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July 5, 1984
5211-84-2127

Mr. R. C. Haynes
Region I, Regional Administrator
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA. 19406

Dear Mr. Haynes:

Three Mile Island Nuclear Station, Unit I (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.

Sincerely,

H. D. Hukill,
Director, TMI-1

HDH/RAS/mle

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

IE01
1/1
IE24

B/A 412021 - Reactor Coolant System
High Point Venting System
Reactor Vessel Head Vent

DESCRIPTION OF PROJECT:

The Reactor Vessel Head Vent has been installed in order to improve the plant's ability to vent a mixture of reactor coolant liquid/steam and/or non-condensable 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 Reactor Vessel Head Vent is controlled by solenoid-operated isolation valves RC-V 42 & 43 which are mounted in series and vents directly to the Reactor Building atmosphere. Flow element FE 1081 and differential pressure transmitter DPT-1081 provide input signals to the "flow/no flow" lamps 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 in light box "G" in the Main Control Room. The Reactor Vessel Head Vent will be maintained and operated via strict administrative controls.

SAFETY EVALUATION SUMMARY:

The purpose of this modification is to provide a remote power operation of the vent line from the Reactor Vessel Head. Administrative procedures will be implemented for controlling the operation of the Reactor Vessel Head Vent 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. The probability of an inadvertent venting of the Reactor Vessel Head Vent is slightly increased, since remote power operation of the double isolation valves is controlled from the Main Control Room; however, strict administrative controls and key lock switches will govern actuation of the Head Vent line.

CHANGE MODIFICATION:

Screenhouse Wall/Emergency Lighting Upgrade B/A 412347

DESCRIPTION OF CHANGE:

This modification adds a UL-A-Labeled 3 hour fire rated roll-up door, 3-hour fire rated penetration seals in the Screenhouse wall, an 8-hour emergency light and a communication handset in the intake screen and pumphouse between fire zones ISPH-FZ-1 and ISPH-F2-Z containing redundant equipment needed to bring the plant to safe shutdown. In addition, an existing 8-hour emergency light and handset will be relocated so as not to interfere with the operations of the door.

This modification also includes the installation of an 8-hour emergency light in the Intermediate Building floor elev. 295'-0", near valve RRV-2.

These modifications were identified in the Safe Shutdown Evaluation Report, "TMI-1 Fire Hazards Analysis and Appendix R, Section III, G" dated June 28, 1982 as needed to eleminate non-compliance with 10 CFR 50 Appendix R.

SAFETY EVALUATION:

The important Safety Functions enhanced by these modifications are that, within the intake screen and pumphouse a single fire in one fire zone could damage a nuclear service river water pump and the redundant nuclear service pump. The addition of a 3-hour fire rated roll-up door, and the 3-hour penetration seals will separate these two pumps and will have no impact on the functional operation of the pumps and associated components.

The installation of the emergency lights in the Intermediate Building and Intake Screen and Pumphouse will have no impact on the functional operation of the plant.

The addition of the roll-up door and the 3-hour penetration seals between fire zones ISPH-FZ-1 and ISPH-FZ-2 will enhance the safety function by separating redundant equipment needed to bring the plant to safe shutdown. This door will remain open at all times. It will have an electro thermal link which will be interlocked with the ionization detection system of both fire zones. In case of a fire, the roll-up door will close and separate fire zones ISPH-FZ1 and ISPH-FZ-2.

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

CHANGE MODIFICATION:

B/A 412375 Turbine Building Contamination Detection and Control

DESCRIPTION OF CHANGE:

This modification upgraded existing systems/components and enhanced utilization methods to provide more accurate and timely OTSG leak detection techniques and monitoring (for accountability). New instruments were added to provide redundancy and relief from manual sampling which was utilized to achieve these functions and included portable steam line (NaI), area and airborne radiation monitors, turbine building sump liquid effluent monitor and composite sampler proportional to flow.

