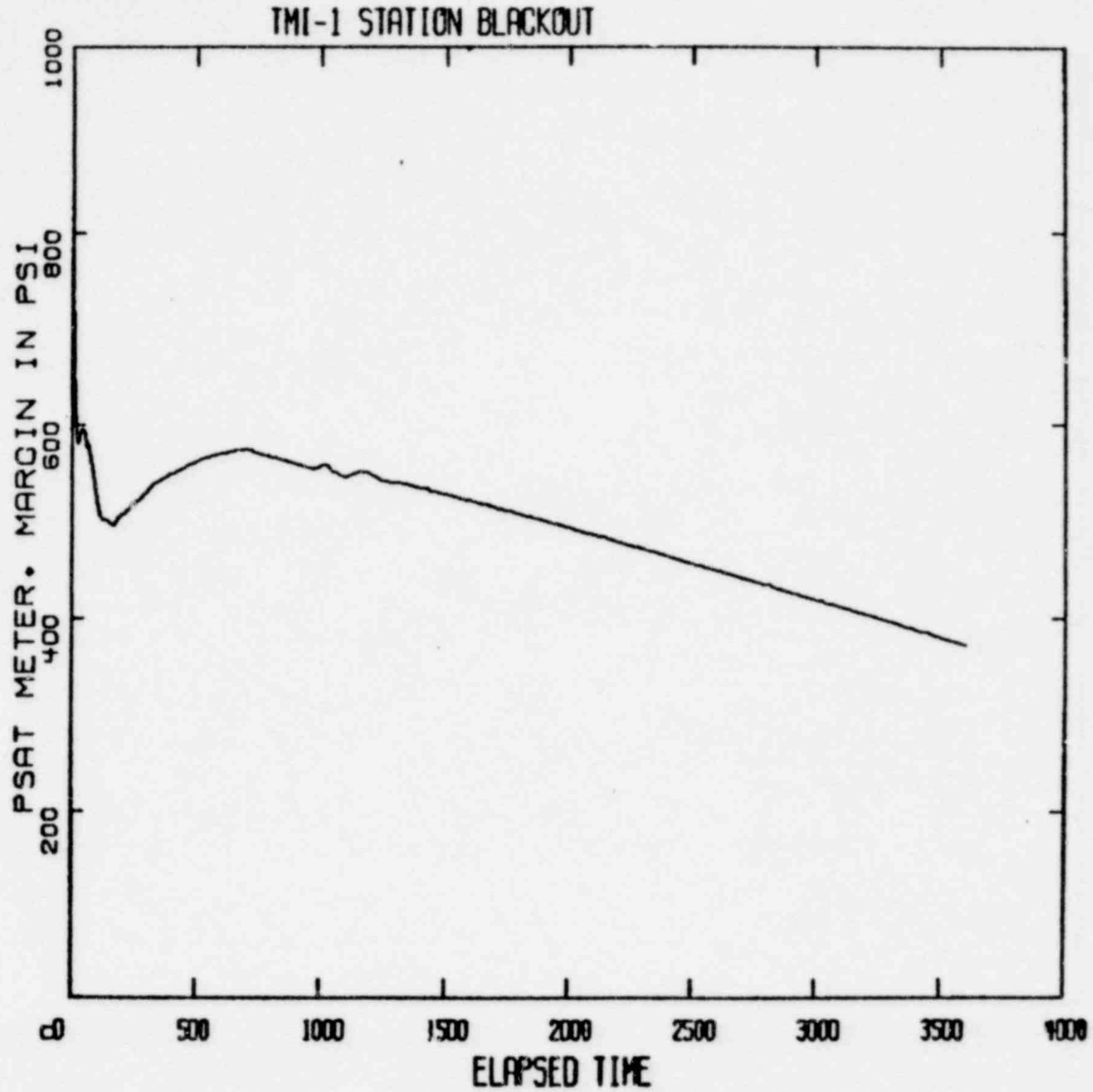


Figure 8A-7
Sheet 9

1438 094

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 either SG, the plant shall be shutdown within 48 hours and placed in a condition not relying onf SG's for cooling within 12 additional hours. If a flow path is unavailable to both SG's or if a 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.

