GPU Nuclear

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April 12, 1982
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Office of Inspection and Enforcement
Attn: R. C. Haynes
Region I, Regional Administrator
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
10 CFR 50.59 Report

In accordance with the requirements of 10 CFR 50.59, enclosed please find two copies of changes to TMI-1 systems as described in the FSAR. There were neither any changes to procedures requiring a change to the FSAR, nor test or experiments performed not described in the FSAR.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:CWS:vjf

Enclosures

cc: Director, Office of Inspection and Enforcement (40 copies)
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Director, Office of Management Information and Program Control
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

John F. Stolz, Office of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Modification - Hittman Mobile Radwaste Solidification System (Pre-operation only)

Description of Change

The Hittman Mobile Radwaste Solidification System uses cement to solidify all three types of waste in a pre shielded container. The shielding is sufficient to protect operating personnel while solidifying spent resin. A disposal liner with an internal mixer is used as the solidification container. The ratios of cement, additives (if required) and waste that will produce a dry product are determined through test solidification in a laboratory in accordance with the PCP.

Safety Evaluation Summary

The Hittman system will not adversely affect nuclear safety based on the following:

1. Stabilization with cement will increase the level of nuclear safety during transport and disposal.

2. A seismic spill containment is provided to collect spills up to 163 ft^3 (contents of a full solidification liner.

3. The venting of the gas space of a liner being filled does not constitute an unmonitored discharge because of the absence of a gaseous radioactive isotopes (at ambient temperature and pressure) in any of the waste streams to be processed. *

4. Filling and solidification controls are remote and separated by a 7-8 inch thick concrete shield thereby reducing the radiation hazard to operating personnel.

5. Hoses used for transfer processes which are periodically tested at system operating pressure and visually inspected prior to use.

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* This aspect is discussed at length in response to NRC Inspection Report 81-34.

Modification: B/A 412237 - Reactor Coolant Pump Motor Surge Capacitor Removal

Description of Change:

The new design removes the reactor coolant pump (RCP) motor surge capacitors from the motor circuit. The capacitors were originally installed inside each RCP motor terminal cabinet and connected line to ground at each phase. This previous design exposed the capacitor dielectric to a possible high radiation level within the containment building in the event of an accident. Studies have shown that a high radiation level promotes capacitor dielectric failure which electrically shorts the capacitor thus rendering the RCP motor inoperable.

Safety Evaluation Summary:

Removal of the surge capacitors from the RCP motor circuit increases the reliability of the RCP motor by eliminating radiation effect as a cause of motor circuit failure.

Since the operation of the RCP motor is not nuclear safety-related, removal of the surge capacitors does not impact the nuclear safety criteria.

There are two potential transformers within the RCP motor terminal box which are used for pump motor power monitoring. This monitoring is a safety related feature. In the process of removing the surge capacitors, the potential transformer leads were disconnected from the capacitor and reconnected to the phase stub bus. This does not change the potential transformer circuit performance and thus this modification leaves no unreviewed safety questions.

Modification: B/A 432017 - Relocate Feed Pump on Waste Evaporator

Description of Change:

The Reactor Coolant and Miscellaneous Waste Evaporator's feed pumps have been relocated to improve accessibility of the feed pumps. The system repiping also added additional suction and drain valves and connections for future feed pumps for both evaporators. The repiped sections of the system were heat traced to prevent precipitation of Boron. These modifications improve the maintainability and minimize the down-time of the evaporators.

In addition, isolation valves were added to the outlet lines of the distillate reservoir tank to provide positive isolation of the tank from the distillate pumps.

Safety Evaluation Summary:

The subject modification adds piping, valves and heat tracing which are equal or better than the existing equipment. The subject modification does not change system operation, design basis, margin of safety or description in the TMI-1 FSAR.

It is therefore concluded that the subject modification does not involve an unreviewed safety question.