SAFETY EVALUATION:

The Liquid Radiation Monitoring System (LRM) for the Turbine Building sump functions continuously during all plant operating modes once the equipment is energized. If a high radiation setpoint is detected the sump pump is tripped off and an alarm is annunciated in the control room. A plant operator must go to the LRM control panel to investigate and resolve the problem. If high radiation is confirmed by the control panel meters and/or analog recorder, operations will be required to either re-align the sump discharge valves from the IWTS to the Unit #2 Condensate Storage tank*, or increase the dilution factor (by increasing MDCT flow and decreasing the release rate from the T.B. Sump) such that the activity level will be below 10 CFR 20 MPC limits at the discharge point. The keylock normal/override switch on the LRM control panel must be placed in 'override' to remove the trip function and restart the sump pumps after valves re-alignment.

The LRM system performs a monitoring function of alarm annunciation and pump trip and no other control functions. The equipment can not cause a casualty event. It assists operations to assure that plant discharge liquid and filtercake will not exceed 10 CFR 20 maximum permissible concentrations.

The LRM is classified as Important to Safety, non-seismic and non-1E.

In Conclusion:

1. The probability of occurrence or consequence of an accident previously evaluated have not been increased.
2. No accident other than those previously considered will be introduced.
3. No safety margins have been reduced.

* This tank COT-1B has been transferred to TMI-1 operations.

B/A 412021 - Reactor Coolant System
High Point Venting System
RCS Loop A & B Hot Leg Vents

DESCRIPTION OF PROJECT:

The RCS Loop A & B Hot Leg Vents have 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. This safety grade modification satisfies Seismic Class I criteria and is supplied with Class 1E electrical and instrumentation power.

The RCS Loop A & B Hot Leg Vents are controlled by solenoid-operated isolation valves RC-V 40A and 41A and RC-V 40B and 41B which are mounted in series and vent directly to the Reactor Building atmosphere. Flow elements FE 1080 and 1082 and differential pressure transmitters DPT-1080 and 1082 provide an input signal to the "flow/no flow" lamps 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 RCS Loop A & B Hot Leg Vents 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 lines from the high points in the RCS Loop A & B Hot Leg piping. Administrative procedures will be implemented for controlling the operation of the RCS Loop A & B Hot Leg Vents 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. The probability of an inadvertent venting of these vents is slightly increased, since remote power operation of the double isolation valves is controlled from the Main Control Room; however, strict administrative controls and key lock switches will govern actuation of the isolation valves.

CHANGE MODIFICATION: H_2O_2 Waste Gas Analyzer
B/A 412215

DESCRIPTION OF CHANGE:

The Hays Gas Analyzer was the only hydrogen/oxygen monitor installed in the WDG System. This project added the second analyzer (Beckman).

The Beckman Analyzer added by this project will alleviate the maintenance and operability problems with the existing Hays Gas Analyzer.

SAFETY EVALUATION:

The addition of a second H_2O_2 Analyzer will allow the Waste Gas Holdup System to remain operational when existing Hays Gas Analyzer is out of service for maintenance.

Therefore, the addition of the Beckman Analyzer does not increase the probability of occurrence or the consequence of an accident or malfunction of equipment and increase the overall plant availability.

CHANGE MODIFICATION:

B/A 412052 - Containment Isolation - Line Break Detection (RM-5D)

DESCRIPTION OF CHANGE:

The 4 psig reactor building pressure signal was deleted from several valves to maintain services to the RC Pumps and motors. The systems involved are closed systems thus preventing a leak path to outside containment. Automatic isolation in the event of line break has been added to preclude loss of containment integrity in the event the closed system is damaged.

Containment Isolation on NSCC and ICC system pipe line break is provided by the addition of redundant IE safety grade actuation trains. The trains are comprised of level transmitters which monitor NSCC and ICC surge tank levels, signal conditioning for indication, detection and alarming tank leakage (pipe line break) and provide an isolation signal to valves IC-V2, IC-V3, IC-V4, IC-V6, NS-V4, NS-V15 and NS-V35.

Low surge tank level coincident with an HPI actuation signal closes the valves on the affected system.

Service Lines Isolated by Line Break with HPI Isolation Signal

Intermediate Cooling Water Outlet Line
Intermediate Cooling Water Supply Line
Intermediate Cooling to CRDM Cooling Coils
Reactor Coolant Pump Motor Cooling Water Supply
Reactor Coolant Pump Motor Cooling Water Return

SAFETY EVALUATION:

1. The system is designed as safety grade and single failure proof. Thus, the system will perform its safety function when required. The probability of containment isolation occuring on demand is increased.