1438 095

SUPPLEMENT 1, PART 1

1438 096

QUESTION:

3. Provide the results of the detailed loading study on the diesel generators that confirm the acceptability of adding the AFW pumps. Provide your schedule and a description of the actual testing planned.

RESPONSE

The engireered safeguard diesel-generator loading sequence table (Table 3-1 attached) represents the heaviest loading on one diesel-generator (D/G) in the event that the redundant D/G failed to start. The total automatic loading of the D/G, including the new loads as described in the Table 3-1 notes, is 2807 kilowatts. This total also includes the reactor building spray pump load which is actuated only when the containment pressure exceeds 4 psig.

The valve operator loads are non-continuous and will have been removed within approximately two minutes after actuation. Thus, the D/G load will reduce to 2716 kilotwatts.

Manual loads M-3, 4, 5 and 7 are actuated within 30 minutes increasing the maximum D/G load under an engineered safeguard actuation to 2913 kilowatts.

For a loss of offsite power situation only, the total automatic D/G load is 1859 kilowatts (see Items 1-2, 1-3, 2-1, 2-2 & 4-1). All necessary manual loads, as described in Table 3-1 notes, can then be actuated. Under these conditions, the maximum D/G load is 2817 kilowatts.

In this study, no credit has been taken for reduction in pumping power requirements that occur with time following a LOCA. Likewise, no credit has been taken for operation of an engineered safeguard function at less than 100 percent capacity, i.e., diverse engineered safeguard functions were considered to be running at 100 percent simultaneously. Thus, the total required D/G loading is less than the diesel-generator 2000 hour rating of 3000 kilowatts.

Startup testing will be conducted in order to verify the diesel load sequence. This testing will be conducted using an appropriately modified version of the preoperational startup test procedure. This procedure will be provided for NRC review by February 1, 1980.

1438 097

TABLE 3-1

Engineered Safeguard Diesel-Generator Loading Sequence

Loading Sequence	Item No.	Quantity	Description	Notes	Total Rated HP or KW	Total BHP	Eff	Total Load KW
Auto Load Block 1	1-1	1	Makeup Pump (High Pressure Inj.)	1	700 HP	700	0.93	562
	1-2	1	Decay Heat Pump (Low Pressure Inj.)	3	350 HP	340	0.92	276
	1-3	Lot	Miscellaneous Valves	2	91 KW			91
	1-4	3	Inverters	1*	45 KW	35 KW	0.78	45
		Lot	Miscellaneous Loads	1	372 KW			372
	1-5	2	Battery Chargers	1	37.5 KW	37.5 KW	0.8	47
Auto Load Block 2	2-1	2	Reactor Building Ventilation Units	3	150 HP	92	0.9	76
	2-2	1	Reactor Building Emergency Cooling River Water Pump	3	400 HP	380	0.92	308
Auto Load Block 3	3-1	1	Nuclear Services Closed Cooling Pump	1	125 HP	115	0.91	94
	3-2	1	Nuclear Services River Water Pump	1	150 HP	150	0.92	122
	3-3	1	Decay Heat Closed Cooling Pump	1	100 HP	91	0.91	75
	3-4	1	Decay Heat River Water Pump	1	200 HP	180	0.91	148
Auto Load Block 4	4-1	1	Reactor Building Spray Pump	10	250 HP	240	0.91	197
	4-2	1	Air Cool F. for DH & NSP AHE-15	1	3 HP	3	0.72	3
	4-3	1	Screen House AHU AHE-27	1	15 HP	15	0.9	13
	4-4	1	Screen House Vent. Equip. Pump SW-P-2A	1	15 HP	15	0.9	13

1438 098

TABLE 3-1

Engineered Safeguard Diesel-Generator Loading Sequence (cont'd.)

