GPU Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

(717) 948-8005

April 20, 1992
C311-92-2047

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit I (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
10 CFR 50.59 Report for 1991

In accordance with the requirements of 10 CFR 50.59, enclosed are summaries of the changes to TMI-1 systems and procedures, which were effected during the period of January - December 1991. Attachment 1 of this report addresses those activities which directly affected systems/components described in the SAR. Attachment 2 addresses those activities for which GPU Nuclear performed a safety evaluation, due to the potential for the activity to impact nuclear safety or safe plant operations but which would not directly impact SAR systems/components.

Sincerely,

T G Broughton

T. G. Broughton
Vice President & Director, TMI-1

WGH

Attachments

cc: Administrator, Region I
TMI-1 Senior Project Manager
TMI Senior Resident Inspector

JE47

Activities Directly Affecting Systems/Components Described in the Safety Analysis Report

I. Tests and Experiments

Test: Inservice Hydrostatic Test of the Nuclear River Water System

Description of Test: In accordance with Technical Specification Section 4.2.1 a hydrostatic test was performed on the Nuclear River Water System to verify that leak-tightness requirements were satisfied on a ten year interval.

Safety Evaluation Summary: Evaluation identified no adverse affect on nuclear safety or safe plant operations on either the Decay Heat Removal or Spent Fuel cooling since insufficient heat remains during shutdown to affect system operation for the duration of the hydro. The test did not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR since the duration of the hydro was less than the interval which the Spent Fuel Pool can go without cooling; cooling was not completely removed. There was no increase in the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the SAR since a satisfactory test result verifies the capability of the system boundary to perform its safety function. The possibility of an accident or malfunction of a type different than previously evaluated was not created since cold shutdown accidents are not addressed by the UFSAR. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the test since the test is required by the Tech Specs.

****

Test: Lithium Tracer Test for Venturi Nozzle Calibration (SE 412565)

Description of Test: The purpose of the test was to calibrate the replacement feedwater flow venturis by accurately measuring feedwater flow to each OTSG using a chemical tracer.

Safety Evaluation Summary: The single aspect of the chemical tracer test which had possible impact on accident probability was related to lithium injection to the secondary side of the plant. The OTSG tube rupture is the only accident evaluated in the SAR which could have been impacted by alteration of chemistry on the secondary side. Based on the analysis in the safety evaluation, lithium was determined to have no degrading effect on the OTSGs. The probability of occurrence of any previously evaluated accident was not increased. The performance of the test did not introduce any new accident precursors. Reactor operations were maintained within license limits. Administrative controls were in place to preclude any adverse condition resulting from improper control of equipment or conduct of the test. The nuclear instrumentation was calibrated based on conservative estimates of core power. Therefore, the margin of safety as defined in the bases for any Technical Specification relative to core power were not reduced. There were no adverse affects on nuclear safety or safe plant operation.

****

II. Procedure/Document Changes

Procedure: AP 1038 Administrative Controls Fire Protection Program (PCR 1-EG 91-0040)

Description of Change: The change included the following items:

a. changed exhibits 2 and 7 to allow 2 year valve cycling versus 18 month cycling based on accessibility during refueling outages,
b. changed transformer deluge testing to a refueling versus an 18 month interval,
c. added notes and new procedures/title changes to exhibits listing implementing documents (exhibit 5),
d. changed exhibit 4 to delete the Unit 2 Admin Computer Room Halon system since it is out of service and to be removed.

Safety Evaluation Summary: The change added references to new/revised implementing procedures needed to maintain systems and components per insurance and NFPA codes and standards. None of the changes were associated with safety related plant areas, systems or components. The change to exhibit 5 adds a note to aid in assuring Rockbestos Fire Zone R cable is inspected per electrical maintenance procedures as committed in the FSAR and NRC SER on Fire Zone R cable used to meet 10 CFR 50, Appendix R, Section III. G requirements. The change to exhibit 2 replaced the Standard Tech Spec 18 month valve cycling with the new section for cycling inaccessible valves on a two year/refueling basis. The original basis (18 months) was for refueling, therefore consistency is maintained. Since the procedure changes did not affect the ability to maintain, monitor or achieve safe shutdown in the event of a fire, the change was allowed per License Condition 2.C.(4). The procedure change did not constitute an Unreviewed Safety Question.

****

Document: Final Safety Analysis Report Section 12.1.1.2.d

Description of Change: The change allows exemption from the requirement that the Director, O&M, TMI-1 have a bachelor's degree in science or engineering. The exemption is determined acceptable by the Office of the President, GPU Nuclear.

Safety Evaluation Summary: The change did not adversely affect nuclear safety since the Office of the President continued to use prudent judgement in accepting candidates for the position of Director, O&M. The change neither increased the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR nor created the possibility of an accident or malfunction of a type different than previously evaluated and did not reduce a margin of safety defined in the basis of the Technical Specifications since no change was made which affects plant equipment. The requirement for a degree for the position was not assumed in any analysis. No change to the Technical Specifications was required. The evaluation concluded that no unreviewed safety question existed. The change did not constitute an Unreviewed Safety Question.

****

Procedure: OP 1101-2 Plant Setpoints (PCR 1-OS-91-0819)

Description of Change: The Core Flood Tank Metal Temperature Low Alarm setpoint was reduced from 115°F to 100°F.

Safety Evaluation Summary: The change was made via direction in Memo 5360-91-254. It was determined that as long as the metal temperature is maintained >100°F the tanks will be above NDTT + 0°F until the tanks depressurize to less than 140 psig during a large break LOCA. Core Flood Tank operability is assured as long as initial temperature is >100°F. As a result, the change did not constitute an Unreviewed Safety Question..

****

Procedure: SP 1300-3M IST of SW-P-2A/B and Valves (PCR 1-OS-91-0327)

Description of Change: The procedure was deleted.

Safety Evaluation Summary: The Safety Evaluation justified the reclassification of the cooling water portion of the Heating, Ventilation and Air Conditioning system (HVAC) for the Intake Screen and Pumphouse from Nuclear Safety Related to "other". The determination concluded, based on the results of calculation C-1101-536-5360-001 R0, that the maximum ambient would not affect component performance. No adverse affect on nuclear safety or safe plant operations was found to exist since the Limitorque and EIM valve operators would continue to operate within their design margins at 117.6°F. The activity did not increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the SAR since equipment continued to perform its intended function following the declassification of SW-P2A and B. The possibility of an accident or malfunction of a type different than previously evaluated was not created because equipment performance was not adversely affected by the change. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the change for the same reason. All equipment important for nuclear safety would continue to operate following the loss of Screen House HVAC. There was no unreviewed safety question.

****

Procedure: SP 1301-1 Shift and Daily Checks (PCR 9-91-0364)

Description of Change: The procedure revision increased the time period allowed for validation of primary to secondary leakrate values in excess of 6 gallons per hour above baseline from 24 to 72 hours.

Safety Evaluation Summary: The change was determined to have no adverse affect on nuclear safety or safe plant operations since abnormal leakage would continue to be detected in a timely manner, continued operation near or at the limit was not unsafe and the resulting additional radiation exposure was limited to a small fraction of the 10 CFR 20 limits. The change did not involve an unreviewed safety question.

****

Procedure: SP 1303-11.18 Reactor Building Local Leakrate Testing
(PCR 1-91-0041)

Description of Change: The procedure change incorporated changes resulting from the addition of test connections which were accomplished under CMR 90-0154 and included sketches to show the connections. Reverse direction testing of WDL-V534 was included in the procedure change. Previous hand drawn sketches were replaced with CAD sketches for clarity and consistency. Additional changes resulting from TCN 91-0132, rewrites of the index, enclosures 4A and 4B and the addition of enclosure 4C were also included.

Safety Evaluation Summary: Review of all changes with the exception of the reverse testing of WDL-V534 found no potential to adversely affect nuclear safety or safe plant operations and required no safety evaluation be performed. Reverse testing (in the non-accident direction) of WDL-V534 was considered to have the potential to adversely affect nuclear safety or safe plant operations and was evaluated on that basis. As a result of the evaluation it was determined that the procedure change did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR since equivalent test results were obtainable. The possibility of an accident or malfunction of a type different than previously evaluated was not created since the procedure was performed during cold shutdown when containment integrity is not required. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the change since cumulative local leakrate testing leakage is typically much less than 0.6 La. The change did not constitute an Unreviewed Safety Question.

****

Procedure: TMI-1 Chemical Cleaning Process Evaluation

Description of Change: Chemical cleaning of the TMI-1 steam generators was performed to restore their capacity to 100%. The process dissolved and removed corrosion products that deposited in the broach openings of the secondary side of the steam generators. The deposits cause an increased steam generator operating water levels that result in forced power reductions to avoid flooding of the feedwater nozzles. The deposits originated from the feedwater and condensate systems and accumulated in the steam generator during operation. Prior attempts to remove the deposits by mechanical means have only provided short term restoration of full power capability.

Safety Evaluation Summary: The chemical cleaning process was evaluated and found not to have an adverse affect on nuclear safety or safe plant operations. It was demonstrated that the process will not result in corrosion of the steam generator materials in excess of allowable limits. An on-line corrosion monitoring system monitored the process and verified that the cumulative corrosion did not exceed allowances. Since the water level in the downcomer is reduced and therefore generator inventory is reduced by chemical cleaning, the margin of safety with respect to recriticality analysis for a main steam line break, for the analysis of containment pressure and offsite dose calculation were not affected. No foreign materials or loose parts were left inside the steam generator which would cause a tube rupture during operation. As a result, no increase in the probability of occurrence of an accident or malfunction of equipment important to safety. The consequences are also unaffected since the activity did not adversely affect the systems which mitigate the consequences of the tube rupture. The probability of an accident or malf... ction of a different type than evaluated previously in the SAR is not created since tube integrity is not affected by the process. No radiological safety concern was determined to exist. The change did not constitute an Unreviewed Safety Question.

****

Document: Fire Hazards Analysis Report

Description of Change: Non-physical plant changes recommended by QA Audit S-TMI-90-20 to the Fire Hazards Analysis Report for TMI-1 were made as deemed appropriate. The changes involved fire barriers and extinguisher.

Safety Evaluation Summary: The description of fire walls and location of extinguisher as changed to correct the description to more accurately describe the physical condition did not adversely affect nuclear safety or safe plant operation since the components involved do not perform a safety function. The probability of occurrence or consequence of an accident or a malfunction of equipment important to safety was determined not to be increased. The activity did not create the possibility for an accident or malfunction of a different type than any previously identified in the SAR since no structures, systems or components were modified as a result of the changes. The margin of safety as defined in the SAR or as defined in the basis for any Tech Spec was not reduced by the change since no margin is defined for barriers or extinguisher. An unreviewed safety question was not created.

****

Document: Increase RCP Number 1 Seal Leakoff

Description of Change: The change permitted operation of the TMI-1 Reactor Coolant Pumps with two pumps having up to 14 gpm leakoff at the number 1 seal simultaneously.

Safety Evaluation Summary: Evaluation of the document found that the margin of safety as defined in the SAR or as defined in the basis for any Tech Spec was not reduced by the change. The limitations on reactor coolant leakage specified in Tech Spec Section 3.1 were not impacted since the increase of leakage flow is to the number 1 seal leakoff which is returned to the RCS via the Makeup System. Nuclear safety and safe plant operations were not adversely affected since the increased flow limit will not increase the probability of seal failure during normal operation. The probability of occurrence or consequence of an accident or a malfunction of equipment important to safety was determined not to be increased. The activity did not create the possibility for an accident or malfunction of a different type than any previously identified in the SAR. An unreviewed safety question was not created.

****

Docu. ent: Revision of the Reactor Coolant Lithium Specification (SP 1101-28-001)

Description of Change: The TMI-1 Primary Water Chemistry Specification for operations, hot shutdown and hot standby were revised to increase the ppm Lithium concentration range from the existing limit of 0.25 to 2.2 ppm to 0.25 to 3.6 ppm Lithium.

Safety Evaluation Summary: The change did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, did not create the possibility of an accident or malfunction of a type different than previously evaluated in the SAR and did not adversely affect nuclear safety or safe plant operations. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the document revision. The change resulted in reduced corrosion of plant construction materials and thereby reduced corrosion product release, transport and deposition on the fuel and other system surfaces resulting in lower radiation fields and improved fuel performance. This change did not constitute an Unreviewed Safety Question.

****

Document: TMI-1 Cycle 9 Reload Design

Description of Change: The core and fuel design as depicted by the core reload design, operating limits and operation with the four Westinghouse Lead Test Assemblies were updated for Cycle 9 operation.

Safety Evaluation Summary: Implementation of the Cycle 9 design and operating limits did not adversely affect nuclear safety or safe plant operation. This is supported by conservatism in the design and operating limits. The probability of occurrence or consequence of an accident or a malfunction of equipment important to safety was determined not to be increased since only the cycle length and core configuration were changed. The activity did not create the possibility for an accident or malfunction of a different type than any previously identified in the SAR since Cycle 9 was developed using B&W approved methodologies and operated using standard plant procedures. The margin of safety as defined in the SAR or as defined in the basis for any Tech Spec was not reduced by the change since evaluation has shown that no previously established design criteria or limits are exceeded. An unreviewed safety question was not created.

****

III. Modifications

Modification: OTSG Blowdown Orifice Installation (BA 128087 / JO 13358)

Description of Modification: This modification replaced the needle valves (CA-V-357A/B) in the OTSG blowdown lines with orifices. This modification provides a more reliable method of restricting OTSG blowdown flow from the secondary side of the OTSGs to the condenser. The needle valves were locked in a slightly open position to restrict flow. Flow restriction is required to prevent excessive vibration. The orifices are more suitable for this application since they are more reliable and resilient to operating conditions.