2. Spurious initiation of an isolation signal will not introduce new accidents into the plant design. Because the system is designed to be single failure proof, spurious initiation of any of the above signals will not isolate any components.

3. No safety margins have been reduced. Only equipment added to insure that radiation is not released from containment.

CHANGE MODIFICATION:

B/A 123004 - Replacement of Containment Isolation Valve WDG-V4

DESCRIPTION:

WDG-V4 was a solid wedge Hancock 950W gate valve with an air-to-open/spring-to-close actuator. The valve stem was sealed by a conventional packing gland. An automatic fluid blocking water supply was connected to the valve bonnet area.

The new valve is a 2" Target Rock Model 80Z-14-008 globe valve with a silicone rubber "O" ring in the disc seating surface, and with a spring-to-close/solenoid-to-open actuator. The valve/bonnet/stem assembly is sealed for zero leakage (no stem packing). Material is 316 stainless steel and the ends are socket-welded. The solenoid actuator uses a low DC current of .2 amps with 90-145 volts. Nameplate rating is 300°F/55 psig/ ASME Section III Class 2

SAFETY EVALUATION:

A detailed purchasing specification was used to assure that the new valve would meet the requirements of WDG system service in the reactor building. The reactor building environment is more severe than that in the auxiliary building where this valve will be installed. Also, the Equipment Qualification Group of EP & I has reviewed all qualification documentation.

The valve design and installation assure that it functions properly for both containment isolation and WDG system boundary. It is expected to perform these functions more reliably than the former system valve. The necessary elimination of the fluid blocking provision is fully justified because of the demonstrated reliability and low leakage of other existing containment isolation globe valves. The leakage is expected to be far below the maximum allowable per 10 CFR Appendix J. The fluid block system will be more reliable for other remaining gate valves when it no longer needs to supply the high leakage to WDG-V4.

The modification will not therefore increase the probability of occurrence or the consequences of an accident.

The valve, during normal plant operation, will function as did the former valve, to allow flow of waste gas from the R. C. Drain Tank to the waste gas treatment system. (This function prevents over-pressure of the R. C. Drain Tank which would lead to damage to the drain tank rupture disc and resultant contamination of containment.) The proposed modification will not therefore increase the probability of occurrence or consequences of a malfunction of equipment I. T. S.

Likewise, it will not create a possibility for an accident or malfunction of a different type than any previously identified in the FSAR since it replaces a valve which already performs the same normal and emergency function.

Use of this valve will reduce radiological exposure in that it will provide tighter WDG system and Reactor Building boundaries. There will be no change in predicted releases beyond license application. Therefore, the designed system will not adversely affect the safety, and is consistent with ALARA requirements.

CHANGE MODIFICATION: BA 412052 - Containment Isolation on High Radiation (RM-5B)

DESCRIPTION OF CHANGE:

A small break LOCA can result in high reactor building radioactivity without generating the 4 psig reactor building isolation signal which would normally isolate the reactor building. Therefore, the actuation of the reactor building isolation is assured by partial isolation and diversity of signals from high reactor building radiation. Piping which could transfer high levels of radiation from either the reactor coolant system or the reactor building are individually monitored to detect high levels of radioactivity and initiate closure of the appropriate valves, or alarm.

Following services lines are detected using existing or new detectors and isolated on high radiation.

Source	Radiation Detector Location	Type of Monitor
Steam Generator Sample Lines	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)
Letdown Line to Purification Demineralizers	Outside R.B.	In line (existing)
Pressurizer and Reactor Coolant Sample Lines	Outside the R.B. between the isolation valve and the sample cooler	Area Gamma Detectors (New)
Reactor Coolant Pumps Seal Return	Outside the RB downstream of the containment isolation Valves (Alarm only, operator action required to close valves)	Area Gamma Detectors (New)
Reactor Coolant Drain Tank Vent Reactor Coolant Drain Tank Pump Discharge	Outside of the tank	Area Gamma Detector (New)
Reactor Building Outlet and Inlet Purge Lines	Outside the R.B.	In line (Existing)
Reactor Building Sump Drain	RB Sump, mounted inside a seismically supported pipe	Sump Area Monitor (New)

SAFETY EVALUATION:

New and existing radiation monitors will be utilized to detect high levels of radioactivity and to initiate the closure of the associated isolation valves in piping which could transfer high level of radiation from either the reactor coolant system or the reactor building to the outside. This modification therefore provides timely isolation of containment on high radiation, reducing the radiological safety concerns without impacting the designed functions of the lines being monitored.

