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May 10, 1983
5211-83-099

Mr. 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 and procedures as described in the FSAR. There were no tests or experiments performed not described in the FSAR.

Sincerely,

H D Hukill

H. D. Hukill
Director, TMI-1

HDH:RAS: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

IE 24

Modification: B/A 412074 - Limit Reactor Building Purge and Vent Valve Angular Opening to 30° (PM-3)

Description of Project:

The Reactor Building Purge and Vent Valves AH-V1A/B/C/D have been restricted to 30° open to ensure that the valve operators can close against the fluid dynamic forces and pressure differential that are created during a LOCA condition. Air-operated valves AH-V1A&D have been restricted by installing mechnical travel stops (bolts) in the cylinder head of each air-operator. The stops prevent the piston from traveling full stroke, i.e., only to the 30° maximum open and have been locked in position. Motor-operated valves AH-V1B&C have been restricted by resetting the Limitorque operator rotor switches to a trip position that corresponds to the 30° maximum open. The new setting will limit the valve opening via action of the limit switch contacts within the valve operator control circuit. No wiring changes have been added and/or deleted. Limit switches which signal the valve "open" position in the control room have been reset to indicate the valve "open" position at the 30° opening. Furthermore, a new air flow sensor and recorder will be installed in the ductwork and control room, and will measure the throttled flow rate during purging/venting operations.

NOTE: This modification is complete. We have subsequently committed to close the valves, however, we have not received a response from the NRC.

Safety Evaluation Summary:

This modification has been implemented so that the torque output to stroke the valves closed from the 30° angle under high differential pressure and flow conditions is within the capability of both the Bettis and Limitorque valve operators. The signals to open/close the valves have not been modified and the ES closure signal remains the same, i.e., as modified via the Containment Isolation modification. Purging/venting of the Containment Building for reducing airborne activity may still be accomplished, however, at a reduced flow rate. During shutdown, the valves may be opened to the full open 90° position. No safety margins will be reduced as a result of this modification.

Modification: B/A 412045 - BWST Level Indication (LM-4A)

Description of Project:

This modification added Barton differential pressure switch to the BWST to provide an additional low level alarm.

Following an accident, the initial injection of water by the decay heat removal system involved pumping water from the borated water storage tank (BWST) into the reactor vessel. With all engineered safeguards pumps operating, and assuming the maximum break size, this mode of operation lasts for a minimum of about 25 minutes. When the BWST reaches a low level, an alarm will be annunciated in the control room. At this time the operator will take action to open the suction valves (DH-V6A/B) from the reactor building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the reactor building sump. The BWST low level alarms, one from each BWST level transmitters, will be supplemented by a low level alarm from the new Barton differencial pressure switch. This switch will provide additional notification that the suction valves to the R.B. sump must be opened, thus removing a potential conflict if one of the existing transmitters indicates an erroneous high level.

Safety Evaluation:

This change provides for an additional alarm for BWST low level to the operator. This alarm will serve to confirm either of the existing low level alarms for the contingency of mis-operation of one of the alarms. Since this change does not adversely affect Nuclear Safety, it will serve to enhance Nuclear Safety; this change does not constitute an Unreviewed Safety Question.

Modification: B/A 412053 - Audio Monitoring System for the Main Steam System Code Safety Valves (Task RM-6)

Description of Project:

The audio monitoring system is an acoustic system that monitors noise resulting from high velocity steam flow when the main steam system code safety valve(s) lift when relieving steam flow. The noise information is transmitted by microphone to the control room to indicate the lift of the valve(s) associated with the specific steam generator (A or B) and the main steam line to the emergency feed pump turbine.

The safety valves are in four compartment areas located within the Intermediate Building. Each compartment comprises 1 to 6 safety valves. In each compartment a microphone is provided in close proximity to the safety valves for monitoring the noise of the relieving steam flow from the valves(s).