Loading Sequence	Item No.	Quantity	Description	Notes	Total Rated HP or KW	Total BHP	Eff	Total Load KW
Auto Load Block 5	5-1	1	Emergency Feedwater Pump	1,4	450 HP	450	0.92	365
			TOTAL AUTO LOAD					2807
Manual Loads								
	M-1	1	Instrument Air Compressor	8	60 HP		0.91	49
	M-2	1	Spent Fuel Pump	8	40 HP		0.9	33
	M-3	1	Control Building Emergency Supply Fan	7	50 HP		0.9	41
	M-4	1	Control Building Exhaust Fan AHE-19	7	10 HP		0.85	9
	M-5	1	Control Building Chiller	7	130 KW			130
	M-6	1	Penetration Cooling Fan	8	75 HP		0.9	62
	M-7	1	Chilled Water Supply Pump	7	20 HP		0.89	17
	M-8	1	Spent Fuel Pump Cooling Fan	8	2 HP		0.82	2
	M-9	1	Turning Gear Motor	9	60 HP		0.9	50
	M-10	1	Boric Acid Pump	11	2 HP		0.82	2
	M-11	4	MSV1A, B, C, D	2	45 HP		0.9	37
	M-12	Lot	Pressurizer Heaters	5,9	126 KW			126
	M-13	1	Post LOCA Hydrogen Recombiner	6	45 KW			45
	M-14	3	Turb. Oil Lift Pumps	9	30 KW		0.87	34
	M-15	1	Turning Gear Lube Oil Pump	9	50 HP		0.9	41
	M-16	1	Backup Inst. Air Comp.	12	5 HP		0.82	8

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NOTES

1. Loaded for loss of offsite power with or without ES actuation.

1.* Including estimated new inverter loads.

2. Non-Continuous operation (i.e. operates for short time only and then is automatically turned off).

3. Manually turned off on loss of offsite power only.

4. New automatically started load.

5. New manually started load within 2 hours of loss of offsite power only.

6. New manually started load approximately one day after an ES actuation.

7. Manually started loads within 30 minutes with or without an ES actuation.

8. Manually started loads contingent upon D/G capacity.

9. Manually started loads locked out on ES actuation.

10. Started only upon ES actuation by Reactor Building pressure of greater than 4 psi.

11. Backup to boration source.

12. Starts on loss of instrument air compressor.

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RESPONSE TO APPENDIX A, QUESTION 1a

The design basis event for sizing the Auxiliary Feedwater System (AFWS) is Loss of Feedwater (LOFW) with a concurrent Loss of Offsite Power (LOOP), and subsequently loss of reactor coolant pumps. The pertinent parameters for this accident relative to the AFWS are design flowrate and required time to full AFWS flow. These parameters reflect the functional requirements of the AFWS to a) remove decay heat, and b) provide a smooth reactor coolant flow transition from RC pump operations to natural circulation. The design values which resulted from this analysis are 720 gpm deliverable to the steam generators within 40 seconds of the initiation signal. The 40 second time was chosen to allow the AFWS to inject feedwater and begin increasing SG level to the 50% operating range level, required for natural circulation, prior to completion of the RC pump coastdown. At that time, the design flowrate was selected to be equal to or greater than the decay heat generation rate. Since decay heat rate changes with time, other values than 40 seconds and 720 gpm could have been used and been acceptable. All other transients which either require or assume the availability of AFW in the Safety Analysis use the design values derived from the LOFW analysis. The results of these other analyses are acceptable and are referenced in Table 1 attached.

Accidents 1, 2 and 3 of Table 1, which specifically require AFW for mitigation, were analyzed using the AFWS performance criteria established by the LOFW accident. The results of these analyses were acceptable and are described in the FSAR sections noted in Table 1. The other accidents listed in Table 1 (4-12) do not require AFW for mitigation though the availability of the AFWS, as defined by the performance criteria established by the LOFW accident, is assumed. The results of those analyses were acceptable and are described in the FSAR sections noted in Table 1.

Addressing the events included in the NRC Appendix A question 1a which have not been included in Table 1, we have the following comments:

LMFW w/loss of onsite and offsite AC power - This event was not a design basis of the plant and, consequently, is not included in Chapter 14 of FSAR.

Plant Cooldown - Plant cooldown with AFW is a new issue as stated in Reg. Guide 1.139 and not a design basis for this plant. The NRC has not indicated how Reg. Guide 1.139 is to be applied to operating plants. The extent of plant cooldown for which the AFWS is designed is discussed in FSAR Section 14.1.2.8.3d.

Turbine Trip with and without bypass - This event does not affect the AFWS unless MFW fails in which case the loss of MFW event previously addressed would bound the AFWS desi

Main Steam Isolation Valve Closure - Again, this event does not directly affect the AFWS unless MFW is lost as discussed above.