All of the lines which receive the high radiation isolation signal currently receive the 4 psig reactor building isolation signal. Therefore, the spurious initiation of any one of these signals does not introduce a new accident or transient into the plant design.

This modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. This modification does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report. No safety margins defined in the basis for any technical specification have been reduced by this modification.

CHANGE MODIFICATION:

RB Spray System Sodium Thiosulfate Tank Deletion - B/A 412073 - LM-7A

DESCRIPTION OF CHANGE:

This modification removes a section of the 4 inch Sodium Thiosulfate Tank supply line to the Reactor Building Spray Pump suction headers and closes the remaining piping by installing weld neck and blind flanges.

SAFETY EVALUATION:

This is an Important to Safety modification only from a pressure boundary integrity standpoint. The DHRS and RBSS pressure boundary integrity is assured by the "Important to Safety" classification and quality assurance pertaining to this modification.

This modification will not increase the probability of occurrence of an accident because the deleted sodium thiosulfate tank had no normal operation function nor did it support the reactor normal or safe shutdown. Also, the modification will not increase the consequences of an accident because the iodine removal function, originally performed by the deleted sodium thiosulfate, will be successfully peformed by the sodium hydroxide. Off-site radiation doses will remain far below 10CFR100 limits.

This modification will not increase the probability of a malfunction of ITS equipment because (1) corrosion effects of the spray solution on ITS components are considered minimal (per requirements of ANS-56.5 and SRP 6.5.2) if the system sprays solution with pH between 8.5 and 11, and (2) where the sodium thiosulfate piping shares space with ITS equipment, a seismic Category I support was added to the sodium thiosulfate piping to assure its integrity during a SSE. Also, the modification will not increase the consequence of a malfunction of ITS equipment because the iodine removal function will be successfully performed by the sodium hydroxide.

A possibility for an accident or malfunction of a different type than any previously identified in the SAR was not created by this modification because (1) the iodine removal function of the deleted portion of the RBSS was transferred to the portion of RSS that existed before and functioned in the same manner, and (2) there were no components or equipment added to the disconnected sodium thiosulfate piping except a seismic Category I support to assure pipe integrity during a SSE

(continued)

B/A 412073

SAFETY EVALUATION (cont'd.)

This modification will not decrease the margin of safety as defined in the basis of any technical specification because the performance of RBSS remains the same as before the modification. Offsite radiation doses are kept below those regulated in 10CFR100 by keeping spray solution pH between 8.5 to 11 and 8.5 to 9.5 during the injection and recirculation phases, respectively, as specified in ANS-56.5 and SRP 6.5.2.

The modification does not violate any license requirements or regulations because the system performance remains within the requirements of 10CFR100 and the limits delineated in TMI-1 FSAR Paragraph 1.4.70 (Criterion 70). Compliance with these regulations is assured by keeping spray solution pH between 8.5 and 11 as required by ANS-56.5 and SRP 6.5.2.

The modification does not involve a radiological safety concern since the modified RBSS will perform the iodine removal function under LOCA conditions by keeping the radiation doses at the site boundary below 189 Rem for the thyroid and 7.6 Rem for the whole body. These are well within the 10CFR100 guidelines (300 Rem for thyroid and 25 Rem whole body dose).

CHANGE MODIFICATION:

Fire Detection System in the Intake Screen and Pump House (B/A 412388, Phase I)

DESCRIPTION OF CHANGE:

B/A 412388, Phase I, installed six (6) ionization (smoke) detectors in each of the Intake Screen and Pump House (ISPH) pump rooms (i.e., fire zones ISPH-FZ-1 and 2). The detectors are monitored by a local control panel located in the adjacent traveling screen area (i.e., fire zone ISPH-FZ-3). The local control panel provides local alarm and relay contacts for a remote alarm in the TMI-1 Control Room. The local panel is equipped with a four (4) hour self contained emergency power supply.