An audio processor cabinet is provided when a signal from a microphone or channel pick-up is converted in order to provide; input to data logger, and signal to the audio monitoring cabinet in the control room. The audio monitoring cabinet includes a speaker, channel selector switch and indicating lights which provide the operator with information concerning valve position indication for 'open' and 'close' conditions.

An uninterruptible power supply is provided as the power source to the monitoring system.

Safety Evaluation Summary

This modification has been implemented so that the control operator can monitor the lifting of the main steam safety code valve(s) in the event the main steam pressure is exceeded in both the A and B steam generator system. The affected steam generator system which the operator can identify from the data logger printout or audio channel indicating light. The operator acts per plant procedures to mitigate the event to preclude cooling of the RCS.

This modification is considered to be non-safety related. Therefore, single failure criteria or seismic conditions are not part of the design modification. However, to ensure the electrical functioning of the monitoring system the power source is taken from a vital AC source (inverter).

The probability or consequences of an accident have not increased.

The modification will not change the operation of the steam generator (A or B) systems. Therefore no new accident conditions will occur as a result of this change modification.

No safety margins have been reduced.

Accordingly, the implementation of this modification does not involve unreviewed safety considerations with regard to the criteria of 10CFR Part 50 Section 50.59 (a) (2).

Modification: B/A 412021 - Reactor Coolant System
High Point Venting System
Pressurizer Vent Only (LM-21A)

Description of Project:

The pressurizer vent line has been installed in order to improve the plant's ability to vent a mixture of reactor coolant liquid/steam and/or non-condensible gases from the Reactor Coolant System, without having an adverse impact on core cooling. This safety grade modification satisfies Seismic Class I criteria and is supplied with Class 1E electrical and instrumentation power.

The pressurizer vent line is controlled by motor-operated isolation valve RC-V28 and solenoid-operated isolation valve RC-V44, which are mounted in series, and vents initially to the Reactor Coolant Drain Tank and ultimately to the Reactor Building atmosphere. Flow element FE-1079 and differential pressure transmitter DPT-1079 provide an input signal to a "flow/no flow" lamp installed on panel "PC". The vent valves are activated from the Main Control Room "PC" panel which includes open/closed key lock switches and indicating lights. Annunciation of inadvertent valve opening has been provided on light box "G" in the Main Control Room. The Pressurizer vent line will be maintained and operated via strict administrative controls.

Safety Evaluation Summary:

The purpose of this modification is to provide remote power operation of the vent line from the Pressurizer to the Reactor Coolant Drain Tank. This modification makes no changes in the procedure as described in the Safety Analysis Report. Administrative procedures will be implemented for controlling the operation of the pressurizer vent line from remote controls in the Main Control Room. This modification does not create the possibility of an accident or malfunction different from any previously evaluated in the SAR, i.e., failure of the PORV which is already an evaluated accident. No safety margins have been reduced as a result of this modification. Strict administrative controls and key lock switches will govern actuation of the pressurizer vent line.

CHANGE MODIFICATION: B/A 412028 - ICS/NNI Loss of Power Critical Plant Parameters Independent of ICS/NNI (LM-43C)

DESCRIPTION OF CHANGE:

The function of the Safety Critical Plant Variables in the control room is to provide the operator with indication of vital information in the event that all power internal to the NNI/ICS is lost. This design is designated as Nuclear Safety Related, Class 1E. (Digital indicators are not yet qualified to Class 1E and they will be replaced with qualified units at a later date, if available).

The indicators are located on the "PCL" back panel (except Pressurizer and Makeup Tank Level which are on main console "CC".) The signal source is from the Safety Grade Remote Shutdown Signal Conditioning cabinets (Task LM-38) via isolated outputs.

The existing NNI/ICS indicators and controls remain. The new digital indicators do not share power or signals with the existing equipment.