Change: B/A 412072, High Pressure Injection System Cross-Connection (RM-14)

Description of Change:

Piping modifications inside the Reactor Building have cross-connected pairs of the High Pressure Injection (HPI) System injection legs. HPI leg A in train A is cross-connected to leg C in train B and leg B in train A is cross-connected to leg D in train B. In addition, a cavitating venturi has been installed in each HPI injection leg upstream of the HPI leg check valve nearest the Reactor Coolant System (RCS) cold leg and downstream of the cross-connection. The normal make-up line has been rerouted to inside the Reactor Building and connects to the "B" injection line downstream of the cross-connection and upstream of the HPI leg check valve nearest the RCS. An additional check valve between the cross-connect and the make-up line tie-in prevents make-up flow to the "D" injection nozzle via the cross-connection. A high capacity make-up valve MU-V-217 has been installed in a bypass line around the normal make-up valve MU-V-17. The make-up flow instrumentation has been modified to read the higher make-up flow rates.

Safety Evaluation Summary:

The design and installation of the modifications described above meet or exceed the requirements invoked on the original design and documented in the TMI-1 FSAR Chapter 6. The cross-connections and cavitating venturi improve plant safety by allowing mitigation of the effects of a small break Loss of Coolant Accident (LOCA) occuring in a HPI line at or near its connection to the reactor coolant system (RCS) or in a RCS cold leg without operator action. The HPI water losses from the broken leg are restricted to approximately one-fourth (1/4) of the total injection pump flow rate and the remaining flow is injected into the RCS via the unbroken HPI legs. The HPI flow through the unbroken legs satisfies small break Emergency Core Cooling System (ECCS) criteria. The larger capacity make-up valve provides a means of quickly restoring pressurizer level following overcooling of the reactor coolant system without starting another make-up pump or thermal shocking an RCS HPI nozzle not normally used for make-up.

Modification: B/A 412012 - MS-V22 A&B Spring Replacement (RM-13H)

Description of Project:

In order to provide over-pressure protection for the Turbine Driven EFW Pump, i.e., do not exceed the maximum operating pressure of 250 PSIG for the turbine steam chest and mechanical governor, the following modifications to the turbine main steam inlet line have been implemented:

1. New springs have been installed on safety valves MS-V22A&B whereby reducing their set pressures from 495 PSIG and 505 PSIG, respectively, to 200 PSIG and 220 PSIG respectively.

2. The set pressure for pressure controller PC-5 has been reduced from 200 PSIG to 175 PSIG.

3. The valve stem travel on main steam pressure control valve, MS-V6, has been limited to 65% of the full valve stem travel distance.

4. All three (3) turbine nozzle valves have been locked in the full open position whereby ensuring sufficient steam flow to the Turbine Driven EFW Pump.

Safety Evaluation Summary:

The modifications to the turbine main steam inlet line have been reviewed to ensure that they will provide the required over-pressure protection for the Turbine Driven EFW Pump. Futhermore, these modifications will not create adverse conditions which will degrade the EFW system performance. The change to open all three (3) inlet steam nozzle valves on the turbine will increase the nozzle flow above design flow, thus, ensuring turbine and pump performance. The changes that have been implemented by this modification improve the EFW system reliability and do not have any adverse effects on either the EFW system or other systems and components.

Modification: B/A 412012 - EFW Backup Instrument Air Compressor System (RM-13)

Description of Project:

A backup source of instrument air has been provided in order to mitigate the event when the main instrument air supply and station service air supply systems are both inoperable. Backup instrument air compressors and systems have been installed in the Turbine and Intermediate Buildings.

Intermediate Building

The backup instrument air supply system consists of an 80 gal reservoir, pressurized to ~100 PSIG, which is supplied from an 18SCFM air compressor, IA-P-2B. When the main instrument air system pressure drops to below 70 PSIG, the supply line from the backup air compressor is opened. Transfer to the backup air supply is automatic and no operator action is required. The power to the air compressor is supplied from the 1B 480V ES Motor Control Center. Equipment serviced by this system are key valves and instruments located in the EFW Main Feedwater and Main Steam systems.

Turbine Building

The backup instrument air supply system configuration for this building is similar to the system installed in the Intermediate Building. The backup air compressor (IA-P-2A), electrical power supply (1A 480V ES Motor Control Center), valves and instruments are different from the valves and instruments that are in the Intermediate Building. Valves and instruments serviced by this system are located in the condensate Main Feedwater and Main Steam Systems.