SAFETY EVALUATION:

The ionization detection system installed was manufactured in accordance with the QA requirements of the USNRC Branch Technical Position (BTP) ASB-9.5-1 "Fire Protection Program".

The balance of the engineering and installation was performed as Important to Safety (ITS). All equipment is seismically mounted so as not to create a missile hazard. Installation of the ionization detection system meets a commitment made to the NRC in GPUN Report "TMI-1 Fire Hazards Analysis Report and Appendix R, dated June 28, 1982, Section III.G., Safe Shutdown Evaluation". This modification, along with other modifications discussed in the above report will result in the ISPH being in compliance with 10 CFR 50, Appendix R. The probability of 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. It is, therefore, concluded that the subject modification does not involve an unreviewed safety question per the criteria of 10 CFR 50.59.

CHANGE MODIFICATION:

Emergency Feedwater Differential Pressure Flow Measurement System (412398)

DESCRIPTION OF CHANGE:

As a part of the Emergency Feedwater System (EFSW) Upgrade, GPUN committed to install by "Restart" safety grade flow devices, which were to indicate in the control room the system operating flow conditions. (Restart Report Section 2.1.1.7). To meet this commitment safety grade differential pressure transmitters system utilizing annubars were installed.

SAFETY EVALUATION:

The EFWS functions continue to operate as designed and are not degraded by the additions of Emergency Feedwater Differential Pressure Flow Measurement System. The plant control performance during normal and abnormal operations remain unchanged.

The changeout of flow sonic transmitters to delta pressure transmitters with associated equipment and cable routing increase reliability of readout information for operators use.

This modification does not in any way increase the probability of an accident or malfunction or decrease the margin of safety.

PROCEDURES

1. Procedures for TDR 400

   Guidelines for Plant Operation with Steam Generator Tube Leakage

PCR. No.	Procedure	Procedure Title
1-OS-83-0266	1203-24	Steam Line Rupture Detection System Actuation
1-OS-83-0268	1102-11	Plant Cooldown
1-OS-83-0269	1101-1	Plant Limits and Precautions
1-OS-83-0270	1102-10	Plant Shutdown
1-OS-83-0271	1202-12	Excessive Radiation Levels
1-OS-83-0272	1301-1	Shift and Daily Checks
1-OS-83-0274	1106-13	Powdex System
1-EG-83-0051	1303-1.1	Reactor Coolant System Leak Rate

The procedural changes incorporating the TDR-400 recommendations provide the necessary operating restrictions to control radioactive contamination of the secondary plant. These restrictions include OTSG Tube Leakage determination and monitoring requirements; disposal as radioactive waste of the condensate "polishing" powdex resins; Turbine Building sump monitoring and contamination limits; Industrial Waste operating restrictions and administrative plant shutdown instructions based on OTSG tube leakage. Incorporation of these recommendations along with operator training on the new procedure guidance will control secondary plant contamination and offsite doses to within the acceptable levels developed by GPUNC. Because this level of contamination is well below NRC guidelines and 10 CFR 20 regulations, there is no Environmental or Nuclear Safety impact.

PROCEDURES

2. Procedures for TDR 406

Steam Generator Tube Rupture Procedure Guidelines

PCR No.	Procedure	Procedure Title
1-OS-83-0230	H-1-8	T Sat Margin A/B Low
1-OS-83-0231	1202-2	Station Blackout
1-OS-83-0232	1202-2A	Station Blackout with Loss of Both Diesel Generators
1-OS-83-0233	1202-4	Reactor Trip
1-OS-83-0234	1202-6B	Small Break LOCA
1-OS-83-0235	1202-6A	Loss of Reactor Coolant within Makeup Capability
1-OS-83-0236	1202-6C	Large Break LOCA
1-OS-83-0237	1202-14	Loss of Reactor Coolant Flow
1-OS-83-0238	1202-36A	Loss of Instrument Air-Backup Air Available
1-OS-83-0239	1202-36B	Loss of Instrument Air-No Backup Air Available
1-OS-83-0240	1202-39	Inadequate Core Cooling
1-OS-83-0241	1102-16	RCS Natural Circulation Cooling
1-OS-83-0242	1202-5	OTSG Tube Leak/Rupture
1-OS-83-0270	1102-10	Plant Shutdown