1438 101

Main Feed Line Break - This event was not a required analysis for this plant and is not included in FSAR Section 14. Main Feedline Break is a more abrupt case of LOFW and results of an analysis would be approximately the same.

Small Break LOCA - The AFW criteria assured for this event is described in Topical Report RAW-10052 updated by Letter Report J.H. Taylor B&W to S.A. Varga NRC 7/16/78 and the recently submitted B&W Report titled "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 FA Plant," 5/7/77.

RESPONSE TO APPENDIX A, QUESTION 1b

The design basis event for sizing the AFWS is LOFW as discussed in response to question 1a. The acceptance criteria for the other transients which include or assume AFW are given in Table 1.

The RCS cooling rate is not a limit relative to accident acceptance criteria. The safety limit for all transients which use AFW for mitigation is that the core remain cooled with ultimate acceptance criteria being those addressed in Table 1. For transients which result in draining the pressurizer or for which natural circulation is slowed or interrupted, restoration of pressurizer level and subcooling is accomplished by swelling due to core heat input and inventory restoration by HPI.

Steam Generator level is not based on decay heat removal rate or cooldown capability. SG level is set low for decay heat removal and high for natural circulation. It is also set high for Small LOCA as described in Topical Report RAW-10052, and in the B&W Report "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks".

RESPONSE TO APPENDIX A, QUESTION 2

As discussed in response to 1a) above, the design basis event regarding AFWS design requirements is loss of main feedwater with concurrent loss of RC pumps; the analysis assumptions for this event are listed below keyed to the letters of the question. Corresponding technical justifications where not specifically listed below, is based on licensing requirements and prudent engineering judgment at the time of the analysis.

a) Max. Rx Power - 100%

b) Time delay initiating event to Rx trip - The reactor will trip on high RCS pressure approximately 5-10 seconds after a LOFW event. The initiation signal for AFW is loss of main feedwater.

c) AFWS initiation signal and time delay - The AFW initiation signal for the LOFW event is loss of both main feed pumps as sensed by steam inlet valve position on the two main feed pump turbines. The design basis time delay from initiation event to full flow of AFW flow into SG is 40 seconds.

d) SG level at initiation event - Steam Generator inventory is dependent on power level. In the most restrictive case, AFW will be fed into the steam generators before they boil dry.

e) SG inventory and decay heat - For discussion of water inventory see d) above. Reactor decay heat rate is shown in FSAR Table 14-13.

f) Max. SG Pressure - 1103 psig

g) Min. no. of SG - The number of generators was not specified in the analysis, heat removal capability is the pertinent parameter and can be accomodated by 1 SG.

h) RC Flow Condition - Both natural circulation and RC pump operation were analyzed.

i) Max. AFW inlet temperature - The maximum AFW inlet temperature assumed was $90^{o}F$.

j) Steam, Feed Line Break time delay - The feedwater line break was not a required analysis for this plant. Refer to FSAR Section 14.1.2.9 for steam line break analytical information.

k) Main Feed Line volume and temperature between SG and AFWS - R/A - There is not piping connection between the MFWS and AFWS.

l) SG Normal Blowdown - N/A - The OTSG's do not have a blowdown system.

m) Water and metal sensible heat used - Plant cooldown capability was not a design basis for AFWS. 1×10^6 BTU/^{o}F was used for removal of sensible heat from power operation to the 0 power reactor trip setpoint.

n) Time at hot standby etc. relative to AFW inventory - The AFW inventory was sized for decay heat removal for 1 day after Rx trip as discussed in FSAR Section 14.1.2.8.3d. The design basis for AFWS is not plant cooldown; the NRC Reg. Guide 1.139 requirements for operating plants have not yet been established.