The critical plant variables are:

OTSG Loop A/B
RC Pressure (wide) loop B
Makeup Tank Level
Pressurizer Level
RC Hot Leg Temperature Loop A/B
RC Cold Leg Temperature Loop A/B

SAFETY EVALUATION:

This modification provides the control room operators with an alternate source with vital information so that corrective action can be taken to effect a controlled and orderly shutdown of TMI-1, if necessary.

Therefore, this modification does not increase the probability or occurrence or consequence of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis.

CHANGE MODIFICATION: B/A 412013 - Iodine and Particulate Sampling (LM-25B)

DESCRIPTION OF CHANGE:

Three (3) new sampling stations have been installed. Each station provides iodine and particulate samples of high and low radiation levels, and also, provide samples of noble gas. Two of the stations are located in the special building located on top of the concrete exhaust duct. Of these 2 stations, one samples the auxiliary building ventilation exhaust and the other the reactor building ventilation exhaust. The third station is associated with radiation monitor RM-A5 and provides samples of the condenser vacuum pump discharge.

The post accident iodine and particulate monitoring is accomplished passing a flowing stream representative of the effluent discharge stream through silver Zeolite cartridges with particulate filters. The range of measurement for the sample system as defined in NUREG 0737 is 10^{+2} Ci/cc. After sampling the cartridges will be removed and counted in a laboratory. Each of the three previously mentioned effluent discharge paths have a post accident iodine and particulate sample system. These sample systems are normally idle and are energized by particulate, iodine and gas monitors RM-A5, A8 or A9 when these monitors detect high radiation.

SAFETY EVALUATION:

The addition of the post accident iodine and particulate monitors in no way adversely affects plant safety. The probability or occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report has not been increased. The ability of the effluent radiation monitoring system to perform its function under accident conditions has been enhanced.

MODIFICATION: B/A 412013 - High Range Containment Monitoring (LM-23)

DESCRIPTION OF PROJECT:

Section 2.1.8b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July, 1979, contains NRC recommendations on installations of in-containment radiation level monitors. The guidance on in-containment radiation level monitors is as follows:

In-containment radiation level monitors with a maximum range of 10^8 rad/hour total radiation shall be installed. Alternately, a photon only radiation measurement with a maximum range of 10^7 rad/hr may be utilized.

In response to the above recommendations, two Victoreen Model 877 radiation monitor detectors have been installed to monitor containment radiation levels during and following a postulated accident.

The detectors have a range of 10 rad/hr. to 10^7 rad/hr. photon measurement only.

The high range radiation detectors and high range area monitors meet the design and quality assurance requirements for a Nuclear Safety Related System. They meet the single failure and testability criteria of IEEE 279-1971. Separation of redundant circuits is maintained in accordance with Section 8 of the FSAR. All components except readout devices and recorders are qualified to IEEE Standard 344-1975 "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The readout devices and recorders are mounted on the PRF Panel. The electrical installation is designated "Nuclear Safety Related" to include the detectors and their associated electrical cabling. The detectors are qualified in accordance with the requirements of IEEE-323 1974 for operation in the containment environment for normal and post accident environment.

SAFETY EVALUATION:

The system design does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. The use of the high range radiation monitor will assist plant operators to assess plant conditions during and following an accident.

No accidents or malfunctions of a different type than that which was previously evaluated in the safety analysis report, will be introduced. The design and installation features of the radiation monitor are designed so as to preclude the compromising of containment integrity or other safety features.

No safety margins as defined in the basis for any Technical Specifications have been reduced. The radiation monitor provides continuing in-containment post-LOCA radiation assessment and is not involved in consideration of any safety margin.

Based upon the above, we conclude that plant modification needed for installation of the high range radiation monitors do not involve any unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59 (a)(2).