The procedural changes incorporating the TDR-406 recommendations provide instructions to the operator for the best method of handling an OTSG tube leak casualty to minimize off-site dose. These instructions are based on up-to-date industry experience in addition to analytical work on multiple OTSG Tube Failures. These instructions include subcooling margin limits, emergency RCP NPSH limits, OTSG Tube to Shell temperature difference limits, OTSG isolation criteria, and RCP trip criteria based on subcooling margin. These new instructions along with operator classroom and simulator training will ensure that an OTSG tube rupture casualty is controlled and terminated to minimize off-site exposure to within NRC accepted limits. The NRC has endorsed the technical content of TDR-406 with their SER concerning our tube rupture emergency procedure.

Reactor Coolant System Cleaning

Reactor Coolant system cleaning was initiated for the removal of residual sulfur contaminants, to the extent possible, from the surfaces of primary system equipment and piping. Reducing the quantity of residual sulfur was necessary to preclude the possibility of reactivation of the corrosion mechanism which had damaged the Steam Generator tubes. Sulfur which was present as deposits in the pressurizer vapor space has been removed by hydrolasing.

The cleaning procedure rapidly converts the insoluble reduced sulfur to an oxidized, soluble form under protective high pH conditions. The sulfur is then removed using installed plant purification system.

The selected cleaning process utilizes a low concentration (~20 ppm) of hydrogen peroxide at an elevated pH (8.0) and at a slightly elevated temperature (~130°F). Hydrogen peroxide is normally formed in slightly lower concentrations (5-10 ppm) in the reactor coolant at every shutdown due to radiation effects on the coolant. Therefore, the use of this additive will not adversely affect the normal system materials. In addition, many PWRs add more peroxide at shutdown to quickly solubilize some nuclides and then remove them to avoid interference in the refueling process. Peroxide concentrations of up to the 15-20 ppm level have been used in this process with no adverse effects.

Stress corrosion tests carried out with highly stressed C-rings fabricated from tubing removed from TMI-1 have shown no detrimental effects from the cleaning process. Longer term loop tests with realistically stressed tubing are also underway.

The TMI-1 primary system was chemically cleaned without adverse effects on the primary system, the remainder of the plant or the public. The Steam Generators were not required for decay heat removal at any time during the cleaning procedure. No unanalyzed accident was introduced and the probability or consequences of any analysed accident was not increased.

FIRE PROTECTION PLAN

The revised TMI-1 Fire Protection Plan was submitted to the NRC on January 1, 1984 by GPUN letter 5211-83-359.

The Plan was updated to reflect changes in organization and fire brigade manning and training. The most significant change in the Plan (which is also the only deviation from the basis of the Fire Protection SER for License Amendment #44) is in the assignment of the Shift Maintenance Foreman (non-licensed) to the brigade leader position. This reassignment relieved the SRO-licensed Shift Foreman from the responsibility of fire fighting. To compensate for this change and to ensure nuclear safety, a shift foreman or CRO is required to respond to all fires which could affect plant safety. Being free of fire fighting duties, nuclear safety is enhanced since with this organization, the SRO is allowed to concentrate on the impact of the fire on the plant and its operation. The ability to control the plant will be demonstrated during fire drills and emergency exercises as outlined in AP 1038, the implementing procedure for the Plan.

Other changes to the Plan reaffirmed the SER requirements on training schedules and attendance. The Plan now clearly requires all fire brigade members to attend well defined initial training, all quarterly training making up the two-year program, drills, and monthly meetings.

The revised Plan exceeds the SER requirements in most areas and in addition now defines requirements for annual firewatch training and annual training of personnel providing a support role to the fire brigade. A final significant change is the removal of TMI-2 from this Plan which was previously covered under a common site program.