TABLE 1

ACCIDENT DESCRIPTION	FSAR SECTION	ACCEPTANCE CRITERIA (1)
1) Loss of Coolant Flow	14.1.2.6	A, B
2) Loss of Electric Power	14.1.2.8	A, B
3) Steam Line Break	14.1.2.9	D
4) Uncompensated Operating Reactivity Changes	14.1.2.1	A, B
5) Start-Up Accident	14.1.2.2	A, B
6) Rod Withdrawal Accident at Rated Power Operation	14.1.2.3	A, B
7) Moderator Dilution Accident	14.1.2.4	A, B
8) Cold Water Accident	14.1.2.5	A, B
9) Stuck-Out, Stuck-In, or Dropped Control Rod Accident	14.1.2.7	A, B
10) Steam Generator Tube Failure	14.1.2.10	B, D
11) Rod Ejection Accident	14.2.2.2	C, D
12) Loss of Coolant Accident	14.2.2.3	D, E

NOTE: (1)

KEY	ACCEPTANCE CRITERIA	TECHNICAL BASIS
A	Max. RCS Press. - 110% Design	ASME Code
B	1.3 with BAW-2	SRP 4.4
C	200 cal./gram fuel limit	Reg. Guide 1.77
D	Acceptable Doses	10CFR100
E	Fuel Cladding $2200^{\circ}F$	10CFR50.46

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Reactor Coolant System Pressure

The pressure detectors are of the diaphram type and are located on the hot legs. Narrow range pressure, 1700psig to 2500psig, for each loop is recorded on a strip chart located on the operator's console. Wide range pressure from 0psig to 2500psig is recorded on a strip chart located beside the narrow range recorders.

Reactor Coolant Flow

The flow detectors are of the gentilli flow tube type in conjunction with a differential pressure instrument. There are four detectors located in each hot leg for a total of eight. Loop A and Loop B flows are displayed on vertical indicators located on the control console and total RC flow is recorded on a strip chart recorder.

Source Range Nuclear Instrumentation

There are two BF_3 proportional neutron detectors installed with a range from0.1cps to 10^6cps. A vertical indicator on the control console and a strip chart recorder above the console provide source range measurement in counts per second.

Reactor Coolant Pump Current

The RC pump motor current is indicated on console cc. The ammeters are scaled from 0 to 150% of full load current and are located adjacent to the respective RC pump control switches.

Analysis is being performed to develop additional guidelines that allow the reactor operator to recognize and respond to conditions of inadequate core cooling under the following conditions:

a) Loss of RCS inventory with the reactor coolant pumps operating.
b) Loss of RCS inventory without reactor coolant pumps operating.
c) Loss of the Decay Heat Removal System.
d) DNB Transient at Power

From these analysis guidelines for operator action and a description of the plant behavior for operator training will be developed. Additional procedural changes developed as a result of these analysis will be implemented within 30 days following the analysis.

Training in instrument response to various accident conditions that include inadequate core cooling is as follows:

(1) Procedures review
(2) Expected instrument and plant response to transients
(3) Safety analysis work shop
(4) TMI Control Room practical session

The lectures will address existing unmodified instrumentation as well as the short term modification to existing instrumentation.

1438 107

SUPPLEMENT 1, PART 2

1438 108

QUESTION

1. Your response to Questions 1 and 10k are not complete.

   A. Provide design drawings for the automatic EFW initiation modifications and provide "as-built" electrical drawings of the present EFW initiation system design.

   B. Provide the test plan (procedure) and results for the proposed EFW initiation functional test.

   C. Update Section 8.2.1 of the Restart Report to clarify what automatic EFW initiation signals will be control grade and what signals will be upgraded to safety grade requirements. It is our position that you have not adequately demonstrated that installation of safety-grade automatic EFW initiation signals (including low steam generator level) is not practicable prior to restart.

RESPONSE

A. Design drawings for the automatic EFW initiation modifications were provided during a meeting on November 19, 1979, to R. Fitzpatrick and S. Newburry of the NRC. The drawings submitted were:

SS-208-129	SS-209-564
SS-208-163	SS-209-565
SS-208-164	SS-209-590
SS-208-168	SS-209-591
SS-208-169	SS-209-660
SS-208-203	SS-209-661
SS-208-204	SS-209-662
SS-208-206	SS-209-663
SS-209-C31	SS-209-664
SS-209-032	SS-209-665
SS-209-463	SS-209-666
SS-209-464	SS-209-667
SS-209-465	SS-209-755
SS-209-490	SS-209-756
SS-209-491	SS-209-919
SS-209-563	SS-209-923

B. The test procedure will be submitted prior to restart testing and the test results will be submitted after completion of the tests.