CHANGE MODIFICATION: B/A No. 412044 - Saturation Margin Monitor (LM-I)

DESCRIPTION OF CHANGE:

This modification provides all necessary instrumentation for detection of inadequate core cooling by operators in Control Room. The Reactor Coolant System Saturation margin monitor provides direct on-line reactor coolant system margin to saturation indication. This is accomplished in each reactor coolant loop by means of continuous computation, display and comparison to the limiting saturation condition for the reactor coolant system. These computations are based on existing reactor coolant system Non-Nuclear Instrumentation (NNI) pressure and temperature signals. Continuous digital display of Reactor Coolant System Saturation Margin (RCSSM) is provided to the plant operator in terms of ^{o}F margin in the Control Room. An audible and visual alarm actuates when the computed saturation margin is less than the limiting value. The RCSSM provides saturation margin and alarm status output suitable for trending and computer input.

SAFETY EVALUATION:

This modification is to provide the control room operator with information to assist in identifying inadequate core cooling conditions. The saturation margin monitor will display, in the control room, the margin between the actual reactor coolant system temperature (Th) and the saturation temperature (Tsat) for the existing reactor coolant system pressure. An alarm willbe initiated if the margin should decrease below a preset value. The signal conditioning equipment and cabinets are seismically qualified and powered from IE buses. The digital panel meters though not seismically qualified are the best quality industrial grade available. Adequate redundancy and diversity exist to ensure that Tsat margin will be available to operator in the Control Room of TMI-I. This modification does not degrade the integrity of any safety-related system or any existing instruments which are required for safe and reliable operation and shutdown of the plant.

Modification: B/A B00225 - Second Level Undervoltage Protection (RM-22)

Completion Date:

Description of Project:

NRC letters dated June 3, 1977, August 8, 1977 and May 2, 1979 on failure of Class 1E equipment to perform their required function due to offsite electric grid system voltage degradation, require a second level of undervoltage protection to protect against such an incident. This modification has been designed on the basis of NRC's 'Safety Evaluation and Statement of Staff Positions, Relative to the Emergency Power Systems' as it pertains to second level undervoltage protection and which was enclosure 1 of June 3, 1977 letter NRC to J. G. Herbein.

On safety related 4160V buses 1D and 1E, there is an addition of six solidstate undervoltage relays, which includes replacement of existing three electromechanical undervoltage relays, one overvoltage relay and associated test switches and time delay relays. Also, it modifies panel wiring to provide re-required logic and alarm.

Safety Evaluation Summary:

This modification provides protection of safety loads from sustained degraded voltage from the offsite electric grid system without causing voltages in excess of maximum voltage ratings of safety loads and without causing spurious separation of the safety buses from offsite power.

When a degraded voltage condition is detected on a 4160V safety bus, that bus will be tripped and the associated diesel generator started. The diesel generator then provides the capability to adequately power safety loads on that bus. The changes that have been implemented by this modification do not create the possibility for an accident or malfunction, and do not have any adverse effects on any system/components.

Modification: B/A 412012 - OTSG Level Indication in the Control Room Independent of ICS (RM-13J)

Completion Date: February 4, 1982

Description of Project:

The OTSG level indication independent of ICS has been installed on the main control console in the control room and assures that the plant operators have proper indication to control OTSG level. This modification is redundant, provides full range (0-640" of water) level indication, and satisfies single failure criteria.

Two independent and redundant equipment trains ('Red and Green') provide assurance that at least one (1) display unit per OTSG is available to the plant operators. The Remote Shutdown Panel safety grade level transmitters and signal processing equipment provide an input source for the level indication signals. The 'Red' level transmitters, LT-775 and LT-788, on OTSG's (RC-H-1A/B) and corresponding isolation modules provide signals to control console indicators LI-775B and LI-788B. Similarly, the 'Green' level transmitters, LT-776 and LT-789, and corresponding isolation modules provide signals to control console indicators LI-776B and LI-789B.

Safety Evaluation Summary:

This safety grade, Seismic Class I modification, provides OTSG water level indication (full range) in the control room independent of ICS. Redundant indicators have been installed on the control console for each OTSG. This modification provides only a display function and does not interface with any existing control and/or protection systems. Therefore, this modification does not degrade any existing safety margins. The load requirements for the additional instrumentation have been reviewed and the impact on the diesel generator loading is acceptable.