Safety Evaluation Summary: Replacement of the needle valves with orifices did not adversely affect the performance of the blowdown system. The orifices are sized for the same pressure differentials which the needle valves provided. The resulting flow rates were unchanged and secondary side water chemistry was not affected. The orifices perform the same function as the needle valves that were replaced. The modification also precluded the possibility of vibration due to inadvertent opening of the needle valves. Therefore, this modification did not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety. This modification was not safety related and did not change plant performance or system performance. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Miscellaneous Waste Evaporator Distillate Pump Vent and Pressure Gauge Modification (BA 128112 / JO 28451)

Description of Modification: This modification installed a pressure gauge and vent valve on the discharge of distillate pump (WDL-P-21) for the Miscellaneous Waste Evaporator. The purpose of this modification was to provide indication of pump performance and to allow venting in order to eliminate air entrapment.

Safety Evaluation Summary: The installation of the pressure gauge and vent valve connection on the discharge of the distillate pump had no adverse impact on the performance of the pump. The gauge serves as an indication of pump performance during operation to aid in preventive maintenance for the pump. The vent provides a pathway to eliminate entrapped air in the system during startup of the pump, which aids pump operation and longevity. No changes to the function of the system occurred. The processing of the distillate was unaltered by this modification. The margin of safety as defined in the Safety Analysis Report or as defined in the basis for any Technical Specifications were not reduced by this modification since no adverse effects on system performance or design criteria occurred. The probability of an accident or malfunction of a type different from any evaluated previously in the SAR was not created since the pressure gauge and vent did not alter the function of the distillate pump, or the design criteria of the system. The vent and gauge help to reduce the chance of pump malfunction by eliminating air entrapment and providing an indication of system pressure as a diagnostic aid. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Differential Pressure Gauges for DH-P-1A/B IST (BA 128112 / JO 33397)

Description of Modification: The modification installed differential pressure gauges on Decay Heat Removal Pumps 1A and B to improve the capability of performing in-service testing on those pumps. The existing discharge pressure gauges (DH-PI-1/2), tubing and supports were removed and replaced with new ΔP gauges, and associated tubing and supports. Additionally, the 0-400 psig discharge gauges designated PI-1224 A/B were replaced with 0-600 psig gauges.

Safety Evaluation Summary: The ΔP gauges are used to indicate system pressures for the purpose of IST. They have no effect on system performance and do not affect nuclear safety or safe plant operations. Similarly, the probability of an accident or malfunction of a type different from any evaluated previously in the SAR was not created since the pressure gauges did not alter the pump function or the design criteria of the system. This modification did not constitute an Unreviewed Safety Question.

****

Modification: ATWS Implementation- 10CFR50.62 (BA 412462 / JO 26291)

Description of Modification: The modification implemented the commitments made to the NRC as described in SAR via GPUN File No. C311-89-374. The commitments were met by the installation of a Diverse Scram System (DSS) and an alternate turbine trip circuit and by actuation of EFW by the Heat Sink Protection System (HSPS) upon sensing low OTSG level. The DSS is fed from existing RCS pressure sensors and trips existing CRD system circuit breakers. New hydraulic oil pressure switches on the main feedwater pumps were added to provide signals for the alternate turbine trip. A DSS manual pushbutton was also added to the control console, and isolated signals from the DSS are sent to the computer for RCS pressure signals PT-949 and PT-963.

While these are not part of the ATWS commitments, this modification also replaced the main feedwater differential pressure switches associated with HSPS with hydraulic oil pressure switches on the main feedwater pumps.

Safety Evaluation Summary: This modification does not introduce any new accident or malfunction not previously evaluated, not does the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the TMI-1 FSAR. This modification did not decrease the margin of safety as described in the Technical Specifications because the modification did not impact any system safety functions and the modification was performed in accordance with existing seismic and electric/physical separation requirements.

The change in the pressure switches for sensing loss of feedwater as input into the HSPS did not affect the safety function of the HSPS.

The modification did not affect the safety function of any interfacing systems and did not affect the UFSAR Chapter 14 accident analysis. Therefore, this modification did not involve an unreviewed safety question or new environmental impact and did not adversely affect nuclear safety.

****

Modification: Installation of Air Sample Station (BA 412528 / JO 18428)

Description of Modification: This modification provided mounting fixtures for air sampling instruments at various locations in the TMI-1 Auxiliary and Fuel Handling Buildings. Previously, numerous regulated air samplers, model RAS-1, and beta particulate monitors, model AMS-3, were in service on temporary tables, carts, etc. The installation of stationary support facilities for the instruments benefit the air monitoring program.

Safety Evaluation Summary. The function or operation of the affected radiation monitors was not affected by the modification. They were physically repositioned from their previous supports to new supports in the same general location. Therefore, this modification, which installed permanent support fixtures as Air Sample Stations in the Auxiliary and Fuel Handling Buildings did not adversely affect nuclear safety or safe plant operations and did not involve an Unreviewed Safety Question.

****

Modification. Containment High Range Radiation Monitors Input to SPDS (BA 412528 / JO 28444)

Description of Modification: The modification removed the Reactor Building High Range Radiation Monitor (RM-G8) from the plant configuration and replaced the RM-G8 input signal to the Safety Parameter Display System (SPDS) with inputs from existing Containment High Range Radiation Monitors RM-G22 and RM-G23. Signal isolation devices were also installed at the output of existing ratemeters, RM-G22 and RM-G23, isolating the signals to the respective recorders RM-R6A and RM-R6B and to the computer.

RM-G8 was part of the original plant configuration and was intended to quantify radiation levels indicative of a loss of coolant accident. RM-G8 was the containment dome monitor able to detect radiation levels to 10^6R/hour. RM-G22 and RM-G23 were installed during the period that TMI-1 sat idle after the TMI-2 accident. They were installed to bring the plant in conformance with NUREG 0737 and Reg. Guide 1.97 and provide radiation level monitoring capability during and following an accident. The monitors are qualified to withstand accident environments and therefore provide a more reliable means of fulfilling the original RM-G8 design function.

Installation of this modification improved the accuracy and reliability of the post accident Reactor Building high range area radiation level information supplied to SPDS. It removed a detector, RM-G8, which was obsolete and identified as a possible source of error from the plant configuration.

Safety Evaluation Summary: The modification to isolate/abandon RM-G8 was determined to have no adverse impact on nuclear safety or safe plant operations. The margin of safety as defined in the basis for the Technical Specifications is not reduced since the replacement monitors exceed the post accident capabilities of RM-G8 and their associated technical specifications ensure their functional capabilities. No Unreviewed Safety Question or environmental impact were found to exist as a result of the modification.

****

Modification: Instrument Air System Upgrade (BA 412551 / JO 20474)

Description of Modification: As a result of a review performed by the B&W Owner's Group Safety and Performance Improvement Program (SPIP), the following modifications were made to the Instrument Air System (IAS):

> A new IAS train was installed consisting of a 440 SCFM compressor, receiver, pre-filter, after-filter, and dryer. The previous compressors and filter/dryer train were retained as standby units increasing overall system reliability and maximum air available to respond to system failures. One of the previous compressors also satisfies the Appendix R remote shutdown requirements.
>
> In order to prevent cycling of the previous compressors (IA-P-1A/B) due to system leakage between existing compressors and the new isolation valves, a cross-tie was installed between the new compressor (IA-P-4) header and the 1" cross-tie between receivers IA-T-1A/B. The new cross-tie permits the required makeup flow to the receivers, but prevents flow of undried air from entering the IA header.

Safety Evaluation Summary: The margin of safety as defined in the SAR or Technical Specifications was not reduced because the modification increased system redundancy and reliability by adding a new IA train to be used as the lead unit. The IA system is non-nuclear safety related and is not part of the engineered safeguards system. Therefore, this modification did not decrease a margin of safety nor create the possibility for an accident or malfunction of a different type than any previously evaluated in the SAR. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Bailey BY Transmitter Replacement (BA 412553 / JO 26287)

Description of Modification: Because of obsolescence and inability to provide sufficient accuracy for system usage needs, the eight Bailey BY transmitters (RC14A-DPT-1 through to 4 and RC14B-DPT-1 through 4) in the RC system were replaced with Rosemount model 1153HD6RC transmitters and two By transmitters in the SP system (SP-1A-LT1 and SP-1B-LT1) were deleted. Isolated signals were routed to the plant computer from the existing OTSG level equipment in the signal conditioning cabinets A1 and B1.

The existing RPS RC flow input Buffer Amplifiers in channels A & B were modified by connecting the internal Reference Voltage Source and terminating the input signal with a resistive load. The existing Bailey Square Root Extractor modules located in the RPS cabinets were modified by connecting its internal Reference Voltage Source and terminating the input signal with a resistive load. An existing spare isolation module was used in Signal Conditioning Cabinet A1 and B1 to provide non-1E signals from the transmitter loops to the plant computer inputs.

Safety Evaluation Summary: This modification has resulted in no decrease to a margin of safety described in the Technical Specifications since the replacement transmitters are equal or better in loop accuracy and response time of the instruments being replaced. The transmitters deleted by the modification were non-safety, utilized for computer input only and had no control or actuation function.

The modification does not increase the probability of occurrence or consequence of an accident previously evaluated in the FSAR or create the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR. No Unreviewed Safety Question or environmental impact were found to exist as a result of the modification.

****

Modification: Station Blackout Modification (BA 412567 / JO 26289)

Description of Modification: This modification involved activities required to modify the TMI-2 Emergency Diesel Generator DF-X-1A and auxiliary support systems for use as an alternate AC supply for TMI-1, to meet the intent of 10CFR50.63 in accordance with the Station Blackout Requirements of NUMARC 87-00. The completed modification is in accordance with the Station Blackout SDD-T1-700A Division I and SDD-T1-700A Division II. The modification included changes to the following mechanical systems: cooling water system, HVAC system, diesel fuel oil system, the starting, service and instrument air systems, the fire service system and the Diesel Building sumps. Electrical systems modified included: the 4160, 480 and 120 VAC systems, the 125 VDC system and the fire detection system.

Safety Evaluation Summary: The modification upgraded the TMI-1 safe shutdown capability by providing an alternate AC supply (SBO D/G) to supply safe shutdown loads during a station blackout or balance of plant loads during a Loss of Offsite Power. This AAC supply is capable of supplying power to TMI-1 safety electrical buses within ten minutes from the onset of a station blackout. Fire service and detection system modifications provided detection readout in Unit 1 and fire fighting capability at the diesel. The modification did not introduce any accident or malfunction not previously evaluated, nor did it increase the likelihood of occurrence. In addition, the modification did not affect the safety functions of any interfacing systems or structures and did not affect UFSAR Chapter 14 accident analysis. Therefore, the modification did not adversely affect nuclear safety and did not involve an Unreviewed Safety Question or new environmental impact.

****

Modification: Turbine Lube Oil Tank Replacement (BA 412588 / JOs 28472 and 30040)

Description of Modification: This modification removed an empty 7,000 gallon Ammonium Hydroxide Storage Tank and replaced it with a 15,000 gallon Turbine Lube Oil Storage Tank. The new tank is located outdoors within a concrete containment pit and a concrete floor slab.

Safety Evaluation Summary: This modification satisfied the requirements of 40 CFR 280 for underground storage tanks, which requires those tanks with hazardous liquids to be provided with release detection, tank tightness testing, or to be replaced depending on tank age. The systems modified were non-NSR and were not part of the Engineered Safeguards System. This modification did not affect the safety function of any interfacing systems and did not affect any accident scenario in the SAR. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Modification of Penetration #221ES and #222ES for OTSG Sleeving (BA 412597 / JO 34047)

Description of Modification: This modification installed blank flanges on Penetrations #221ES AND #222ES, which penetrate the Reactor Building from the Turbine Building. The flanged end pipes had their blank flanges removed to provide accessible openings for the temporary routing of electrical wires, cables and hoses required to support outage related work. For the 9R outage these penetrations were used for OTSG tube sleeving, plugging, and eddy current operations.

Safety Evaluation Summary: The modification of Penetrations #221ES and #222ES provided accessible openings for the temporary routing of wires, cable and hoses from outside of the containment into the Reactor Building to support the OTSG outage activities. Installation of a blank flange outside the reactor Building and a blank flange inside the reactor Building will provide double isolation barriers at these two penetrations. The associated Penetration Pressurization tubing was removed when the pipe caps were cut from these two electrical spare penetrations. However, the PP tubing connections are available to provide a tap for isolation barriers leak rate testing of penetrations #221ES and #222ES via new valves PP-V227 and PP-V228, respectively.

This modification did not increase the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report since the integrity of pressure boundary and seismic boundary of the Containment Isolation System are maintained. Therefore, this modification did not adversely affect nuclear safety or safe plant operation. This modification did not create a possibility for an accident or malfunction of a different type from any previously identified in the FSAR because the design basis and system performance of the Containment Isolation system was not changed. This modification did not decrease the margin of safety as defined in the basis of any technical specification since the system performance and the safety function of the Containment Isolation system were not changed. This modification did not constitute an Unreviewed Safety Question.

****

Modification: NI 1 & 2 Changeover to NI 11 & 12 (BA 412598 / JO 38251)

Description of Modification: The existing NI-1 and NI-2 Source Range Instrument Loops were susceptible to noise and difficult to maintain. As a result, the existing Bailey Meter Co. NI-1 and NI-2 source range monitoring loops were disconnected from peripheral equipment. The NI-1 and NI-2 loops will remain in place and functional until the modules and spared wiring are removed as a separate modification. That work is tentatively scheduled for refueling outage 10R.