Safety Evaluation Summary:

With the installation of the backup air compressors and system, the ability of plant personnel to effect a controlled and orderly cooldown of TMI-1 will be increased even when the normal instrument air supply and station service air supply systems are rendered inoperable. The backup air supply will ensure that the main feedwater and emergency feedwater systems, as well as, the main steam dump lines to both the condenser and atmosphere remain operational.

Change Modification: Manual Control of PORV
B/A 412062; LM-39

Description of Change:

A key locked control switch to PORV (RC-RV2) has been added to allow the operator to manually open the valve from the control room under certain emergency conditions as called out in Plant procedures.

The switch employed by this modification is of the same quality as the existing equipment. The switch has two positions, normal & open and spring return to normal when released from open position. The switch is keylocked in the normal position and valve is closed at that position and is under automatic controls. The external wiring added by this modification from the relay cabinet to the control room are all in conduit and/or existing trays.

This modification performs the same functions in the same manner as the unmodified system. The only result of this modification to the PORV (RC-RV2) control system is to allow the operator to open manually the PORV from the control room.

Safety Evaluation:

It is determined with regard to the PORV (RC-RV2) Manual opening from the control room:

1) The probability or consequences of accidents previously evaluated have not not been increased. The modification described above is key-locked to prevent inadvertent actuation and the automatic functioning of the PORV is not changed.

2) No accidents other than previously considered will be introduced since the modification will not change the operation of the Safety Systems.

3) No safety margin has been reduced. The PORV control system will continue to operate as designed.

For reasons presented above, implementation of the design changes associated with the manual opening of PORV (RC-RV2) from the control room do not involve unreviewed safety considerations.

Modification: B/A No. B00221 - Raise steam generator level transmitters SPLT1, SPLT2 and SPLT4 (RM8).

Description of Change:

This change includes the raising of steam generator level transmitters SPLT1, SPLT2 and SPLT4, to 5'-9 3/4" above Reactor Building (RB) floor, from their existing lower level where they were vulnerable to high water level due to accidental flooding in the RB.

Safety Evaluation Summary:

Steam generator level is one of the key parameters used to accomplish safe shutdown and maintenance of natural circulation. The transmitters for these level instruments located at the lowest level of RB were vulnerable to accidental flooding in the RB. Moving these transmitters to 5'-9 3/4" elevation above RB floor puts them above predicted post LOCA flood level and hence upholds the integrity of the level instruments for a safe shutdown. This modification does not result in any change to the system as described in the FSAR.

Modification: B/A 412012 - EFW Safety Grade Auto Start on Loss of Four (4) Reactor Coolant Pumps or on Loss of Both Main Feedwater Pumps (RM-13E)

Description of Project:

In addition to the turbine-driven EFW pump, both of the motor-driven EFW pumps have been modified to receive an auto-start signal on loss of both MFW pumps or loss of four (4) RCS pumps. This modification has been implemented by utilizing contacts from the RCS power monitors or by sensing the differential pressure across the MFW pumps. The RCS pump power monitors are a safety grade system. The MFW pump differential pressure switches have been seismically tested and are tied into the respective EFW initiating circuits through isolation relays. The DP switches and circuitry are considered safety grade to the extent possible, i.e., qualified components installed in the turbine building which is a non-seismic Class I structure.

The actuation system is arranged in two (2) trains which utilize two pressure switches to sense the loss of both MFW pumps and four (4) contacts from each of the redundant RCS pump power monitors to sense the loss of all four (4) RCS pumps. The power for each actuation train is supplied by the "A" station battery/diesel generator and the "B" station battery/diesel generator. The "A" and "B" train sensors and cable have been located and routed to satisfy the safety grade separation criteria in Chapter 8 of the TMI-1 FSAR. The motor-driven EFW pumps are powered from the redundant class 1E 4160 volt power supply and have been block loaded on their respective diesel generator.

Safety Evaluation Summary:

The safety grade auto-start of all EFW pumps based on the above signals and block loading of the motor-driven EFW pumps on the diesel generators will ensure adequate flow to the steam generators. Furthermore, the ability of plant personnel to effect a controlled and orderly cooldown of TMI-1 will be increased, as well as the reliability of the Emergency Feedwater System.

Potential overcooling and overfill of the Steam Generators have been analysed and found acceptable (and have received NRC concurrence in NUREG 0680 p. C1-2 and NUREG 0680 Supplement 3 p. 15) See also RM-13C.