1438 109

C. See Section 8.2.1. Engineering and procurement for a safety grade actuation system has been proceeding in parallel with that for the control grade system. It is scheduled for installation (without a low steam generator signal) by March 21, 1980. If plant start-up procedures commence subsequent to that date, the safety grade system will be available before start-up. Engineering for the addition of a low steam generator signal will be completed March 1, 1980. Long lead time items will be ordered by January 15, 1980. Delivery of safety grade transmitters and other instrumentation is anticipated to be 14 to 16 weeks. Installation should be completed by June 1, 1980

1438 110

QUESTION

2. Your response to Question 9 is not complete. Answer the indicated concerns. In addition:

   A. Provide design drawings of the modified instrument air system.

   B. Provide the test plan (procedure) and results for the proposed EFW control valve failure mode verification test.

   C. Provide the B&W evaluation on the consequences of overfilling the steam generator.

   D. Provide the calculations which indicate that operator action is required within 7 to 15 minutes to prevent potentially adverse steam generator overfill conditions. Include sufficient information to allow us to identify the allowable time delay beyond which the consequences would produce unacceptable effects. Describe each manual action required.

   E. Provide the revised procedures for preventing steam generator overfill conditions, and indicate that adequate operator training in these procedures has been completed.

RESPONSE

A. The design drawings of the modified instrument air system were provided to R. Fitzpatrick and S. Newburry of the NRC during a meeting on November 19, 1979. The drawings submitted were:

ECM Package WO-023	D-215-044
S-212-007 CH1102	E-215-053
S-212-007 CG1059	E-215-013
SS-208-712	E-215-011
B-210-527A	C-302-271
B-210-528A	C-302-272
B-201-044	E-304-275
B-201-043	E-304-277

B. The test procedure will be submitted prior to restart testing and the test results will be submitted after completion of the tests.

1438 111

C. An overfill condition occurred at Rancho Seco in March of 1978. Analysis performed by B&W and others, including detailed stress analysis, on the consequences of this incident revealed that no unacceptable damage to the steam generators, steam liners or other equipment had occurred and the the Unit could return to services without any major repairs. The Rancho Seco overfilling incident, therefore, provides assurance that no unacceptable consequences (i.e. steam generator rupture or main steam line break condition) will result from inadvertent overfilling. To verify these conclusions are valid for TMI Unit #1, a stress analysis will be performed on the consequences of flooding the TMI Unit #1 Main Steam line. The results of deadweight internal pressure and thermal expansion analysis will be provided by December 14, 1979. The possibility of transient hydraulic phenomena in the main steam line will be investigated and provided prior to start-up.

D. The times provided by B&W for steam generator overfill were based on calculations performed for B&W's generic 205 FA plant design and assumed flow rates as high as 1600 gpm to one steam generator. Based on above it has been concluded that a plant specific calculation should be performed and provided for your review. This evaluation will consider the specific pump and system arrangement installed at TMI-1 and the fact that the TMI-1 system can not deliver flow rates as high as that assumed in the 205 FA calculations. It is expected the TMI-1 specific calculations will demonstrate that times on the order of 10 minutes are available for operator action. The plant specific calculation for TMI-1 will be provided by December 12, 1979.

Based on the flow instrumentation and level instrumentation available to the operator, the back-up controls and instrument air supplies that have been provided for the auxiliary feedwater control valves and key components, and the reasons discussed below, reliance on operator action to prevent steam generator overfill conditions is considered warranted. For the long term, the emergency feedwater system will be upgraded to meet safety grade criteria and further reduce the probability of steam generator overfill. The conceptual design for our long-term design changes and our schedule for accomplishing these changes will be provided by January 4, 1980. Due to the extent of the changes required, it is not expected that these changes can be completed prior to start-up.

The TMI Unit #1 Integrated Control System (ICS) is designed to control auxiliary feedwater flow to preset steam generator levels. If Reactor Coolant pumps are available the ICS will control steam generator level to 30 inches on the start-up. If RC pumps are not available, the ICS will then control steam generator level to 50% of the operating range level. If automatic level control is not achieved or if overcooling conditions start to occur, operator will take manual control of the emergency feedwater control valves, EF-V30 A & B, using either the ICS manual controls or the back-up manual control station provided in the control room.

