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

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
10 CFR 50.59 Report for 1990

In accordance with the requirements of 10 CFR 50.59, enclosed are summaries of the changes to TMI-1 systems and procedur s, for the period of January to December 1990, as described in the Safety Anaylsis Report (SAR). Attachment 1 of this report addresses those activities which directly affected systems/components described in the SAR. Attachment 2 of this report addresses those activities for which a GPU Nuclear safety evaluation was performed, due to the potential for the activity to adversely affect nuclear safety or safe plant operations, but which do not directly impact SAR systems/components.

Sincerely,

T. G. Broughton

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

TGB/RDW
cc: R. Hernan - Senior Project Manager
T. Martin - Regional Administrator
J. Stolz - Director, Plant Directorate I-IV
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Enclosure

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I. Tests and Experiments

Activity: Special Test Procedure (STP) 1-90-039 Once Through Steam Generator (OTSG) Operating Range (OR) Level Verification (SE 000224-009 Rev. 1)

Description of Activity:

The purpose of this test was to increase the OTSG OR level to demonstrate that: a) safety analysis assumptions are not exceeded even if the downcomer is full of water; and b) loss of feedwater heating will not adversely affect plant safety. (See Section II, "Procedure Changes", of this report for discussion of the results of this test).

Safety Evaluation Summary:

The OR level indication was originally conceived as a relative measure of OTSG inventory during power operation. Although the OR level is often referred to as downcomer level, both field data and tests show that there is no steam-water interface in the downcomer, but rather a steam water mixture that extends from just below the main feedwater nozzle down to the downcomer orifice. Raising of the OTSG high level to maximum had the potential to affect the feedwater heating function if the feedwater nozzles were flooded and could increase the secondary side steam and water mass inventory as related to main steam line break analysis. As a result of raising the OTSG high level limit, two (2) possible effects on system performance were considered: 1) downcomer flooding as related to maintaining the feedwater heating function; and 2) secondary side steam and water mass inventory as related to the design basis accident analysis.

This test procedure was developed to slowly increase power (thus the downcomer level) to approach the maximum achievable level. The purpose of this test was to determine the indicated level at which the plant can be operated without losing feedwater preheating. The main steam line break (MSLB) accident from hot shutdown conditions is not affected by a change in the high level limit since the steam generators are operated in a flooded nozzle condition. Calculations determined that at 100.2% OR level at 100% power, the feedwater nozzle will not be flooded and proper feedwater heating is attained. The margin of safety defined in the SAR was not reduced since the total inventory at the proposed higher level is less than the value of 62,600 lbm assumed in the SAR and 55,000 lbm assumed in other analyses. The running of an OTSG high level test did not increase the probability of occurrence of any accident or transient. Neither increased downcomer inventory/nor decreased downcomer temperature initiate accidents or transients.

Activity: Lithium Tracer Testing for Venturi Nozzle Calibration (SE 412565-002 Rev. 0)

Description of Activity:

The purpose of the CE Chem Trac lithium tracer test was to calibrate the replacement feedwater flow venturis by accurately measuring feedwater flow to each OTSG using a chemical tracer (see related modification BA 412565 under Section III of this report).

Safety Evaluation Summary:

The only aspect of the chemical tracer test which could have impacted an accident probability related to secondary side lithium injection. The OTSG tube rupture is the only accident evaluated in the SAR whose probability could have been impacted by secondary side chemistry alteration. Based on the analysis in this SE, lithium had no degrading effect on the OTSGs; thus, the probability of occurrence of any previously evaluated accident was not increased. The conduct of this test did not introduce any new accident precursors. Reactor operations were maintained within licensed limits. Administrative controls were in place to preclude any adverse condition resulting from improper control of equipment to conduct the test. The nuclear instrumentation was calibrated based on conservative estimates of core power. Thus, the margin of safety as defined in the bases for the Technical Specifications relative to core power was not reduced.

II. Procedure Changes

Procedure: AP 1035 Control of Transient Combustible Materials (PCR 1-EG-90-0003)

Description of Change:

This change incorporated a control rod storage box (untreated lumber) to be stored in the fuel pool area approximately three (3) months after each refueling outage.

Safety Evaluation Summary:

The TMI-1 Fire Hazards Analysis Report (FHAR) identifies the type of transient combustibles stored in each fire area/zone. This procedure change will be incorporated into the next update of the FHAR. All changes in the fire loading are within the limits established to support the 10 CFR 50 Appendix R evaluation.
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Procedure: AP 1038 Administrative Controls - Fire Protection Program (PCRs 1-EG-90-0001, 1-EG-90-0004, and 1-EG-90-0015).

Description of Change:

PCR 1-EG-90-0001 corrected procedural references, and updated Exhibit 2, Table 1 to show reduced number of fire detectors associated with the Mod Comp Halon System.

PCRs 1-EG-90-0004 corrected a procedural reference from OP 1105-21 to OP 1105-20, Remote Shutdown System."

PCR 1-EG-90-0015 implemented fire drill changes discussed in GPU Nuclear letter C311-90-2073 dated June 1, 1990 and reviewed (prior to implementation) by NRC letter dated November 21, 1990. An unrelated change was also made to Section 5.8 to delete the requirement to issue a quarterly fire drill results summary.

Safety Evaluation Summary:

The correction and update of procedure references is administrative in nature. The deletion of one fire detector associated with the Mod Comp Halon System was performed under BA 412281 and evaluated in SE 412281-005, Revision 2, and FPE TI-412281-004, Revision 1. The minimum number of required operable detectors remains at three (3) for the halon system. One (1) detector was eliminated as the result of a reduction in the size of the monitored area. Since the fire detection system is a priority matrix system and since the remaining trench area is exposed to the observation room underfloor, the detection system will still

Safety Evaluation Summary (Cont'd):

operate as required. The deletion of the quarterly drill results summary is administrative in nature since drill experiences are addressed in monthly training sessions. Therefore, these changes do not adversely affect nuclear safety or constitute an unreviewed safety question.