The existing Gamma-Metrics NI-11 and NI-12 have had their Source Range Channels extended by one decade to assure that the Tech Spec requirement for greater than one decade overlap with Intermediate Range Channels is satisfied. The disconnected NI-1 and NI-2 peripherals were connected to NI-11 and NI-12 through signal isolators. The signals were count rate, count rate of change, reactor building evacuation alarm, and rod stop control signal to the CRD control system. The isolated outputs from NI-1 and NI-2 were connected to the RPS cabinets for future connection to a portable, audible transducer when needed by nuclear engineering.

Safety Evaluation Summary: Plant safety is enhanced by the design of the more reliable monitoring instrumentation. The modified instrumentation was designed to operate during all modes of plant operation. The modification was performed in accordance with existing seismic and electrical/physical separation requirements. The modification did not affect the safety functions of any interfacing systems and did not affect the FSAR Chapter 14 accident analysis. It did not reduce the margin of safety. Therefore, this modification did not adversely affect nuclear safety and did not involve an Unreviewed Safety Question or new environmental impact.

****

Modification: RM-G22/23 Redundant Power Supply Modification (BA 412602 / JO 37568)

Description of Modification: This modification corrected a lack of required physical separation between the 120 VAC power source cables for the Containment High Range Radiation Monitors RM-G22/23. Redundant Class 1E 120 VAC power source cables were provided for the aforementioned monitors. The modification also extended the fire detection system to the interior to PRF.

Safety Evaluation Summary: Installation of redundant, appropriately separated 120 VAC power cables to radiation monitors RM-G22/23 and a fire detector in panel PRF did not affect the safety functions of any interfacing systems and did not affect the FSAR Chapter 14 Accident Analysis. Therefore, this modification did not adversely affect nuclear safety and did not involve an Unreviewed Safety Question or new environmental impact.

****

Modification: Replacement of Seal Injection Filter (MU-F-4A) Phase 1 (BA 412610 / JO 38925)

Description of Modification: This modification is being performed in two (2) phases. The first phase concerns MU-F-4A only. The second phase, which affects MU-F-4B, is scheduled to be completed during cycle 9 or the 10R outage.

This modification provided increased filtration of Reactor Coolant Pump (RCP) seal injection water via the installation of new filter housings which can use larger and finer filter elements. This increased filtration optimizes seal performance and reduces the potential for unplanned (midcycle) seal maintenance.

Safety Evaluation Summary: As a consequence of the greater filter element flow capability and lower pressure loss when filter element is dirty, the portion of the system affected by this modification will contribute lower maximum pressure losses than the previous configuration. This has no effect on seal injection flow rate, seal leak off rate, letdown flow rate to maintain desired RCS inventory and seal injection pressure. There are no SAR or Technical Specification safety margins that were affected by this modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Replacement of Main Generator and 1A/1B Auxiliary Transformer Watt Hour Meters (BA 412702 / JO 34323)

Description of Modification: The modification replaced the existing watt hour metering (WHM) for the Main Generator and the 1A & 1B Auxiliary Transformers with state-of-the-art programmable multi-functional meters. The new WHMs also perform the functions for the Printing Demand Meters and the Main Transformer Loss Compensator; therefore, these devices were disconnected and removed.

Safety Evaluation Summary: This modification replaced existing WHM with state-of-the-art programmable multi-functional meters. The reliability of the Main Generator CT and PT Circuits were improved by removal of transducers that were no longer in use for control or indication functions. Thus the probability of Main Generator malfunction due to a failure of one of these devices was eliminated. It did not interface with any NSR device, therefore, electrical isolation, single failure criteria, and environmental qualification were not affected by the modification. Neither nuclear safety nor safe plant operations were affected since the modification did not alter any control or protective device function. This modification did not constitute an Unreviewed Safety Question.

****

Modification: HP Access Control Point/Hot Instrument Repair Shop Relocation (BA 418748 / JOs 12706 and 22759)

Description of Modification: This modification affected the following facilities:

- High volume during outages, through the HP access control point at elevation 306' of the Control Building caused congestion and delays in processing personnel in and out of the radiological areas. To alleviate this problem, a hallway was added for exiting the HP access control point separate from the hallway used for entering.

- Laboratory space limitations made it difficult for the Chemistry Department to perform water chemistry work. Thus, the secondary side chemistry lab was enlarged to include a portion of the existing hot instrument repair shop.

- The hot instrument repair shop was relocated to a prefabricated in-plant type office structure in the Chemical storage Building located at the 331' elevation of the Auxiliary Building.

- The lunch room area for radiological controls workers was enlarged by utilizing the remaining portion of the hot instrument repair shop.

Safety Evaluation Summary: There are no SAR or Technical Specification safety margins that were affected by this modification. None of the above modifications are nuclear safety related or required by the engineered safeguards system. Therefore, failure of any added component has no affect on the ability to shut the plant down safely. Failure of the HEPA filters in the HVAC system for the new hot instrument repair shop could result in a release of radioactivity to the environment. However, the resultant quantity released would be the same as would be released upon failure of the HEPA filters in the HVAC system for the previous hot instrument repair shop. The amount of radioactivity released by this accident condition would be a small fraction of the limits set forth in 10 CFR 100, and would also be far below the design basis accident release amounts previously evaluated in the SAR. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Turbine Building South Addition (Operations Office Building) Domestic Water and Fire Service Tie-ins (BA 418749 / JOs 22382, 23676, 24653, 23636, 27883 and 28170)

Description Of Modification: This modification provided Fire Service water and Domestic water System connections for the Turbine Building South addition. This facility is located on the south side of the Unit 1 Turbine Building within the protected area. The Fire Service water connection provides a supply line for the installed sprinkler systems (dry pipe spronkler and hose standpipe). The domestic water connection provides a cold water supply source for the plumbing in the new addition.

Safety Evaluation Summary: The probability of occurrence or consequences of an accident or malfunction of equipment Important to Safety previously evaluated in the SAR is not increased. The tie-in of the fire service yard main to the Turbine Building south addition did not impact the design capability of the yard fire main. The modification did not increase the probability of occurrence or the consequence of a malfunction of equipment since the new components are of equal or better quality and the modification was performed per the original design codes. Additionally, the Post Indicator Valve (PIV) in the fire system permitted the isolation of the branch line without impacting the balance of the system. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Replacement of the 230 KV Digital Fault Recorder (BA 418787 / JO 40199)

Description of Modification: The existing 230 KV substation Hathaway Auto Fault Recording System was replaced by three Hathaway Digital Fault Recorders (DFR). Each DFR is comprised of a Data Acquisition Unit (DAU), an independent printer unit, alarm relay assembly and fast transient filter assembly. Each DAU is capable of recording 16 analog and 32 digital event parameters. Each DFR continuously scans all analog and event inputs at a preset rate, continuously overwriting old data. When triggered the system stores additional data resulting in acquisition of prefault, fault and postfault data. The DFRs are capable of providing critical chronological information such as location, time, nature and severity of faults along with the clearing time of protective devices and magnitudes of voltage and current. This data is essential in evaluating equipment performance and analyzing system stability.

The new DAUs monitor substation and transmission line, main generator, and main and auxiliary transformer voltage, current and protective devices.

Safety Evaluation Summary: Connection of the new digital inputs and transfer of the existing analog inputs to the DFRs did not affect the operation of any safety related equipment or systems. Safe plant operations have been improved by the ability to provide Operations and Engineering with a more effective means of evaluating equipment performance and analyzing system stability. The modification was a passive non-controlling addition which monitors the affected system's conditions and alarms.

The modification was classified as Regulatory Required due to the need to penetrate fire barriers. The fire barriers were resealed in accordance with approved plant procedures and the Fire Hazards Analysis Report. A fire hazards analysis has been performed to evaluate the combustibles added by the new cable. There was no adverse affect on nuclear safety or safe plant operations. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Elimination of Deep Stuffing Boxes (CMR 0585M)

Description Of Modification: This modification reduced the number of packing rings in valves having deep stuffing boxes to five rings. To accomplish this, the carbon rings were inserted into the stuffing box chamber to effectively reduce the box depth. The valves were repacked using braided graphite wiper rings top and bottom and laminated graphite rings in between.

Safety Evaluation Summary: The originally installed packing leak-off configuration sacrificed reliable packing performance for the ability to collect leakage. The elimination of the leak-off provision and the reduction of the amount of packing along with live-loading improved the reliability of the packing thereby reducing the chance of leakage and improving maintainability of the valve packing. Therefore, this change did not increase the probability or consequences of an accident previously evaluated in the SAR. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Removal of Caustic Piping to CA-T2 (CMR 89-119)

Description of Modification: The modification removed accessible piping between the 1WT area in the Turbine Building and CA-T2 in the Auxiliary Building. The piping was abandoned as a caustic flow path because of a history of clogging, thermal damage and leaks for which no method for maintaining operability and reliability could be found. Liquid caustic is added by direct pumping of caustic drums to CA-T2.

Safety Evaluation Summary: Evaluation identified no adverse affect on nuclear safety or safe plant operations since the safety analysis did not assume caustic supply from CA-T2. The SAR assumed margin of safety was not reduced since the supply of caustic from CA-T2 post accident is not assumed in the Safety Analysis. The elimination of one of the non-safety grade methods of delivering caustic to the tank does not reduce the margin of safety. Neither an increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR nor a possibility of an accident or malfunction of a type different than previously evaluated was created by the modification. The piping was not classified as important to safety and not associated with equipment that was important to safety. As a result of the piping demolition, a more reliable method of caustic addition was provided. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Replacement of RM-A5 and A15 Detectors (CMR 90-038)

Description of Modification: The Mylar windows on the beta scintillation detectors used in RM-A5 and A15 were subject to chemical attack from the ammonia hydroxide levels in the Condenser Offgas. Deterioration of the protective Mylar film eventually resulted in damage to the scintillation disk and affected detector response. On the recommendation of the detector vendor, titanium end window detectors were installed for this application.

Safety Evaluation Summary: The function of the detectors is to monitor main condenser vacuum exhaust noble gas activity in order to detect and provide indication of leakage between the primary and secondary systems. Estimates indicate that sensitivity to Xe-133 was maintained. With the exception of the detector end window film, all functional, physical and electrical characteristics of the replacement detector are identical to the original. Evaluation identified no adverse affect on nuclear safety or safe plant operations.The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, did not create the possibility of an accident or malfunction of a type different than previously evaluated and did not reduce the margin of safety defined in the basis of the Technical Specifications. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Installation of Solar Induced Current Recorder (CMR 90-068)

Description of Modification: A geomagnetic current (GIC) recorder was installed and connected to the 1B main transformer to monitor the magnitude of geomagnetically induced current resulting from solar magnetic disturbances(i.e., sunspots). The purpose of this recorder is to supply information to monitor the effects of current induction on the main step-up transformers and to determine if geomagnetically induced currents are going to ground via the high side neutral of the TMI main transformers.

Safety Evaluation Summary: The GIC recorder is a monitoring device only, and does not tie-in to any control/interlock circuits. Since the recorder is only a monitoring device, operating on a BOP power source, its operation does not adversely affect nuclear safety. This modification did not constitute an Unreviewed Safety Question.

****

Modification: WT-P-33A/B Suction Pressure Switch (CMR 90-101)

Description of Modification: Removal of the pressure switch (WT-PS-1355) from the suction of Lubrication Water Pumps WT-P-33A and B was accomplished to reduce inadvertent tripping of the operation pump. Pressure excursions of the filtered water supply exceed the margins of the pressure switch setpoint and can not be corrected by setpoint adjustment.

Safety Evaluation Summary: Evaluation of the removal of the pressure switch on the River Water Pumps Lubrication System (RWPLS) found no adverse affect on nuclear safety and safe plant operations. The function of the RWPLS was maintained without the susceptibility for inadvertent lube pump trips due to filtered water supply transients. Operability was found to be improved by the modification and the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR were not increased. Lack of interface with other plant systems established that the possibility of an accident or malfunction of a type different than previously evaluated was not created. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Reactor Building Fire Detection System (CMR 90-110)

Description of Modification: Thermal Heat Detectors replaced the ionization detectors in the Reactor Building D-Ring zones 2, 7 and 8.

Safety Evaluation Summary: The change of equipment did not affect the overall ability of the fire detection system to sense a fire and provide remote annunciation. For this reason, nuclear safety and safe plant operations are unaffected by the modification. The change to detectors operating on a different principle did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Likewise, the possibility of an accident or malfunction of a type different than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Containment Isolation Test Connection Upgrade (CMR 90-154)

Description of Modification: The modification added Local Leak Rate Testing (LLRT) test connections outside the Reactor Building to maximize Operating Personnel efficiency to drain, vent and test the respective system penetrations and support establishment of the LLRT configuration.

Safety Evaluation Summary: The new capped test connections are passive components during plant operations and fulfill their safety function as isolation barriers. Installation of the test connections did not adversely affect nuclear safety or safe plant operations. The test connections are within the test boundary for containment isolation valves that are periodically tested in accordance with 10 CFR 50 Appendix J. They are thereby maintained leak tight and no margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR since the test connections maintain a passive pressure boundary during Power Operations or when Refueling Containment Integrity is required. Similarly, the possibility of an accident or malfunction of a type different than previously evaluated was not created. This modification did not constitute an Unreviewed Safety Question.

****

IV. Temporary Mechanical Modifications, Lifted Leads, and Jumpers

Modification: WDL-V-275 Blocked Open (TMM 1)

Description of Modification: Valve WDL-V-275 was blocked open while the 1M 480 Volt switchgear was removed from service. This was done to maintain the BWST on cleanup while the switchgear was deenergized.