Modification: B/A 412008 - Decay Heat and Core Flood Check Valves Leak Monitoring System (LM-42)

Description of Change:

This test system has been installed in order to determine the leak rate across isolation check valves DH-V22, CF-V4 and CF-V5, and to preclude over-pressurization and rupture of the low pressure decay heat removal system piping outside of containment. The test system consists of commercial grade tubing up to the points where it ties into the DH and CF system piping. At the tie-in points, the pipe and fittings are nuclear class to maintain the existing system integrity and pipe classification. All portions of this modification are designed to Seismic Class I requirements.

Safety Evaluation Summary:

This modification covers the installation of a test system for the purpose of performing leak rate tests of the DH and CF check valves. The test system is a commercial grade system which has no safety related function during plant operation and does not alter any safety functions nor reduce the margin of safety as defined in the FSAR station technical specifications. The system is disconnected from the CF and DH system piping during normal plant operation, and is seismically supported to avoid potential interaction with safety systems, RCS pressure boundary isolation is maintained via existing double isolation valves or by an isolation valve and blank flange arrangement. The entire system is located within the Reactor Building.

Change Modification: RB Spray System D/P Instrumentation Modification
B/A 412073; LM-7A

Description of Change:

This modification adds a differential pressure instrument between the BWST and SHST to provide a simple and direct means of measuring the fluid level differential between the two tanks, which will aid in maintaining the differential within prescribed limits.

Safety Evaluation:

This is an Important to Safety instrument only from a pressure boundary integrity standpoint. It is not required to function in order to achieve the RBSS safety functions. The DHRS and RBSS pressure boundary integrity is assured by the "Important to Safety" classification and quality assurance pertaining to it, and the seismic class S-I design criterion. The addition of a d/p instrument 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.

Modification: B/A 412039 - Tripping of loads on engineered safeguards buses (LM-32)

Completion Date:

Description of Project:

TMI-1 Licensee Event Report (LER) 80-01/01T reported a potential overload condition on the 1P480V bus during loss of 1S480V bus. Subsequently, Technical Data Report 185 Rev. 0 confirmed the overloading of 1P and 1S buses when 1S and 1P were not available respectively.

Following six loads will trip automatically, due to this modification, on ES signal.

I) Spent Fuel Cooling Pump Air Unit AH-E-8A
II) Penetration Cooling Fan AH-E-9A
III) & IV) Spent Fuel Pit Cooling Pumps, SF-P-1A & 1B
V) Boric Acid Mix Tank Heater
VI) Boric Acid Tank Mixer

The automatic tripping of selected loads along with administratively controlled manually applied loads e.g. 480V power receptacles will bring the 480V buses 1P and 1S within their design limits.

Struthers-Dunn auxiliary relays and indicating lights have been added to the IAES and IBES motor control centers. Cabling runs between ES actuation A cabinet 4 and IAES MCC and ES actuation B cabinet 5 and 1B ES MCC. Necessary coordination curves have been developed and the setting of the incoming breaker for the 1B ES MCC has been increased corresponding to its load.

Safety Evaluation Summary:

This modification restores the integrity of the safety related buses under all modes of plant operation as is required per general design criteria 17 of 10CFR 50 Appendix A.

Loads that trip automatically can safely be tripped without any design concerns. In the event both spent fuel cooling chains were out of service simultaneously, the heat capacity of the water contained in both pools is such that approximately 25 hours would elapse before the water in them would heat up to an excessive temperature. Abnormal spent fuel pool water temperature is presently alarmed in the control room (per FSAR article 9.4.7). GAI letter #GAI/TMI-1/ICS/3354 W. A. Steidle and R.M. Rogers to D.G. Slear on "Loss of Penetration System" dated 6/17/1980, indicates that restart of this system can be delayed six hours without design concerns.