Change Modification: Rx. Bldg. Sump Level - Control Grade
B/A 412047; LM-8A

Description of Change:

This modification installed a level transmitter to measure the water level above the containment floor up to 10 feet. This instrumentation supplements the instrumentation in the sump and extends the range to monitor water levels from the bottom of the sump to 10 feet above the containment floor. The level indication for the new instrumentation is local outside the containment.

Safety Evaluation:

This modification does not change any of the safety analysis described in the FSAR, nor in any way increase the potential for any of the hypothetical accidents described therein. This modification extends the range of the instrumentation provided to monitor water levels in the containment due to large pipe breaks and thereby enhances the nuclear safety in plant operation.

Modification: B/A No. B00215 - Raise pressurizer level transmitters RCI-LT1, RCI-LT2 and RCI-LT3 (LM-9).

Description of Change:

This change includes the raising of pressurizer level transmitters RCI-LT1, RCI-LT2 and RCI-LT3 to 6'-1/2", 5'-11 3/4" and 5'-11 3/4" respectively above Reactor Building (RB) floor from their existing lower level where they were vulnerable to high water level due to accidental flooding in the RB.

Safety Evaluation Summary:

Pressurizer level is one of the key parameters used to accomplish safe shutdown and maintenance of natural circulation. The transmitters for these level instruments located at the lowest level of RB were vulnerable to accidental flooding in the RB. Moving these transmitters to these new elevations above RB floor puts them above predicted post LOCA flood level and hence upholds the integrity of the level instruments for a safe shutdown. This modification does not result in any change to the system as described in the FSAR.

Change Modification: Modification of Power Supply to ICS/NNI System. B/A #412029; RM-17.

Description of Change:

This modification installed a non-automatic transfer switch on elevation 322' of Inverter Room 1A, with manual pushbutton controls and annunciator located in the main control room. This manual transfer switch will provide a means for the operator to remotely connect the ICS/NNI supply bus to an alternate, regulated feed upon loss of input voltage to the bus from the Inverter.

Safety Evaluation:

This modification adds a remote switching capability of transferring an auxiliary supply to power the NNI/ICS system in the event the static automatic transfer switch 1A fails as a power feeder to the NNI/ICS system.

The transferred power supply does not power safety related equipment.

The components to be installed will be mounted to preclude seismically generated missiles.

The modification will impose an additional continuous load of 30VA on the emergency diesel generator system in the event of an off-site power failure. The added 30VA will not degrade the diesel generator system.

The implementation of the modification does not involve unreviewed safety considerations with regard to the criteria 10CFR, Part 50 Section 50.59 (a) (2) or degrade any safety related equipment.

Modification: B/A 412012 - Manual Loader Stations for EFW Control Valves EF-V30A/B (RM-13D).

Description of Project:

The manual loader stations for EFW control valves EF-V30A&B have been installed in the control room on the main control board at sections "CC" and "CL." The manual loader stations will provide the plant operators with the ability to control EFW flow independent of ICS, as well as upon loss of power to ICS.

Safety Evaluation Summary:

This modification provides redundancy to the controls for EFW control valves EF-V30A&B. It will preclude the loss of EFW flow control upon loss of power to the ICS. Furthermore, the ability of plant operators to effect a controlled and orderly cooldown of TMI-1 will be increased, as well as the reliability of the Emergency Feedwater System.

Modification: B/A 412024 - Installation of Emergency Feedwater Cavitating Venturis

Description of Project:

Cavitating venturis have been installed in the EFW system, i.e., in each of the EFW supply lines to the OTSG's (one (1) venturi per line) between the EFW control valves EF-V30A&B and the respective EFW containment penetration. The venturis have been installed in order to (1) limit EFW flow to a ruptured OTSG; (2) limit the mass and energy release from a ruptured OTSG and prevent overpressurization of the Containment Building; (3) ensure sufficient EFW flow to the intact OTSG; and (4) reduce the potential for overcooling of the Reactor Coolant System. The venturis have been sized to provide at least 500 GPM EFW flow for small break LOCA or feedwater line break accidents.