Safety Evaluation Summary: WDL-V-275 performs no safety function. The integrity of the Rad Waste system was not challenged by the temporary modification. The flow path can be isolated without removing the temporary modification if necessary. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: WDL-V-277 Blocked Open (TMM 3)

Description of Modification: Valve WDL-V-277 was blocked open while the 1M 480 Volt switchgear was removed from service. This was done to maintain the BWST on cleanup while the switchgear was deenergized.

Safety Evaluation Summary: WDL-V-277 performs no safety function. The integrity of the Rad Waste system was not challenged by the temporary modification. The flow path can be isolated without removing the temporary modification if necessary. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Open and Hold Open AH-D-8A (TMM 4 and 7)

Description of Modification: The normally locked closed damper AH-D-8A was manually opened and held open to provide make-up air to assist in increasing ventilation flow to the Intermediate Building for 322' elevation for heat removal.

Safety Evaluation Summary: Since the damper is a manual damper with no interface with any other system or controls, the action taken had no adverse affect on any safety related system or controls. The damper was returned to its normally closed position prior to operating the Reactor Building Purge System. It was determined that there was no impact on nuclear safety or safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Installation of a Temporary Valve Downstream of CA-V-1 and 3 Due to Leakage Past the Existing Valves (TMM 7 and 8)

Description of Modification: The modification installed more leak tight tubing manual needle valves downstream of both motor operated isolation valves (CA-V-1 and 3) since it was believed that leakage past the valves was affecting total gas sampling analysis results.

Safety Evaluation Summary: The temporary tubing isolation valves did not compromise the operation of the Primary Sampling System, containment isolation provisions or any other safety related system or component. Though the containment isolation valves exhibit excessive sampling leakage at RCS pressure, they acceptable seat leakage and perform their low pressure containment integrity function. This was verified to have during the 9R outage. There was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: CA-V-145B Blocked Open (TMM 9)

Description of Modification: Valve CA-V-145B was blocked open while the 1M 480 Volt switchgear was removed from service. This was done to maintain the BWST on cleanup while the switchgear was deenergized.

Safety Evaluation Summary: CA-V-145B performs no safety function. The integrity of the Rad Waste system was not challenged by the temporary modification. The flow path can be isolated without removing the temporary modification if necessary. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: CA-V-202B Blocked Open (TMM 11)

Description of Modification: Valve CA-V-202B was blocked open while the 1M 480 Volt switchgear was removed from service. This was done to maintain the BWST on cleanup while the switchgear was deenergized.

Safety Evaluation Summary: CA-V-202B performs no safety function. The integrity of the Rad Waste system was not challenged by the temporary modification. The flow path can be isolated without removing the temporary modification if necessary. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Tie In Connection Between Instrument Air Compressor IA-P-4 and the Service Air System (TMM 11)

Description of Modification: A tie in connection was attached to the outlet of the 1A Instrument Air dryer at the service connection to provide the capability to supply oil-free and CO/CO_2-free breathing air. The connection to Service Air was made at a hose connection at SA-V-83.

Modification: Installation of a Temporary Valve Downstream of CA-V-1 and 3 Due to Leakage Past the Existing Valves (TMM 7 and 8)

Description of Modification: The modification installed more leak tight tubing manual needle valves downstream of both motor operated isolation valves (CA-V-1 and 3) since it was believed that leakage past the valves was affecting total gas sampling analysis results.

Safety Evaluation Summary: The temporary tubing isolation valves did not compromise the operation of the Primary Sampling System, containment isolation provisions or any other safety related system or component. Though the containment isolation valves exhibit excessive sampling leakage at RCS pressure, they acceptable seat leakage and perform their low pressure containment integrity function. This was verified to have during the 9R outage. There was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: CA-V-145B Blocked Open (TMM 9)

Description of Modification: Valve CA-V-145B was blocked open while the 1M 480 Volt switchgear was removed from service. This was done to maintain the BWST on cleanup while the switchgear was deenergized.

Safety Evaluation Summary: CA-V-145B performs no safety function. The integrity of the Rad Waste system was not challenged by the temporary modification. The flow path can be isolated without removing the temporary modification if necessary. This temporary modification did not constitute an Unreviewed Safety Question

****

Modification: CA-V-202B Blocked Open (TMM 11)

Description of Modification: Valve CA-V-202B was blocked open while the 1M 480 Volt switchgear was removed from service. This was done to maintain the BWST on cleanup while the switchgear was deenergized.

Safety Evaluation Summary: CA-V-202B performs no safety function. The integrity of the Rad Waste system was not challenged by the temporary modification. The flow path can be isolated without removing the temporary modification if necessary. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Tie In Connection Between Instrument Air Compressor IA-P-4 and the Service Air System (TMM 11)

Description of Modification: A tie in connection was attached to the outlet of the 1A Instrument Air dryer at the service connection to provide the capability to supply oil-free and CO/CO_2-free breathing air. The connection to Service Air was made at a hose connection at SA-V-83.

Safety Evaluation Summary: The modification was evaluated and it was determined that the probability of occurrence or consequences of an accident or malfunction previously evaluated in the SAR was not increased. There was also no accident or malfunction of a different type than previously evaluated being created by the modification. No margin of safety was reduced. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Temporary Pressure Gauge Installation (TMM 13)

Description of Modification: PT 217 yields unreliable indication by either failing low or reading high. A temporary gauge was installed to provide true indication until a work order is completed to replace PT 217.

Safety Evaluation Summary: The ability to obtain more reliable pressure indication by temporarily installing a gauge at PT 217 was not considered an adverse affect on plant safety. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Rerouting EFW Flow from EF-V-27B (TMM 17)

Description of Modification: The discharge flow from EF-V-27B was temporarily rerouted to the RB Tendon Access Gallery through the addition of 2" piping to support the full ΔP MOV testing of EF-V-2A/B.

Safety Evaluation Summary: The modification was evaluated and it was found that there were no credible failure modes that impacted the EFW System or its piping. The temporary piping was restrained and flow existed only during valve testing. No adverse impact on plant operations was identified. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Prevention of MU-V-3 Closure on Intensifier Failure (TMM 29)

Description of Modification: An spool piece consisting of "P" tubing with a rating greater than 100 psig was run from IA-V-837 to the MU-V-3 solenoid valve to prevent inadvertent closure of MU-V-3 on the failure of the air intensifier.

Safety Evaluation Summary: The temporary modification did not reduce the ability of MU-V-3 to perform its intended function per the FSAR for resin overheating or containment isolation since its required position in both cases is closed and on a loss of air it fails closed. The temporary modification allowed MU-V-3 to be remotely opened and closed from the control room to provide letdown control. The temporary modification did not change the system component description as defined in the FSAR. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Terminate a Tubing Air Leak to CO-V-7 (TMM 31)

Description of Modification: The temporary modification placed a Tygon tube patch over the copper tubing air supply to CO-V-7. The modification was accomplished because the existing leak was un-isolable without affecting major plant equipment.

Safety Evaluation Summary: Failure of the temporary Tygon tubing patch was determined to have no affect on the accidents analyzed in the FSAR since the air supply tubing and CO-V-7 have no safety function. It would have led to an increased consumption of instrument air only. There are no Technical Specifications LCOs associated with CO-V-7 or its air supply. There was no adverse impact on safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Duct installation Following Replacement of AH-D-2 (TMM 32)

Description of Modification: An 18 inch section of 1" Fire Protection pipe and a sprinkler head were removed and the connection plugged to alleviate interference with the installation of duct work for AH-D-2. The pipe and sprinkler head were replaced upon completion of the duct work.

Safety Evaluation Summary: The temporary modification had no affect on nuclear safety or the ability to safely shutdown or monitor a shutdown in the event of a fire since the administrative controls on combustibles and fire watches were implemented. The particular sprinkler head involved provides no protection for class 1E circuit separation therefore, its temporary removal had no affect on circuit separation. Restoration was controlled by existing approved procedures and system operability verified. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Temporary Replacement of the Miscellaneous Waste Evaporator Rupture Disk with a Blank Flange (TMM 1)

Description of Modification: The rupture disk on the Miscellaneous Waste Evaporator was temporarily replaced with a blank flange during the performance of a Helium leak check of the Miscellaneous Waste Evaporator.

Safety Evaluation Summary: The leak check was performed with the component out of service. The rupture disk was replaced on completion of the Helium leak check. It was determined that there was no adverse affect on nuclear safety or safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Open WDL-V-291B with a Temporary Air Supply During Helium Leak Check of the Miscellaneous Waste Evaporator (TMM 3)

Description of Modification: A 0 to 15 psig air jumper was installed on WDL-V-291B positioner to maintain the valve open during the Helium leak check of the Miscellaneous Waste Evaporator.

Safety Evaluation Summary: The evaporator was out of service during the performance of the leak test. No releases were in progress and the WDL system boundary was not modified as a result of this temporary modification. An evaluation determined that there was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Installation of a Temporary Recorder to Monitor a Makeup Level Transmitter Signal (TMM 34)

Description of Modification: Connect a temporary test recorder on the output of LT-778 I/E converter and LT-778B isolator within the signal conditioner cabinet to monitor MU-LT-778 level signal for the purpose of determining the source of the spike that occurred on 1/10/91.

Safety Evaluation Summary: Evaluation of the temporary modification determined that the probability of an accident was not increased since failure of the instrument loop during troubleshooting would not cause a design basis accident. The Makeup Tank level indication are not used in the mitigation of accidents. There are no control interlocks associated with the level detection loop. There are no Technical Specifications associated with the level instrumentation. Therefore there was no adverse affect on nuclear safety or safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Removal of the Disk and Disk Guide from WDL-V-365 (TMM 50)

Description of Modification: The disk and disk guide were removed from WDL-V-365 due to the significant exposure involved in repair to prevent restriction of the associated line flow.

Safety Evaluation Summary: Waste water from the miscellaneous waste storage tank is transferred through WDL-V-365 to the miscellaneous waste evaporator. The lift check valve provided passive assurance that back flow from the miscellaneous waste processing line would not interfere with other waste processes due to an error in the valve lineup. Back flow isolation is provided by other system valves. The temporary modification was determined to have no adverse affect on nuclear safety or safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Secondary Chemistry Lab Floor Drain Plugs (TMM 61 to 63)

Description of Modification: Install 4" inflatable plugs into three of the floor drains in the Secondary Chemistry Lab Sampling Room to prevent radioactive contaminates from entering the sewage system.

Safety Evaluation Summary: The three floor drains described drain to the sewage collection system. The potential exists for secondary plant condensate to contain trace amounts of radioactivity. Although the sample racks drain to the sample collection tank, some spillage does occur. Trace amounts of activity conveyed to the sewage plant via the floor drains concentrates in the sewage sludge. The temporary modification has precluded the possibility of contaminates from concentrating in the sludge by preventing their entry into the sewage system. There was no adverse affect on nuclear safety or safe plant operation as a result of this temporary modification. Therefore, this temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Installation of a Whittaker Dissolved Hydrogen Analyzer (TMM 67)

Description of Modification: A Whittaker Dissolved Hydrogen Analyzer was installed to provide a comparison analysis result for the current stripping technique currently utilized.

Safety Evaluation Summary: The modification did not affect safe plant operation. The equipment was designed to meet RCS temperature and pressure specifications. Only the sampling valve lineup was affected by the installation of the new equipment. The valve lineup change was administratively controlled. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Removal of the Chlorine Detection System from Service (IL 1 through 4)

Description of Modification: Use of Chlorine was discontinued at the site. The chlorine detection system probes were deselected for service from IC-145 and the probes removed.

Safety Evaluation Summary: The only remaining use of bulk chlorine at the TMI site is at the sewage treatment plant via 150 pound containers. These containers are below Technical Specification 3.5.6 limits and are located in the sewage treatment plant which is greater than 100 meters from the air intake structure. As a result the Chlorine Detection System is no longer required to be operable. There was no adverse affect to safe plant operation resulting from this modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Open WDL-V-258B and 259B to Permit Helium Leak Check of the Evaporator Boundary Without the Feed Pump Operating (J 7)

Description of Modification: Jumpers were installed at terminal blocks of the Miscellaneous Waste Evaporator feed motor starter to allow the opening of WDL-V-258B and 259B to perform a Helium leak check of the evaporator boundary without the feed pump operating.

Safety Evaluation Summary: The Miscellaneous Waste Evaporator was out of service while the leak test was performed. No safety systems were bypassed or made inoperable. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Installation of a Manual Transfer Pushbutton on Inverter 1E (J 13 and 50)

Description of Modification: A momentary normally open pushbutton was installed on the Inverter 1E static switch to provide a safe means to transfer Inverter 1E back to ATB from TRB.

Safety Evaluation Summary: Installation of the pushbutton on Inverter 1E would not prevent its transfer from the alternate source (TRB) in the event of inverter failure or an overcurrent/undervoltage condition. Inverter 1E does not support a vital bus and it is not safety related. There was no adverse affect on safe plant operation that resulted from installation of the jumper. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Operate AH-E-6A Without AH-E-7 Running (J 15)

Description of Modification: The jumper bypassed the control logic for AH-E-7 and allowed the operation of AH-E-6A without AH-E-7 operation. The fan was run to provide a limited makeup air supply.

Safety Evaluation Summary: The Reactor Building Purge was not in operation while the jumper was installed. There was no interaction with other plant systems and had no affect on safety related systems or their controls. There was no adverse affect on nuclear safety or safe plant operation as a result of this temporary modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Power the 1N Bus from the 1G Bus (J 39)

Description of Modification: A jumper was installed to allow the 480V switchgear 1N to be powered from 480V switchgear 1G with the 1N Feeder Breaker 1N-02 open.