In the extremely unlikly event that control of EF-V30 A/B cannot be obtained from the control room, the operator has several other means to control auxiliary feedwater flow and steam generator level from the control room. These are as follows:

1) The motor driven emergency feedwater pumps can be started and stopped from the control room.

2) The flow of auxiliary water from the turbine driven pump can be controlled using motor operated EFW valves EF-V-2 A/B. These valves are powered from 480V Engineered Safeguards Control Centers and the controls available in the control room.

3) If necessary, the turbine driven emergency feedwater pump can be stopped using Main Steam valves MS-V13 A/B. The controls for these valves are available in the control room. MS-V13 A/B are powered using instrument air and control power from 125 volt vital power sources. Valves MS-V13 A/B also receive instrument from the back-up instrument air system.

At the time it is determined that remote control of EF-V30 A/B is not available, an auxiliary operator will also be dispatched to the intermediate building to take local control of the emergency feedwater control valves.

The TMI Unit #1 operating procedures will be modified prior to restart to reflect the overfill potential and actions to be taken to present overfilling the steam generator.

E. The procedure revisions for preventing steam generator overfill conditions are in the draft stage and have not completed the review and approval process required by Administrative Procedure #1001. The draft procedure revisions instruct the operators to:

1) Take manual control of EF-V30A/B from the control room as appropriate on high OTSG level to prevent overfill.

2) If control of EF-V30A/B is not available due to loss of control power on instrument air, then an auxiliary operator, in communication with the control room, takes manual handwheel control of EF-V30A/B to prevent overfill.

3) If sufficient time does not exist for the auxiliary operator alone to prevent overfill, the control room operator takes immediate action to stop EFW flow to the high OTSGs by shuting the EF-V2A/B valves, as appropriate, or by stopping the appropriate EFW pumps from the control room until local manual control of the EF-V30A/B and OTSG levels can be established.

When the procedure revisions are approved, adequate operator training in these procedures will be completed.

QUESTION

3. Your response to Question 8 is not complete. Provide the B&W study on transients such as loss of feedwater and loss of offsite power which verify that minimum EFW flow requirements meet the 550 gpm technical specification commitment. Provide the revised TMI-1 Technical Specifications for our review prior to restart. Justify the applicability of the B&W study to TMI-1.

RESPONSE

Attached is the B&W study (document identifier 86-1102587-00) on the auxiliary feedwater flow requirements following a loss of main feedwater. The analyses performed included the following assumptions:

1) The initial power level at the initiation of transient was 2772 Mwt.

2) The reactivity feedback coefficients used were representative of approximately 100 EFPD operation.

3) The ANS 5.1 decay heat curve was used with a 1.0 safety factor. The key input parameters used are documented in Table 3.2-1 of the B&W Report to the NRC dated May 7, 1979 and entitled "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant." The input parameters assumed in this study are applicable for TMI-1 which is only a 2535 Mwt plant. As noted in the B&W study, auxiliary feedwater flow rates as low as 370 gpm were found to provide satisfactory performance.

Additional work has also been done by B&W to demonstrate that 500 gpm auxiliary feedwater flow is adequate following upset transients such as the loss of offsite power and the loss of normal feedwater. These analysis will be provided in December, 1979. These analysis were based on the following assumptions:

1. The anticipatory trip circuitry is in place and the reactor trips within 1.0 seconds following the loss of normal feedwater flow.

2. The initial power level at the initiation of the transient is 102% of 2568 Mwt.

3. The highest worth control rod remains stuck out of the core following a reactor trip such that only a .1% shutdown margin is available.

4. A safety factor of 1.2 is applied to the ANS 5.1 decay heat curve.

5. The moderator coefficient of temperature is zero.

6. The OTSG inventory following the loss of normal feedwater is based on a conservative time history taken from test data. The initial inventory was conservatively assumed to the 18,400 lbm per OTSG.

7. As with the other analysis the PORC setpoint was assumed to be 2450 psig.

The Technical Specifications for EFW will be provided in January, 1980.

The acceptance criteria for the minimum auxiliary flow rate were that (1) the pressurizer does not go solid and (2) the electromatic relief valve does not actuate. An auxiliary flow rate of 500 gpm was found to meet these criteria. The assumptions used for these analysis are conservative for TMI Unit #1.