Safety Evaluation Summary:

This modification has been implemented in order to limit EFW flow to a ruptured OTSG. The cavitating venturis will improve the EFW system's ability to remove heat from the RCS in a controlled and orderly manner. This modification does not introduce any accidents or malfunctions not previously evaluated nor does it increase the likelihood of occurrence or consequences of any accidents. No safety margins will be reduced as a result of this modification.

Change Mod #1082: Install Alarm to Indicate Electromatic Relief Disabled

Description of Change: An alarm was added in the control room to alert the operator in the event the Power Operated Relief Valve (PORV) is blocked. The alarm occurs if the block valve is closed or if the NDTT switch is not in the desired position. This alarm is to warn the operator if over-pressure control of the PORV is not available.

Safety Evaluation Summary: Work was done in accordance with drawings provided by GAI, the design requirements are compatible with the original design.

The change affects alarm circuits only and therefore does not affect the electromatic relief valve operation.

For the above reasons the change does not:

1. increase the probability of an accident,
2. create a different type accident possibility or reduce the margin of safety as defined in the technical specifications, or
3. involve an unreviewed safety question.

Modification: B/A #412003 - Tailpipe Temperature Detector on PORV and Code Safety Valves.

Description of Change:

The PORV and Code Safety Valve tailpipe thermocouples were replaced with pairs of thermocouples forming differential detectors to monitor the discharge line temperatures relative to the local ambient temperature. The pipe thermocouple is mounted on each discharge line inside of the secondary shield wall. The ambient thermocouples are on brackets mounted on the secondary shield wall adjacent to its pipe thermocouple.

The differential temperature signals become part of the computer data base. A pressurizer relief/code safety valve discharge line differential temperature of 45 degrees Fahrenheit or more will result in the computer displaying that a pressurizer relief/code safety valve discharge line "high temperature" exists on the control room cathode ray tube (CRT). The CRT high temperature display clears when the differential temperature drops to 40 degrees Fahrenheit or less. The differential temperatures can be displayed in the control room on the CRT, printer or continuous recorder. The control room operator may confirm a valve has reseated by monitoring the tailpipe temperature trend following a valve opening versus the predicted trend for a reseated valve.

Safety Evaluation Summary:

The differential temperature indications are used by the control room operator to detect that a relief/safety valve has opened, verify that it has reseated or detect valve seat leakage under steady state conditions. The differential temperature indication function as confirmation to the control grade indication provided by differential pressure transmitters connected across elbow taps downstream of the relief/code safety valves. In addition, the relief valve will be monitored by accelerometers mounted on the valve. The differential temperature detectors do not affect safety related equipment and have been seismically installed to prevent missile hazards to safety related equipment.

It is therefore concluded that the subject modification does not involve an unreviewed safety concern.

Change Modification: Existing Air Operated RB-V-7 Valve Replacement by a Motor Operated Valve (PM-1, B/A 412060)

Description of Change:

The RB-V-7 Valve used in the normal reactor building cooling water system, serving to isolate the water line at the containment boundary, was replaced with a motor operated valve. The new RB-V-7 motor operated valve was powered from 480V E.S. MCC 1B. This valve was provided with status indication and control facility at the control room.

Safety Evaluation Summary:

This modification eliminates the poor leakage performance associated with the original valve. It is determined that the potential for the accident and its consequences as described in the FSAR are not increased by this change.

Modification: B/A 412012 - Main Steam Rupture Restraint to Protect EFW Pumps/Pipe (RM-13H)

Description of Project:

The main steam rupture restraint was installed to protect the discharge pipe of the turbine-driven EFW pump (EF-P1) from a potential MSLB and pipe whip accident, as created by the 12" MS inlet line to the turbine-driven EFW pump. The rupture restraint protects the EF-P1 pump discharge pipe between check valve EF-V13 and the common EFW system discharge manifold at valves EF-V2A&B, i.e., in cubicle 1J of the Intermediate Building on EL 295'-0".

Safety Evaluation Summary:

This modification protects the EFW system discharge pipe/common manifold from a MSLB and pipe whip accident. It ensures that no single failure (MSLB) can disable the EFW system with no operator action required. The changes that have been implemented by this modification improve the EFW system reliability and do not have any adverse effects on either the EFW system or other systems/components.