Safety Evaluation Summary: The plant was shutdown when the jumper was installed. All work completed was performed on the balance of the plant components with no ties to safety related systems or components. Normal power flow is from the 1N to the 1G bus, however, it was acceptable to have 1G as the source during outage work. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Open WDL-V-294B, 295B and 556B to Permit Helium Leak Check of the Evaporator Boundary Without WDL-P-22A Operating (J 75)

Description of Modification: Jumpers were installed at terminal blocks to allow the opening of the valves referenced to form a Helium leak check of the evaporator boundary without the pump operating.

Safety Evaluation Summary: The Miscellaneous Waste Evaporator was out of service while the leak test was performed. No safety systems were bypassed or made inoperable. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Open WDL-V-296B, 297B and 557B to Permit Helium Leak Check of the Evaporator Boundary Without WDL-P-22B Operating (J 76)

Description of Modification: Jumpers were installed at terminal blocks to allow the opening of the valves referenced to form a Helium leak check of the evaporator boundary without the pump operating.

Safety Evaluation Summary: The Miscellaneous Waste Evaporator was out of service while the leak test was performed. No safety systems were bypassed or made inoperable. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: AH-E-9A to Provide Penetration Cooling (J 79)

Description of Modification: A jumper was installed across AH-TS-706A to allow continued operation of AH-E-9A for penetration cooling while adjusting the setpoint of AH-TS-706A to prevent spurious shutoff of the fan without the presence of a fire.

Safety Evaluation Summary: Adding a jumper to the referenced temperature switch allowed operation of the Penetration Cooling system. High temperature indication was still provided by existing alarms. The temporary jumper did not reflect a condition not analyzed in the FSAR. This temporary modification did not constitute an Unreviewed Safety Question.

****

Activities For Which A Safety Evaluation Was Performed
But Did Not Impact SAR Systems/Components

Modification: CA-V-13 Gear Change (BA 123220 / JO 7382)

Description of Modification: The gear ratio of CA-V-13 was increased. The higher ratio supplies additional mechanical advantage to the motor reducing power consumption for a given unit of work output. The modification was made to compensate for increased loading resulting from installation of new packing material.

Safety Evaluation Summary: Installation of higher ratio gears on CA-V-13 did not adversely affect its safety function. The increased mechanical advantage of the motor did not adversely affect valve operation since the valve design was such that the increased load of stall conditions can be handled.

The modification did not reduce plant safety; therefore, the possibility of an accident or malfunction different from any type that had been previously evaluated has not been created. The margin of safety was not decreased nor did it represent a radiological safety concern. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Motor Operated Valve Closed Position Indication (BA 123251 / JO 30939)

Description of Modification: Improper close position indication that resulted in high cooldown rates as documented by IEN 86-29 and recommendations in INPO SOER 86-2 initiated this modification. It provided positive ES close indication on eight motor operated valves. Two valves that did not have ES indication were modified to include it.

Safety Evaluation Summary: Placement of ES closing indication on a different rotor allowed positive close indication for the modified operators. The problem was generic to all Limitorque motor operators as suggested in SOER 86-2. Rewiring of the close indicator circuit did not adversely affect either the normal operation or safety function of the Limitorque operator and doing so provided positive close indication in the Control Room for plant operation.

The modification did not affect nuclear safety nor have an affect on the environment. An Unreviewed Safety Question as defined in 10 CFR 50.59 was not found to exist.

****

Modification: Use of B&W Rolled Plugs and Ribbed Plugs in the TMI-1 OTSGs (BA 123275 / JO 34948)

Description of Modification: The tube plugging was performed to remove from service certain TMI-1 OTSG tubes through use of B&W rolled and ribbed plugs. The plugs were used in defective tubes to establish an adequately leak tight pressure boundary between the OTSG primary and secondary sides. Inconel 690 alloy were used.

Safety Evaluation Summary: Defective OTSG tubes were plugged as required by the established plugging criteria. Since the plugs met the same criteria as the remainder of the RCS pressure boundary, the probability and consequences of any analyzed accident occurring has not been increased. For the same reason, the margin of safety has not been decreased. B&W ribbed and rolled plugs comply with the applicable codes and standards governing safe operation of the plant. Testing demonstrated that the tube/plug joint met the leak-tightness requirements of the TMI-1 OTSG functional specifications.

The modification did not involve a radiological safety concern or a change to the plant environmental interfaces since the primary-to-secondary leakage was demonstrated to be a fraction of the total leakage assumed in the basis for the plant Technical Specifications. Consequently, the release of radioactivity will remain far below established limits since the pressure boundary will remain intact under normal, transient and accident conditions. This modification did not constitute an Unreviewed Safety Question.

****

Modification: RMS Ratemeter Backpanel (BA 128086 / JO 9844)

Description of Modification: This modification installed a jack/plug wiring arrangement on the RMS ratemeters. The majority of the Victoreen model 842 and 846-1 ratemeters have terminal strips on back of the ratemeters for making field connections. Because of the terminal strip connections and some maintenance activities, the wiring had shorted to ground and/or adjacent wiring, which caused loss of a vital bus. A jack and plug arrangement were added to the back of the ratemeters. The existing terminal strips remain and are used to connect the ratemeter to the new plug/jack.

Safety Evaluation Summary: The margin of safety as defined in the basis for any Technical Specification was not reduced. No RMS setpoints or interlocks were changed as a result of this modification. The implementation of this modification did not adversely affect nuclear safety or safe plant operations. The ratemeters affected by this modification are not nuclear safety related. Installation of plugs and jacks did not alter the operation of the ratemeter. The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased. The installation of the plugs and jacks did not alter the interlocks between the RMS and equipment used to mitigate the consequences of a radiation release. The modification did not affect any post accident monitors. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Industrial Waste Treatment System (IWTS) and Industrial Waste Filter System (IWFS) Nuisance Alarms (BA 128112 / JO 28450)

Description of Modification: As designed any alarms on the local IWTS annunciator also alarmed on control room annunciator PRF 7-2, "IWTS TROUBLE." Additionally, the local IWFS annunciator also alarmed on control room annunciator PRF 8-2, "IWFS TROUBLE." To reduce operator burden the IWTS and IWFS annunciator logic was modified. This modification resulted in PRF 7-2 alarming when an IWTS local alarm is received. When PRF 7-2 is acknowledged in the Control Room the control room alarm light will stop flashing and remain visible. After the alarm is acknowledged locally the PRF 7-2 alarm light will extinguish. The sequence will be repeated for any additional alarm that is required to alarm locally and in the Control Room. Similar changes will be made to PRF 8-2.

Safety Evaluation Summary: The implementation of this modification did not adversely affect nuclear safety or safe plant operations. The IWTS and IWFS are not Nuclear Safety Related (NSR) nor do they interlock with any NSR systems. When a control room alarm is received from either IWTS or IWFS an auxiliary operator will be sent to the IWTS building to investigate the alarm. If an operator cannot be sent to the building to acknowledge the local alarm any additional alarms will still alarm in the Control Room. Those alarms that have been deleted from input to the control room alarms have been reviewed by Plant Operations, Human Factors, and Engineering. These alarms were considered to be non-critical and not requiring immediate operator response. The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased. The IWFS and IWTS do not perform a safety related function and are not addressed in Chapter 14 of the SAR. This modification did not constitute an Unreviewed Safety Question.

****

Modification: D/G KW Meter Replacement (BA 128112 / JO 34082)

Description of Modification: The modification replaced the existing analog KW meter of each diesel generator, EG-Y-1A & 1B. The power outputs from the DGs as provided to the Control Room by the analog wattmeters had an accuracy of approximately 2%. There was also potential reading error due to parallax. Since the DGs are expected to be loaded near capacity during the worst case accident, the DGs could have been overloaded inadvertently beyond the manufacturer's approved rating if the meter was reading lower than the actual load. In order to improve the accuracy of the DG meter, the existing meters were replaced with digital meters with combination bargraph/digital displays. The replacement of the KW meter improved the accuracy of the digital meter from 2% to less than 0.3%.

Safety Evaluation Summary: Technical Specifications were not affected and no performance degradation resulted from the modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Diesel Generator Metering Circuits (BA 128112 / JO 34324)

Description of Modification: The modification consisted of connecting each TMI-1 DG's current, watt and var signals to the plant computer and installing new metering transducers for DG watt and var measurement. Sliding link terminal blocks were installed to simplify maintenance.

Safety Evaluation Summary: The modification to the Diesel Generator metering circuits did not affect nuclear safety nor did it affect the environment. An Unreviewed Safety Question as defined in 10 CFR 50.59 was not found to exist.

****

Modification: Reactor Head Vent Valve Connector Replacement (BA 128112 / JO 36176)

Description of Modification: The modification replaced the Patel connectors and cables associated with the Reactor Vessel vent valves RC-V-42 and 43 and DPT-1081. The existing Patel connectors were highly susceptible to thread galling when disconnected and reconnected. The problem was eliminated by installation of new style quarter-turn Patel connectors.

Safety Evaluation Summary: The installation of quick disconnect electrical connectors enhances refueling operations. The safety functions of the Reactor Vessel head vent valves were not affected. The modification did not adversely affect the safety and health of the general public. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Class 1E Diesel Generator Up-To-Voltage and Overload Relay Replacement (BA128112 / JO36261)

Description of Modification: The modification increased the reliability of the emergency diesel generator and facilitated relay calibration. Westinghouse type CV-7 up-to-voltage and BL-1 thermal overload electro-mechanical relays were replaced by Brown Boveri type 59N and 49 solid-state relays respectively in the diesel generator local control panels.

Safety Evaluation Summary: Replacement of the up-to-voltage and overload relays was completed to increase reliability. There is no affect on the operation of the emergency diesel generators and therefore nuclear safety and safe plant operations were not affected. The modification was evaluated and found not to increase the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR or an accident or malfunction of a different type. No radiological safety concerns exist since the circuitry does not control or monitor radiological releases. This modification did not constitute an Unreviewed Safety Question. Similarly, there is no affect on the environment.

****

Modification: HSPS Low Level Alarm Logic (BA 128112 / JO 37565)

Description of Modification: The modification limits unnecessary operator intervention in FW control post-trip, caused by an outlying OTSG start-up signal. The existing alarm logic was converted from a 1 out of 4 to a 2 out of 4 logic. This made the alarm consistent with the 2 out of 4 automatic EFW initiation logic.

The modification required installation of logic modules and rewiring of the OTSG A and B start-up level signals to those gates. Test switches were installed to permit testing of the new alarm logic.

Safety Evaluation Summary: The modification to the HSPS OTSG low level alarms did not affect nuclear safety nor the environment. No Technical Specification change is required as a result of the modification. An Unreviewed Safety Question as defined by 10 CFR 50.59 did not exist.

****

Modification: ES Manual Actuation Reset Pushbutton Replacement (BA 128149 / JO 41863)

Description of Modification: Quarterly performance of ES surveillance resulted in several inadvertent ES actuations. They were due to the malfunction of the holding coils in the existing pushbuttons and operator error (premature release of a pushbutton prior to resetting manual actuation relays). The modification replaced the existing momentary pushbuttons with maintained contact pushbuttons and eliminated PB8. The new pushbuttons were alarmed to preclude the possibility of being left in the DEFEAT position. Two new ENABLE/DEFEAT bushbuttons were added to the 30 psig manual actuation circuit and indications were changed to accurately display the actual condition of the pushbuttons.

Safety Evaluation Summary: The modification was installed to eliminate or reduce the potential for component failure and operator error during the performance of ES testing. Evaluation of the modification found no nuclear safety or safe plant operations concerns to exist since: the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased, the possibility of an accident or malfunction of a different type than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Rx Building "D" Ring Handrail Upgrade (BA 128149 / JO 38967)

Description of Modification: The minor structural modification restored personnel fall protection by restoring the handrail at the 365'6" elevation of the Reactor Building. The existing anchorage was degraded and required the installation of a structural steel angle to the top of the secondary shield walls with concrete expansion anchors and appropriate bolting.

Safety Evaluation Summary: The interior structure of the Reactor Building is unaffected in its ability to contain radioactive material that might be released from the reactor core as a result of a loss of coolant accident. The modification is classified Regulatory Required due to the requirements for seismic resistance and the installation of anchors in the Class I concrete. Although classified as Regulatory Required, the modification involved independent passive hardware changes and it has no adverse affect on nuclear safety or safe plant operations. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Reactor Coolant Pump Lube Oil System Upgrade (BA 412512 / JO 37222)

Description of Modification: The modification was made to correct oil leakage and difficulties experienced with filling the oil reservoirs. The modification upgraded piping and tubing to welded joints as much as possible to minimize leaks, installed redesigned local fill reservoirs and drip pans, installed remote fill stations for on-line oil addition capability and replaced existing oil level sight gauges and level indication transmitters.

Safety Evaluation Summary: The Reactor Coolant Pump (RCP) Lube Oil System lubricates the RCP motor bearings. The RCPs are not required for safe plant shutdown. Piping that contains lube oil requires seismic support to prevent a rupture during a seismic event that could result in oil contact with hot surfaces having the potential to start a fire. Lube oil system piping containing oil is also required to be within splash shields or over drip pans designed to route oil from random leaks to a collection tank. Use of welded fittings in the modification permitted NRC approval of an exemption from Section III.O of Appendix R to 10CFR50 for the remote fill piping which only contains oil during the addition process. Level indication instrumentation has no nuclear safety function. An interlock prevents starting an RCP with the oil level 2 inches below the normal level.