The changes that have been implemented by this modification do not create the possibility for an accident or malfunction, and do not have any adverse effects on any system/components.

1. RCS Overcooling Guidance

PCR No.	Procedure	Procedure Title
1-OS-82-0007	1202-2	Station Blackout
1-OS-82-0008	1202-2A	Station Blackout With Loss of Both Diesel Generators
1-OS-82-0010	1202-4	Reactor Trip
1-OS-82-0011	1202-5	OTSG Tube Leak-Rupture
1-OS-82-0004	1202-6A	Loss of RC/RCP Within Capability of Makeup Sys (RC Pressure Above ESAS Set Point)
1-OS-82-0005	1202-6B	Loss of RC/RC Pressure (Small Breakloca.) Causing Automatic High Pressure Injection
1-OS-82-0006	1202-6C	Loss of RC/RCP Causing Auto HP Inj. Core Flood & LP Inj.
1-OS-82-0009	1202-14	Loss of Reactor Coolant Flow/RCP Trip
1-OS-82-0012	1202-26A	Loss of Steam Generator Feed to Both OTSG's
1-OS-82-0013	1202-26B	Loss of Feed to One Steam Generator
1-OS-82-0014	1202-36A	Loss of Instr. Air-Backup Air Available
1-OS-82-0015	1202-36B	Loss of Instr. Air-No Backup Air Available
1-OS-82-0016	1202-39	Inadequate Core Cooling (No Local)
1-OS-82-0017	1203-24	Steam Line Rupture Detection System(SLRD) Actuation

These changes improve the margin of safety since they improve the operator guidance on plant response and plant operation within the expected operating limits. Potential for challenges to the plant safety limits and limiting safety system settings are thus reduced.

The RCS overcooling guidance helps to ensure that the heat balance between Reactor Heat Generation is equal to the RCS Heat Rejection though the OTSG's, thus assuring adequate decay heat removal without overcooling the RCS.

The HPI throttling criteria assures that sufficient RCS inventory control is established under abnormal plant conditions to maintain maximum decay heat removal without damage to either the HPI pumps or the Reactor Vessel.

2. Loss of OTSG's as a Heat Sink

1-OS-82-0493	1202-39	Inadequate Core Cooling (No Local)

This change provides specific operator guidance for conducting a safe plant cooldown without using the OTSG's as a heat sink. Since a total and extended loss of the OTSG's as a heat sink was not previously addressed in the emergency procedure, this change improves the margin of safety by providing guidance for this contingency.

3. Failure of Feedwater and PORV

1-OS-82-0495 1202-29 Pressurizer Systems Failure

This change improves the operator guidance for plant emergencies in which a coincident failure of all OTSG feedwater and failure of RC-RV-2 (PORV) were to occur. By alerting the operator to proceed to the Inadequate Core Cooling Procedure (EP 1202-39 under the above circumstances, the operator will maximize the available core cooling methods. The safety margin is improved by this change, since it provides guidance for a contingency which previously was not provided in this Emergency Procedure.

4. Reducing Challenge to MSRV's

1-OS-82-0694 1202-5 OTSG Tube Leak-Rupture

These changes provide more guidance for taking the turbine off line and shutting down the reactor while reducing the potential challenge to the main steam safeties. The margin of safety is not reduced since the change minimizes the main steam safety valve discharges assumed in the F.S.A.R.

5. Condensate Inventory

1-OS-82-0791 1202-26A Loss of Steam Generator Feed to Both OTSG's

This change improves the margin of safety since it more accurately reflects the available condensate inventory within the condensate storage tanks when the low-low level alarm is actuated.

6. Pressurizer Heater Power Source

1-OS-82-0470 1202-29 Pressurizer Systems Failure

This change improves the margin of safety since it provides additional power sources to the pressurizer heaters, thus increasing the ability of the operator to maintain the plant subcooled.