Change Modification: Provide Auto Reset for ESF Actuation System
B/A 412058; LM-33

Description of Change:

This modification provides manual reset capabilities in the control room for ESFAS bistables in HPI, LPI and reactor building isolation system channels (all safety related) for TMI-1. The ESFAS trip bistables were adjusted for zero deadband, thereby allowing reset when the monitored variable goes above its predetermined set point. The ESFAS 2 of 3 logic actuation circuits were modified with a relay contact and remote reset switch located on the control room console which will allow the operator to reset the ESFAS trip channel without leaving the control room.

Safety Evaluation:

The protection functions of the ESAS system continue to function as designed and are not degraded by these modifications. Remote reset of the ESFAS functions shall add to the plant control performance during emergency conditions when the operator is required to attend to many control systems at the main console. Further, it is concluded that:

1) The probability or consequences of accident, previously evaluated, have not been increased. The modification described above will not have any effect on the operation of the ESFAS. Accordingly, the consequences of accidents for which ESFAS is required to be operable will not change.

2) No accidents other than previously considered will be introduced. Since the modification will not change the operation of ESFAS, no new accident conditions will occur as a result of the ESFAS operation.

3) No safety margins have been reduced. The ESFAS will continue to operate as designed to mitigate the consequences of Loca's within established safety margins.

For the reasons presented above, implementation of the design changes associated with the remote-reset capabilities of the bistables for all the ESFAS channels do not involve unreviewed safety considerations with regard to the criteria of 10 CFR, Part 50 Section 50.59 (a) (2).

Modification: B/A 412012 - EFW Flow Indication in the Control Room
(RM-13B)

Description of Project:

Redundant EFW flow indicators have been installed on the main control console in the control room and will provide the plant operators with positive EFW flow indication for monitoring OTSG level. This safety grade modification is supplied with Class 1E power and satisfies both single failure and Seismic Category I criteria.

Two independent and redundant equipment trains provide assurance that at least one (1) flow indicator per OTSG is available to the plant operators. Each of the EFW supply lines has been provided with two (2) flow sensing devices, i.e., a pair of ultrasonic transducers, display computers and flow indicators. The ultrasonic transducers have been installed between the EFW control valves EF-V30A&B and the EFW containment penetrations. The 'Red' ultrasonic transducers FE-777 and FE-789 and corresponding transmitters FT-778 and FT-790 provide input signals to control console indicators FI-779 and FI-791. Similarly, the 'Green' ultrasonic transducers FE-780 and FE-786 and corresponding transmitters FT-781 and FT-787 provide input signals to control console indicators FI-782 and FI-788.

Safety Evaluation Summary:

This safety grade modification is designed to provide only positive EFW flow indication in the main control room and no control function. The plant operators will use the OTSG level indication for controlling EFW flow. The changes that have been implemented by this modification improve the EFW system reliability, do not create the possibility for an accident or malfunction, and do not have any adverse effects on the EFW system or other systems/components. Furthermore, the loading of instrumentation will not degrade the operability of the diesel generators.

Modification: B/A 412012 - EF-V30A&B Fail Open on Loss of Instrument Air (RM-13C)

Description of Change:

In order to provide assurance that emergency feedwater can be delivered to each steam generator when required, the failure mode of control valves EF-V30A&B have been modified to fail in the open position and remain in this position upon loss of air. EF-V30A&B are air cylinder actuated control valves which are located in the parallel emergency feedwater lines. The fail open position for EF-V30A&B has been accomplished by installing a separate Fisher trip accessary mounting package which includes a 75% trip valve and 1322 cu. in. capacity air accumulator on each control valve.

Safety Evaluation Summary:

The purpose of this modification is to have the EF-V30A&B control valves fail safe, i.e., in the open position. This modification does not effect the margin of safety as defined in the basis of Tech Spec 3.4. Futhermore, the ability of plant personnel to effect a controlled and orderly cooldown of TMI-1 will be increased, as well as, the reliability of the Emergency Feedwater System.

Ten minutes at a minimum is available for operator action to prevent overcooling in the extremely unlikely event of multiple failures resulting in the EFW control valves failing open. In addition, overcooling as a result of anticipatory filling of the Steam Generators from 30" to 50% has been analyzed and found to be acceptable (fill rate is limited by procedures as well as the newly installed EFW cavitating ventures - B/A 412024). See also RM-13E.