The RCPs and their associated lube oil systems are not part of the engineered safeguards systems. Work performed in the modification activity was in accordance with established system quality requirements and seismic classifications and complied with applicable regulations and Technical Specifications. Evaluation of the modification found no nuclear safety or safe plant operations concerns to exist since: the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased, the possibility of an accident or malfunction of a different type than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Liquid Waste Disposal System (LWDS) Annunciator Acknowledge Pushbutton (BA 412556 / JO 32937)

Description of Modification: The acknowledge pushbutton for LWDS annunciator is located on the LWDS panel. The location of the pushbutton required the operator to leave the main control console area to acknowledge a LWDS alarm. This modification installed an additional acknowledge pushbutton and mimic on the computer console. This pushbutton will allow the operator to acknowledge an alarm without leaving the main control console area.

Safety Evaluation Summary: The implementation of this activity did not adversely affect nuclear safety or safe plant operations. Safe plant operation is assured since the additional acknowledge pushbutton for LWDS annunciator will allow the operator to stay in the main control console area while acknowledging a LWDS alarm. Nuclear safety is not affected since the pushbutton and annunciator are not nuclear safety related equipment. The possibility for an accident or malfunction of a different type than as previously evaluated in the SAR was not created. The installation of the pushbutton did not affect the passive integrity of the computer console. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Control Room Alarm Enhancements (BA 412556 / JO 32938)

Description of Modification: Recommendations on restructuring the control room main annunciator alarm system organization identified by SPIP, contractor studies and an internal alarm working group were compiled in the modification. New alarms, those to be deleted, combined, relocated, and/or added to the plant computer were addressed by the modification. A microprocessor based annunciator system was installed to incorporate alarm logic changes for reformatting and simplify any future alarm reformatting. Although a Beta sequential events recorder replaced the existing Panalarm annunciator, the existing annunciator light boxes and cabling were maintained in service.

Safety Evaluation Summary: The modification was evaluated and found not to affect nuclear safety or safe plant operations. No environmental impact was identified. No unreviewed safety question was found to exist since: the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased, the possibility of an accident or malfunction of a different type than previously evaluated was not created. Since related Technical Specification alarms and equipment will be maintained, the margin of safety was not reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: 9R Underprotected Cable Fix Modification (BA 412561 / JO 28135

Description of Modification: The Underprotected Cable Study was initiated as a result of the Short Circuit and Coordination Review performed to assure compliance with 10 CFR 50, Appendix R. The in-depth follow-up evaluation recommended cable changeout and breaker replacement in cases where cables were not properly protected in accordance with their design ampacity or by their primary overcurrent device. A portion of the work was performed during 8R. This modification addressed the remaining cases of underprotected cables.

Safety Evaluation Summary: The safety function of the systems affected by the cable/breaker changeout was evaluated. The modification was designed with provisions to ensure that neither nuclear safety-related functions nor safe plant operations were adversely affected. Cable for the modification was routed entirely within seismic trays and conduit in Class I Structures. Fire barriers penetrated were resealed. The replacement breakers/overload heaters were adequately sized for the load conditions and system performance was found unaffected. Evaluation of the modification found no nuclear safety or safe plant operations concerns to exist since: the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased, the possibility of an accident or malfunction of a different type than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Reactor Coolant Pump 1A/1D Access Platforms and Main Steam Line Hanger Catwalks (BA 412578 / JO 33745)

Description of Modification: Platforms were installed at RCPs 1A and 1D and catwalks were installed at selected Main Steam line hangers to provide safe, permanent work areas of sufficient size to support projected maintenance activities on the components. The modification consisted of fabrication and erection of three platforms with grating, handrails, safety chains, toe plates and permanent ladder access. Relocation of lighting and service conduits was accomplished as needed.

Safety Evaluation Summary: The modification was evaluated and found to have no affect on nuclear safety or safe plant operation. The additional loading on the secondary shield walls was analyzed and found acceptable. The platforms and catwalks are passive components and were designed to maintain their structural integrity under all originally postulated loading conditions. Anti-falldown criteria were applied to the relocated lighting and service conduits. The modification did not affect the ability to assure containment integrity or the ability to bring the plant to a safe shutdown condition. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The possibility of an accident or malfunction of a different type than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Station Effluent Flow Computer Point (BA 412579 / JO 25538)

Description of Modification: The modification connected the station effluent flow instrument signal to the plant computer thereby allowing computer logging and calculation of plant discharge flow.

Safety Evaluation Summary: The modification was evaluated and found not to affect nuclear safety or safe plant operations. The station effluent flow instrumentation and plant computer are not nuclear safety related components. The new circuit does not interlock with any plant equipment. The circuit is passive and has no interface with any plant system. Therefore, there is no affect on nuclear safety or safe plant operations. No environmental impact was identified. No unreviewed safety question was found to exist since: the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased, the possibility of an accident or malfunction of a different type than previously evaluated was not created. Since related Technical Specification alarms and equipment will be maintained, the margin of safety was not reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: EFW Auto-Start ES Test Lights (BA 412579 / JO 28468)

Description of Modification: To facilitate testing of the emergency feedwater pump ES blocking and auto-start contacts, the modification eliminated the need to open sliding links and connect an ohmmeter for contact verification. The Group 2 blocking contact was changed to a Group 1 contact and an indicating light was installed to indicate the contact status. The light is normally illuminated and goes off while the blocking contact is open for block 1 loading and comes on again when the auto-start contact closes during block 4 loading.

Safety Evaluation Summary: Evaluation of the modification found that changing the Group 2 to a Group 1 ES blocking contact only affects ES testing and does not alter the circuit operation under actual ES conditions. The indication lights provide verification of contact operation during testing only. For the same reasons, safe plant operation and nuclear safety were not affected by the modification. Similarly, the modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The possibility of an accident or malfunction of a different type than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Installation of Shielding for DH-V-3 (BA 412579 / JO 28475)

Description of Modification: The purpose of this modification was to install permanent shielding for DH-V-3, located at elevation 281'-0" of the Auxiliary Building. Temporary shielding under the valve and local piping has been in-place due to hot spots on the line, valve and pipe nipple. Previous to installation of this temporary shielding the area was a "locked, high radiation area." Since the hot spot problem is recurrent, permanent shielding replaced the temporary shielding. The scope of this modification included the fabrication and erection of structural steel to support lead shielding underneath the 12" pipe.

Safety Evaluation Summary: The shielding and support structure have no nuclear safety function and are not required to function during or after a Safe Shutdown Earthquake but are required to maintain passive integrity. Therefore, they were classified as Seismic Class II, Anti-falldown. The added shielding and support fixtures were evaluated, designed and installed in accordance with appropriate criteria and the modification involves only passive hardware changes. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Floor Hatch Handrail, RB Elevation 346' (BA 412579 / JO 34050)

Description of Modification: The purpose of this minor structural modification was to provide personnel fall protection at the floor hatch at Elevation 346'-0", Reactor Building. The hatch is normally covered with grating and removable beams. During outages all or some of the grating and support beams are removed to enable movement of equipment and material between the operating floor and lower levels. Temporary railings were constructed as fall protection around the open hatch. This modification provided for the fabrication and installation of permanent handrail mounting brackets and fabrication of handrail sections to be erected as needed during plant outages.

Safety Evaluation Summary: This proposed activity had no affect on plant safety since the handrail support brackets constitute a negligible additional mass to the structure. Weldments performed in accordance with the GPUN Welding Program did not degrade the integrity of the concrete slab or supporting structural steel The support brackets and handrail perform no nuclear safety function and are not required to function during or after a SSE but are required to maintain passive integrity. This modification did not constitute an Unreviewed Safety Question.

****

Modification: RC-P-1 Seal Leakoff Flow Computer Points (BA 412579 / JO 34051)

Description of Modification: To aid in the analysis of RC pump seal failures the #1 wide range seal leak-off flow signals from MU9-FT 1,2,3,4 were connected to the plant computer. Cables were routed between NNI cabinet 0 and the multiplexer cabinet for the plant computer.

Safety Evaluation Summary: The implementation of this activity did not adversely affect nuclear safety or safe plant operations. Connecting the seal leak-off flows to the plant computer provides another means of monitoring the performance of the RC pumps. The plant computer and the #1 seal leak-off instrumentation are not part of any nuclear safety related systems. The probability of occurrence or the consequences of an Accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased since this modification improves the monitoring of the RC pump seal leak-off flow. The seal leak-off flow signals are not required for any Design Basis Accidents or provide any safety functions. The margin of safety as defined in the basis of any Technical Specification was not reduced. While the makeup system is extensively addressed in the Technical Specification there is no operability requirements for MU9-FT 1,2,3,4. No leak-off values are in the basis of any safety limits established by Tech Specs. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Radwaste Line Shielding at the Operations Control Office in the Aux Bldg. (BA 412579 / JO 34060)

Description of Modification: Lead shielding was fabricated and installed around a 2.5 inch diameter radwaste solidification line adjacent to the Operations Control Office in the Aux Building. This was done to reduce the long term radiation exposure to personnel stationed in the office.

Safety Evaluation Summary: The safety function of the shielded line is not affected by the installation of the shielding. The additional weight supported by the piping supports was analyzed and found acceptable. The shielding design is such that it will remain attached to the pipe during a seismic event. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR since system operation is not affected and integrity is maintained under all postulated loading conditions. The possibility of an accident or malfunction of a type different than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Radiation Shielding in the Makeup Valve Alley - Auxiliary Building (BA 412579 / JO 34061)

Description of Modification: This modification involved the following activities:

- Lead shielding was fabricated and installed around two (2) portions of the Makeup and Purification (MU&P) System lines located in the Makeup Valve Alley. The first portion was a 2-1/2 inch diameter elbow between valves MU-V-97B and MU-V-109B. The second portion was a horizontal run of 2-1/2 inch line upstream of valve MU-V-140 and extending to the wire mesh barrier at the northern end of the alley compartment.

Safety Evaluation Summary: Nuclear safety and safe plant operations were not adversely affected by this modification since there is no interaction between the shielding and any system during all modes of plant operation. The probability of an accident or malfunction of a different type than previously identified in the SAR is not created as the shielding is a passive component designed to maintain its integrity under all postulated loading conditions. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Industrialized Water Treatment (IWT) Conductivity Cells and Flowmeters Upgrade (BA 412511 / JO 5136)

Description of Modification: The caustic conductivity cell, C-7, was previously located in the IWT Sampling Sink. In order to improve the time response of the cell to changes in concentration, it was changed to an in-line cell. Additionally, this modification installed new flowmeters and indicators/totalizers.

Safety Evaluation Summary: The margin of safety as defined in the basis for any Technical Specification was not reduced. The conductivity sample points and the flowmeters are not defined in the basis of any Technical Specification safety limit. The new computing totalizers serve the same function as the previous flowmeters. The new conductivity cell remains connected to the existing conductivity monitor. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Install Racks for Temporary Shielding (BA 412579 / JO 34062)

Description of Modification: This modification installed permanent racks on which temporary shielding blankets may be hung during outages. The affected areas are located inside the Secondary Shield Wall (D-Ring) at the 281' level of the RB.The racks permit shielding the letdown line in the "B" D-Ring and the primary shield access gate.

Safety Evaluation Summary: The shielding racks perform no nuclear safety function and are not required to function during or after an SSE, but are required to maintain passive integrity. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Spent Fuel Handling Bridge Upgrades (BA 412580 / JO 26283)

Description of Modification: This modification encompassed the following activities:

- The hydraulic control rod mast system was replaced with a hoist driven mechanical system.
- Replaced the fuel assembly grapples on the fuel handling and control handling masts with grapples capable of handling a Mark B-4 fuel assembly.
- Replaced the motor controls with a variable frequency control system.
- Replaced the Dillon load sensing system with a programmable Sensotec Load cell and a remote digital readout in the control console.
- Replaced the control console on the Spent Fuel Handling Bridge (SFHB) with a console containing programmable logic including all safety circuits and status lights.
- Installed programmable geared limit switches on the SFHB.
- Installed a television positioning system on the SFHB.
- Installed fixed lighting on the SFHB.

Safety Evaluation Summary: The above modifications did not increase the probability of occurrence or consequence of a malfunction of "Regulatory Required" equipment. These upgrades increase fuel handling equipment reliability while decreasing equipment maintenance. The modifications did not create the possibility for an accident of a different type than any previously evaluated in the SAR since operating modes and system control were not altered. This modification did not constitute an Unreviewed Safety Question.

****

Modification: RCS Shutdown Level Indicator (BA 412589 / JO 31565)

Description of Modification: The modification has been designed to meet the requirements of NRC Generic Letter 88-17 and reduce the risk of loosing Decay Heat Removal (DHR) during reduced Reactor Coolant System inventory operation. A level transmitter independent of the existing loop, a digital display and an isolation module were installed and a double pen recorder replaced a single pen recorder.

Safety Evaluation Summary: The RCS Shutdown Level Indicator Modification was redundant to an existing instrument loop and acts to enhance the safety of the existing systems and reduce the risk of loss of DHR capability. Two independent RCS level indications increase the capability of continuously monitoring the DHR system performance.

The modification indicates plant status and has no direct effect on it. It was evaluated and found not to introduce any new accident or malfunction. It did not increase the likelihood of occurrence or consequences of any accident as analyzed in the TMI-1 UFSAR. The modification did not decrease the margin of safety as described in the Technical Specifications. Nuclear safety and safe plant operations were unaffected by the modification The modification was determined not to involve an Unreviewed Safety Question or a new environmental impact.

****

Modification: BIRO T/C Cable Replacement (BA 412590 / JO 32940)

Description of Modification: The Backup Incore Temperature Read Out (BIRO) system is a diverse and separate system to the plant computer for monitoring core exit temperature as required by NUREG 0737 and Reg. Guide 1.97. Sixteen incore thermocouples can be monitored, one at a time, at the BIRO select and display panel. Existing thermocouple conductors used in the BIRO were disconnected and spared. Sixteen extension cables and two spares were routed within the TMI-1 Reactor Building to electrical penetrations 201, 202, 311 and 312. The new thermocouple cabling interfaces with the newly installed environmentally qualified incore detector assemblies installed as part of a separate task.

Safety Evaluation Summary: The BIRO system thermocouple cable replacement modification was reviewed to determine its effect on safety. The installation of the BIRO thermocouple extension cables with Cannon connectors was classified as Regulatory Required. The modification did not affect Appendix R commitments since fire barriers and penetration boxes were resealed or restored to their original conditions in accordance with plant procedures. Evaluation identified no adverse affect on nuclear safety or safe plant operations. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The possibility of an accident or malfunction of a type different than previously evaluated was not created and no margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: 230 KV High and Low Voltage Computer Alarms (BA 412594 / JO 40201)

Description of Modification: In response to NRC Information Notice 84-02, which addresses the operation of nuclear plants at lower than analyzed voltage levels, a computer alarm for 230KV low voltage was added. This modification added a computer point for 230KV bus 4 and 8 voltage, with low alarm set for 230KV and high alarm set for 242 KV. To obtain the inputs for this computer point a voltage transducer was installed in the PCL panel, and connected to the bus 4 and 8 selector switch on the panel.

Safety Evaluation Summary: The addition of the transducer and computer alarm point did not affect the operation of any safety related systems. Providing Operations personnel with high and low voltage computer alarms will better prepare them to deal with high and low voltage grid conditions which could result in equipment damage. Therefore, safe plant operations and reliability were improved. Nuclear safety was not adversely affected because this computer circuit is not safety related and does not perform a safety function. Safe plant operations were not adversely affected because the new computer alarms will enable Operations personnel to more quickly identify and deal with potential high or low grid voltage levels that could cause damage to safety related equipment. This modification did not constitute an Unreviewed Safety Question.

****

Modification: OTSG Tube Sleeving (BA412597 / JO 35949)

Description of Modification: The modification installed 80 inch Alloy 690 mechanical sleeves in the lane wedge area OTSG tubes. These tubes have a high potential for high cycle fatigue failure. The Alloy 690 tubing material meets the basic requirements of ASME Code sections II and III, and Code Case N-474-1.

Safety Evaluation Summary: Evaluation identified no adverse affect on nuclear safety or safe plant operations since the tube sleeving will minimize the likelihood of a tube leak and tube rupture due to high cycle fatigue and also maintain the tube integrity as originally designed. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Rupture and tube collapse tests have demonstrated the structural integrity of the Alloy 690 mechanical sleeve as equivalent to the original tube. The possibility of an accident or malfunction of a type different than previously evaluated was not created. The only accident relevant is a single tube rupture event. Sleeve design criteria met Reg. Guide 1.121 requirements and an accident or malfunction of a different type is highly unlikely. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification since the tube plugging criteria remains at 40% through wall as defined in the TMI-1 Tech Specs per Reg. Guide 1.121 guidelines. The modification did not involve a radiological safety concern since the OTSG tube rupture analysis in the original SAR is not affected by tube sleeving. The modification did not involve an Unreviewed Safety Question.

****

Modification: 9R Foxboro Transmitter Replacement (BA 412603 / JO 38328)

Description of Modification: Eight OTSG Startup Level Foxboro transmitters were replaced because of drift outside the specified range and failure to provide the required accuracy for system usage. Preliminary Safety Concern 90-07 addressed the initiation of EFW on low OTSG level and the effects of larger errors causing false trips. Rosemount model 1154 were installed in place of the original Foxboro transmitters tagged LT-1042, 1043, 1046, 1047, 1050, 1051, 1054 and 1055.

Safety Evaluation Summary: Evaluation identified no adverse affect on nuclear safety or safe plant operations to exist as a result of the modification. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR since the replacement equipment installed was equal to or better than the original equipment. The possibility of an accident or malfunction of a type different than previously evaluated was not created since the transmitters were a one-for-one replacement which used existing impulse lines and circuits. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification. This modification did not constitute an Unreviewed Safety Question.

****

Modification: TMI Security System Upgrade (BA 418058 / JO 16951)

Description of Modification: This modification involved the replacement of the TMI Security computers, computer software, associated computer terminals/printers, and the access control field hardware.

Safety Evaluation Summary: This modification improves system performance, reliability, and increases speed of alarm responses. Neither nuclear safety or safe plant operations were affected by this modification. The replacement and previous computers operated during the change over to provide the required functionality. Appropriate security compensatory measures were provided during the temporary loss of required system functions. This modification did not constitute an Unreviewed Safety Question.

****

Modification: AH-E-14A Monorail (BA 418682 / JO 30945)

Description of Modification: This modification installed a monorail trolley system above the AH-E-14A fan/motor assembly to facilitate removal and reinstallation of the assembly for maintenance purposes.

Safety Evaluation Summary: This modification did not affect any regulatory documents, Technical Specifications, or the SAR. This modification did not adversely affect any safety related systems, structures, or components and did not constitute an Unreviewed Safety Question.

****

Modification: Control Building Chiller Modification (BA 418695 / JO 7407)

Description of Modification: This modification increased the reliability of the Control Building Chillers (AH-C-4A&B) by allowing the units to operate during periods of low-load on the chillers. The scope of this modification involved the installation of a hot gas bypass on each chiller.

Safety Evaluation Summary: This modification did not degrade the safety function of the Control Building Chilled Water System or the Control Building Ventilation System. The modification did not reduce the margin of safety defined in the SAP because the purpose of the modification was to improve the reliability of the chillers under any condition. The modification facilitates continued chiller operation under low load conditions without changing the function or overall performance. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Reactor Building Pressure Indication (BA 418702 / JO 36638)

Description of Modification: This modification installed a digital meter on control room panel center left (PCL) for indication of Reactor Building pressure. Plant Operations personnel indicated that a digital indicator for Reactor Building pressure would help during transient diagnostics and normal operation. The new meter will turn off when RB pressure is greater than ≈ 14.9 psig and turn back on when pressure drops below ≈ 14.8 psig.

Safety Evaluation Summary: This modification did not adversely affect nuclear safety or safe plant operations. The HSPS logic circuits for EFW initiation and FW isolation were not effected by this modification. Safe plant operation is maintained since indication of Reactor Building pressure is more readily available to the operator. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR was not increased. The modification did not add or delete any safety related control functions. The setpoint for EFW actuation on high Reactor Building pressure was not changed. The single failure criteria, channel independence and channel redundancy criteria assumed in the original design are maintained. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Removal of Emergency Diesel Generator (EDG) Fan Drive Lubricant Heater (CMR-89-159)

Description of Modification: This modification removed the existing heating elements for the EDG, and replaced them with two (2) 125 watt heating elements in the EDG gear drive oil sump. This was necessary to prevent the carbonizing of the lubricating oil by the previous heating element.

Safety Evaluation Summary: The EDGs remained operable, as required by the Tech. Specs., during this modification. The new heater has enough wattage to maintain 30°F without carbonizing the lubricating oil. Two (2) heaters were installed to ensure adequate safety margin for gear box pump operability. The electrical current of the replacement heater is less than that of the existing heater so there was no adverse affect on the electrical supply. Nuclear safety and safe plant operations were not adversely affected by this modification. Adequate oil heating capacity to maintain gear box to oil temperature was installed while reducing heater watt density to acceptable levels. By ensuring that gear box oil was maintained above its minimum required temperature, the operability and availability of the EDGs was not adversely affected. This modification did not constitute an Unreviewed Safety Question.

****

Modification: CRDM Replacement Gaskets (CMR 90-10)

Description of Modification: Due to CRD motor tube flange gasket leaks and the potential damage to the carbon steel reactor vessel head from boron, fourteen CRDM gaskets were replaced during 8R. The existing asbestos spiral wound gaskets were replaced with spiral wound stainless steel/graphite gaskets. The graphite gaskets are expected to provide better leak sealing performance than the existing gaskets.

Safety Evaluation Summary: The change did not affect any regulatory documents, Plant Technical Specifications, or the UFSAR. Although the CRDM pressure boundary flanges are classified as Nuclear Safety Related the associated gaskets perform no safety function. Therefore the change had no adverse impact on the function, design or integrity of the CRDMs or nozzle flanges. The probability of occurrence or the consequences of an accident or malfunction of equipment evaluated in the SAR was not increased nor the possibility of an accident or malfunction previously evaluated in the SAR created. Concurrently, the margin of safety as defined in the basis for any Technical Specification or the Safety Analysis Report is not reduced or affected. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Reactor Building Sump Screen Hatches (CMR 9[illegible]87)

Description of Modification: Stainless steel plate hatch covers were installed on the Reactor Building Sump replacing the existing screen hatch covers.

Safety Evaluation Summary: Evaluation identified no adverse affect on nuclear safety or safe plant operations since there is no adverse affect on safety systems that depend on the RB sump to supply screened water after a LOCA. The modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR since the head loss resulting from the new hatch covers is insignificant and more than accounted for by the conservative calculation of the DH and BS pump NPSH. The possibility of an accident or malfunction of a type different than previously evaluated was not created by the modification since the passive function of the sump screen is maintained. No margin of safety defined in the basis of the Technical Specifications was reduced as a result of the modification because pump DH and BS pump NPSH were unaffected. This modification did not constitute an Unreviewed Safety Question.

****

Modification: NaOH Tank Chemical Addition Point (CMR 90-172)

Description of Modification: The corrective change made permanent a temporary mechanical modification which installed a 0.5" diameter ball valve in the suction piping of Building Spray Pump 2 (BS-P-2) that provided a NaOH addition point for the NaOH tank.

Safety Evaluation Summary: Installation of the valve in support of a chemical addition point did not adversely affect nuclear safety or safe plant operation since the valve and the piping to which it was attached serves no safety related function. The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated were not increased since the piping is non-seismic and serves no safety related purpose. Installation of the valve did not affect the BS-T-2 isolation valves or associated piping and did not therefore create the possibility for an accident or malfunction of a different type than previously evaluated in the SAR. The change was found not to affect any margin of safety defined in the Technical Specifications. The change requires no revision to the plant SAR or Technical Specifications and represents no Unreviewed Safety Question.

****

Modification: Diesel Generator DF-X-1A Inter-Polar Connectors Removal (CMR-90-158)

Description of Modification: This modification removed inter-polar connectors from TMI-2 Diesel Generator 2DF-X-1A which is being utilized to meet station blackout requirements of 10 CFR 50.63.

Safety Evaluation Summary: TMI-2 Diesel Generator DF-X-1A was modified during the 9R outage to provide an alternate source of power during a station blackout event. Removal of the inter-polar connectors provides assurance that the DF-X-1A generator will be capable of performing its regulatory required function for a station blackout event. Therefore, nuclear safety and safe plant operations were not affected. This modification did not constitute an Unreviewed Safety Question.

****

Modification: 250 Volt DC Fuse Replacement (CMR 91-063)

Description of Modification: As a result of an evaluation performed in response to INPO bulletin OE-3990 regarding the adequacy of voltage ratings of DC distribution fuses, a corrective change was performed to replace fuses with insufficient voltage ratings with fuses with a minimum rating of 300 Volts DC. Certain other 125 volt DC fuses were required to be replaced to maintain coordination with replaced upstream fuses.

Safety Evaluation Summary: Evaluation identified no adverse affect on nuclear safety or safe plant operations since replacement fuses were sized in accordance with Technical Data Report, were tested at an independent lab and properly rated. The corrective change was not found to increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR or malfunction of equipment important to safety. The possibility of an accident or malfunction of a type different than previously evaluated was not created since the replacement fuses have no effect on the ability of the ES system to shutdown the plant safely. No Technical Specification margin of safety was identified which was reduced as a result of the corrective change. This change did not constitute an Unreviewed Safety Question.

****

IV. Temporary Mechanical Modifications, Lifted Leads, and Jumpers

Modification: Nitrogen Supply to "A" OTSG (TMM 1)

Description of Modification: Installation of a temporary line to supply N2 to the "A" OTSG until a replacement metal bellows hose can be supplied by the vendor.

Safety Evaluation Summary: The temporary supply line was used during the 9R outage and removed prior to heatup. The replacement line performed the same function as the original equipment and rated the same. Evaluation determined that the temporary supply line had no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Moisture Separator Sampling Capability (TMM 1)

Description of Modification: Tubing was installed to appropriate valves to permit sampling of the moisture separators via the corrosion product samplers.

Safety Evaluation Summary: Since the turbine plant sampling system design and operating details are not described in the FSAR and there are no Technical Specifications associated with the iron samplers or their sample inlet lines, the addition of the sampling lines did not adversely affect nuclear safety or safe plant operations. This modification did not constitute an Unreviewed Safety Question.

****

Modification: Bypass of WT-T-20 (TMM 1)

Description of Modification: Temporary hoses and bag filters were utilized during the bypass of WT-T-20 to permit operation of pretreatment while the gravity filter was out of service for rebuilding.

Safety Evaluation Summary: The Pretreatment system is not safety related and not required for safe plant shutdown. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Fire Service Water Cooling of Feedwater Samples (TMM 1 and 3)

Description of Modification: Fire Service Water hose connections at stations at FS-V-119 and 126, in the Intermediate and Turbine Buildings respectively, were fitted with quick disconnect couplings and utilized Fire Service Water to cool feedwater samples obtained during Chemtrak Testing.

Safety Evaluation Summary: The use of the associated Fire Service Water hose reels did not significantly reduce the fire suppression capability in the general area because of the ability to quickly reconnect the hoses to the system. Additionally, alternatives such as extending hoses to other hose lines at lower elevations provided acceptable fire suppression capabilities. There was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Operation of the Circ Water System Without the CW-V-17A Internals (TMM 2)

Description of Modification: The internals were removed from CW-V-17A and flow was directed to the chlorine injectors via another leg of the system until the valve was repaired during the 9R outage.

Safety Evaluation Summary: The Circ Water system is classified as "Other" and as such is not considered in the accident analyses. There was no adverse affect on nuclear safety or safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Temporary Connection Between LO-T-3 and LO-T-1 (TMM 3)

Description of Modification: A 4" hose was connected between LO-T-3 and LO-T-1 during a casualty to the turbine plant that requires dumping oil to the lube oil tank during such an emergency.

Safety Evaluation Summary: The temporary modification would be implemented only if a casualty to the turbine plant required emergency transfer of oil between the turbine lube oil reservoir to the underground lube oil tank. All components of the system are classified as "Other" and no reactor plant systems or components were affected by the temporary modification. There were no adverse affects to nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Fire Service Water Protection for the Temporary Toilet Facilities (TMM 3)

Description of Modification: During the 9R outage, a Fire Service Water supply was provided to the fire system sprinklers installed in the temporary toilet facility trailer on the south-east side of the turbine operating floor. The water was supplied from FS-V-136 through a temporary tee that allowed the fire hose reel station at the valve to remain in service.

Safety Evaluation Summary: The temporary modification was determined not to affect the operation of the fire system or hose reel FS-V-136 since this hose reel remained in service. The fire exposure risk to the plant was diminished by supplying the trailers fire sprinklers. There was no adverse affect on nuclear safety or safe plant operation from this temporary modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Operation of CW-V-17B Without Internals (TMM 5)

Description of Modification: CW-V-17B was operated without its internals since the valve could not be repaired within a reasonable period after the internals were found broken.

Safety Evaluation Summary: The Circ Water system is classified as "Other" and was not considered in the accident analyses. CW-V-16B was used for loop isolation during the period CW-V-17B went without being repaired. The temporary modification had no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Test Connection to Support Feedwater Flow Rate Determination (TMM 6)

Description of Modification: A valve was installed off of the tee downstream of FW-V-47 and 48 to enable the collection of flow data used by Combustion Engineering test equipment in the determination of main feedwater flow rate for flow element calibration.

Safety Evaluation Summary: The temporary modification was classified as "Other" and had no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Demineralized Water Source to the B&W OTSG Chemical Cleaning System and Waste Water Return Path (TMM 13)

Description of Modification: A hose was connected between DW-V-60 and the B&W Chemical Cleaning System to provide a source of demineralized water. A waste water return path was also provided.

Safety Evaluation Summary: The major concern involved with the hose installation was localized flooding as a result of hose failure. Appropriate steps such as hose design and system specifications were taken into consideration to minimize a hose failure. There was no adverse affect to safe plant operation that resulted from this temporary modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Filtered Water Source to the B&W OSTG Chemical Cleaning Post Cooling Tower for Makeup (TMM 14)

Description of Modification: A hose was connected between WT-V-733 at the normal makeup to the industrial cooler expansion tank and the B&W Chemical Cleaning System to provide a source of filtered water.

Safety Evaluation Summary: The capability to supply simultaneous makeup to both the industrial cooler expansion tank and the B&W Chemical Cleaning system exists. However, the demand for makeup by each is intermittent. There was no adverse affect to safe plant operation that resulted from this temporary modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Turbine Generator Exciter Base Cooler (TMM 21)

Description of Modification: The modification was installed to minimize the exciter vibration by applying heat and cooling to opposite corners of the exciter base via temporary air conditioning units.

Safety Evaluation Summary: The generator exciter is classified as "Other." The electrical load for the air conditioners supplying cooling air was evaluated and found acceptable. The heating and cooling effect is external to the rotating assembly and did not adversely affect safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Main Condenser Vacuum Breaker/Auto Vent Leak Repair (TMM 23)

Description of Modification: A temporary plate welded inside the waterbox in combination with a flange installed outside of the condenser were used to isolate the leak on the auto vent pipe. The temporary modification was applicable to both the A side of the condenser.

Safety Evaluation Summary: The leak was stopped by isolating the auto vent. The intermediate waterbox would be vented through alternate vents and the tubes if required. Proper operation and shutdown capability of the Circ Water system was not affected by the temporary modification. Safe plant operation was not adversely affected. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Provide Additional Cooling Air Flow to the Main Vacuum Pumps (TMM 25)

Description of Modification: The existing fan discharge diffuser at the fan discharge flange of AH-E-55A was removed and replaced with an angled diffuser to provide additional air flow and cooling to the main vacuum pumps VA-P-1A/B/C.

Safety Evaluation Summary: The Turbine Building heater bay fan , AH-E-55A is classified as "Other" and will continue to operate as before. There was no adverse affect on safe plant operation as a result of this temporary modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Reactor Building Service Air Supply via Chemical Cleaning During Outages (TMM 26 and 28)

Description of Modification: The Chemical Cleaning system piping was utilized to supply service air to the Reactor Building during the 9R outage. A temporary fixture was installed to reconfigure the system piping to support the temporary modification and removed prior to restart after the outage.

Safety Evaluation Summary: The Chemical Cleaning system was modified temporarily to supply service air to the Reactor Building. It was returned to its normal configuration prior to return to power operation and had no affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: "A" High Pressure Injection Line at MU-H-43 Monitoring (TMM 28)

Description of Modification: Thermocouples were attached to the top and bottom of the "A" HPI line at MU-H-43 to provide readout outside the Reactor Building for evaluation of MU-V-95 back flow.

Safety Evaluation Summary: The temporary modification was evaluated and found to have no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Suction Filters on EG-Y-3A/B Oil Lift Pumps (TMM 26 and 27)

Description of Modification: The previous failure of the EG-Y-3B oil lift pump was attributed to dirt in the oil as determined by the vendor representative. Suction filters were installed as recommended to improve pump reliability.

Safety Evaluation Summary: EG-Y-3A/B are not safety related components and are not described in the FSAR or the Technical Specifications. The temporary modification did not adversely affect safe plant operations or constitute an Unreviewed Safety Question.

****

Modification: Alternate Makeup Supply to the Industrial Coolers via the Fire Service System (TMM 26)

Description of Modification: A 1.5" fire hose was installed between FS-V-119 and WT-V-733 via a temporary tee to provide continued water makeup to the industrial coolers while the normal supply was isolated.

Safety Evaluation Summary: AH-C-1A/B are not safety related components. Loss of the spray side of the coolers in the event of a failure would not eliminate all Reactor Building cooling. The air side of the coolers would maintain Reactor Building temperatures low for several hours to permit repair of the normal supply path. Reactor Building Emergency Cooling was unaffected by the temporary modification and would continue to perform its intended function. The Fire Service system remained operable. Nuclear safety and safe plant operation were not adversely affected and the temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Main Condenser Vacuum Breaker/Auto Vent Leak Repair (TMM 30)

Description of Modification: A temporary plate welded inside the waterbox in combination with a flange installed outside of the condenser were used to isolate the leak on the auto vent pipe. The temporary modification was applicable to both the B side of the condenser.

Safety Evaluation Summary: The leak was stopped by isolating the auto vent. The intermediate waterbox would be vented through alternate vents and the tubes if required. Proper operation and shutdown capability of the Circ Water system was not affected by the temporary modification. Safe plant operation was not adversely affected and the temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Modification of Replacement Breaker for the Service Water Post Cooling Tower to Provide Equivalent Capabilities of the Original (J 57)

Description of Modification: A jumper was installed on the replacement Service Water Post Cooling Tower control center breaker to provide for forward rotation and permit normal fan functions.

Safety Evaluation Summary: The fans controlled by the breaker reduce the temperature of plant cooling water before it is returned to the river. They are not required for safe plant operation. The jumper installation had no adverse affect on safe plant operation and the temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Rad Waste Evaporator Vent Cooler (TMM 58)

Description of Modification: Due to excessive steaming during operation of the Rad Waste Evaporator a temporary air to tube heat exchanger was installed on the condensate return unit vent piping to condense the steam prior to its return to a floor drain.

Safety Evaluation Summary: The temporary modification was evaluated and it was determined that no NSR components are in the general area that would be damaged if the additional piping were to fall down. There is no missile hazard since the "condenser" is vented to the atmosphere. The existing condensate return unit system is classified as non-seismic and "Other." There were no adverse affects from this temporary modification. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Isolation of Circ Water Pressure Switch 214 for Repair (TMM 60)

Description of Modification: To facilitate repair of CW-PS-214 a pipe nipple and valve were added to the discharge line of CW-P-1D to provide an isolation for the pressure switch.

Safety Evaluation Summary: The temporary modification provided an isolation valve to prevent leakage during the repair of CW-PS-214. No adverse affects on safe plant operation were identified. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Pressure Gauge Installation to Support Steam Trap Performance Data Gathering (TMM 70, 71 and 72)

Description of Modification: Three 0 to 30 psig pressure gauges were installed on the Rad Waste system auxiliary steam supply to permit monitoring of system operating conditions and evaluate performance.

Safety Evaluation Summary: The temporary modification was made to a system classified as "Other" and did not adversely affect safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Bypass of Chill Water Flow Switch AH-C-38 (J 16)

Description of Modification: A temporary jumper was installed on faulty flow switch AH-C-38 to permit the chillers to operate regardless of the chill water flow.

Safety Evaluation Summary: The system was evaluated and the temporary jumper was determined to have no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Temporary Power Feed to Communications during Bell System Outages (J 30)

Description of Modification: During a maintenance outage of power panel ATB, a temporary power feed to communications power panel D-22 was established from a spare 50 amp breaker in power panel D-1B.

Safety Evaluation Summary: The addition of the jumper permitted alternate off-site communication capabilities during the Bell System outage . Off-site communication was available to support any plant emergency during the period the jumper was installed. There was no adverse affect on safe plant operations. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Power the 1N Bus from the 1G Bus (J 39)

Description of Modification: A jumper was installed to allow the 480V switchgear 1N to be powered from 480V switchgear 1G with the 1N Feeder Breaker 1N-02 open.

Safety Evaluation Summary: The plant was shutdown when the jumper was installed. All work completed was performed on the balance of the plant components with no ties to safety related systems or components. Normal power flow is from the 1N to the 1G bus, however, it was acceptable to have 1G as the source during outage work. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Auxiliary Fuel Bridge Operational with Mast Removed (J 40)

Description of Modification: With a lead lifted and a jumper installed, the aux fuel bridge was made operable to check the bridge/trolley control functions while the fuel mast was removed.

Safety Evaluation Summary: Only the bridge/trolley control functions were affected by this temporary modification. No fuel was lifted or moved while the modification was installed. There was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Control Room Heating and Ventilation Control (J 79 and 92)

Description of Modification: An Ampacitor test of the feed to 1A Reactor Plant Heating and Ventilation (H&V) required that a jumper be placed to keep AH-P-7C/D in service. A jumper was installed on the LS267X relay.

Safety Evaluation Summary: Installation of the jumper assured that cooler spray was maintained. The low level alarm switch remained functional and prompted operators to secure the pump. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Open WDL-V-535 With Low Sump Level (J 92)

Description of Modification: A jumper installed in the XCR cabinet to bypass an interlock and permitted the opening of WDL-V-535 with a low sump level.

Safety Evaluation Summary: The reactor plant was in refueling shutdown with all fuel removed from the core during the active period of this modification. There was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Lifted Leads on the Unit 1 Side of the Unit 1 to Unit 2 Tie to the #4 and #8 Bus Primary and Backup Lockout Relays (LL 5-10, 15 and 18)

Description of Modification: The series of lifted leads opened sliding links removed the Unit 1 to Unit 2 tie equipment which was out of service but remained connected to the #4 and #8 bus primary and backup lockout actuation circuitry to prevent inadvertent or incorrect actuation schemes.

Safety Evaluation Summary: Opening the links ensured that if the TI-T2 and TI-E2 breakers should be racked in, the out of service devices would not inadvertently trip the #4 or #8 bus lockouts. Opening the link assured the reliability if the off-site power source. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Auxiliary Fuel Bridge Operational with Mast Removed (LL 21)

Description of Modification: With a lead lifted and a jumper installed, the aux fuel bridge was made operable to check the bridge/trolley control functions while the fuel mast was removed.

Safety Evaluation Summary: Only the bridge/trolley control functions were affected by this temporary modification. No fuel was lifted or moved while the modification was installed. There was no adverse affect on nuclear safety or safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: 1N-12 Trip Fuse Removal (LL 118)

Description of Modification: The 30 amp trip fuses were removed from the 1N-12 tie breaker to prevent tripping of 1N-12 when 1N-02 was open.

Safety Evaluation Summary: The tie breaker was capable of being manually tripped and overcurrent protection was provided by the 1G-02 breaker. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.

****

Modification: Power the 1N Bus from the 1G Bus (LL 118)

Description of Modification: A lifted lead was installed to allow the 480V switchgear 1N to be powered from 480V switchgear 1G with the 1N Feeder Breaker 1N-02 open.

Safety Evaluation Summary: The plant was shutdown when the lifted lead was installed. All work completed was performed on the balance of the plant components with no ties to safety related systems or components. Normal power flow is from the 1N to the 1G bus, however, it was acceptable to have 1G as the source during outage work. There was no adverse affect on safe plant operation. This temporary modification did not constitute an Unreviewed Safety Question.