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Procedure: Revised OTSG High Level Limit Following STP 1-90-039 (SE TI-115403-005)

OP 1101-1 Plant Limits and Precautions (PCR 1-OS-90-359)
OP 1101-2 Plant Setpoints (PCR 1-OS-90-357)

SP 1303-11.37 A-D HSPS-OTSG Level and Pressure Channel Tests (PCRs 1-MT-90-8559, 8560, 8562, and 8563)

Alarm Response Procedures H-1-1 and H-1-2 Main Annunciator Panel H (PCR 1-OS-90-358)

Description of Change:

The purpose of the above procedure changes was to establish a maximum steady state OTSG downcomer level and associated control system settings based on the results of STP 1-90-039 (see Section I, "Tests and Experiments"). The results of the plant test to raise OTSG level were used in establishing the ICS high level limit and high level alarm setpoints.

Safety Evaluation Summary:

STP 1-90-039 was conducted to determine the maximum OR level for operation without flooding the Feedwater (FW) nozzles. The test was performed by slowly raising unit load, thus downcomer level, while monitoring for a significant reduction in FW preheating caused by flooding the nozzles. The results of this test indicated that the downcomer level was beginning to encroach on the mixing region below the FW nozzles above approximately 99% OR level. A significant decrease in downcomer temperature did not occur during the conduct of this STP. The plant response during this STP indicated that there is about a 14 inch (4.8% OR level) margin available to accommodate slow and small upsets in plant conditions and not cause FW nozzle flooding. Downcomer temperature was relatively constant until OR level increased above approximately 95%. In summary, there were no indications of unacceptable plant response during the elevated level test. Steam superheat, cold leg temperature, OTSG pressure, core power and downcomer temperature responded within acceptable limits.

Safety Evaluation Summary (Cont'd):

A significant reduction in lower downcomer temperature during a rapid transient could affect tube stresses, but the operator can reduce FW flow before the shell to tube ΔT could increase above the compressive stress limit. This activity did not decrease the margin of safety as defined in the basis of the Technical Specifications. The TMI-1 Technical Specifications do not have limits on either OTSG level or shell-to-tube differential temperature. However, a procedural limit was established to assure that thermal stresses were kept within design value. Nuclear safety and safe plant operations were not adversely affected by raising the high level limit to a point close to flooding the FW nozzles during steady state operations. The OR levels experienced during the STP will not be exceeded during steady state plant operations. This STP validated that operating at the new high level limit would not cause nozzle flooding.

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Procedure: OP 1104-5 Reactor Building (RB) Spray System (TCN 1-90-0134)

Description of Change:

The procedure was temporarily revised to provide for sodium hydroxide additions using a chemical addition pump instead of a crane.

Safety Evaluation Summary:

This procedure change created a potential pathway by which sodium hydroxide could drain from Sodium Hydroxide Storage Tank (BS-T-?). The sodium hydroxide tank level was maintained lower than the BWST level such that the BWST/SHST differential level was maintained within Tech. Spec. limits. This change did not affect the major components of the RB Spray System. A temporary rig was attached to the recirculation piping which had its own isolation valves. The inventory of the sodium hydroxide tank was maintained throughout the entire chemical addition; therefore, this change did not increase the probability or consequence of an accident previously evaluated in the FSAR.

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Procedure: Abnormal Procedure 1203-19 River Water System Failure, Decay Heat & Secondary Services (DR/SR) (PCR 1-OS-90-0260)

Description of Change:

This procedure change provided guidance to cross-connect Nuclear Services River Water (NR) to Secondary Services River Water (SR) in the event of a total and sustained loss of SR.

Safety Evaluation Summary:

This procedure change implements a design feature which would permit emergency cooling to secondary components while plant shutdown was in progress. While implementation of this design feature degrades the safety classification of the NR system, long term operation in this mode is prevented by the existing Tech. Spec. requirements. Use of this option is considered to be safe since sufficient NR cooling will be maintained during the shutdown process. Since this emergency condition would require a plant shutdown within the time constraints established by Tech. Specs., this change had no adverse impact on the TMI-1 FSAR.

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Procedure: SP 1302-1.1 Power Range Calibration (PCR 1-OS-90-0576)

Description of Change:

This procedure change provided guidance for not performing power range calibration when the Nuclear Instruments (NIs) exceed heat balance power by more than two (2) percent during temporary power reduction.

Safety Evaluation Summary:

This change did not affect nuclear safety. It allowed the power range NIs to be greater than heat balance power for plant conditions of temporary power reduction. This provides a protective system actuation at a lower actual power than if the NIs were adjusted down to agree with the heat balance. The probability of occurrence or the consequences of an accident were not increased by this change because it affects only calibration and does not change any protective functions.

The margin of safety was improved by this change. FSAR section 7.3.1.3 states, "the sum (of upper and lower power range detectors) will be recalibrated whenever it is determined that the sum disagrees with the heat balance by 2 percent or more." This change created a larger margin to safety for the condition of a temporary power reduction. This change remains in compliance with Technical Specification Table 4.1-1.

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Procedure: Alarm Response Procedure H&V A-1-9 Reactor Compartment Air Temperature High (PCR 1-OS-90-0343)

Description of Change:

This procedure change increased the setpoint of TR-802 points 1,4,7, and 11, from 210°F to 235°F.

Safety Evaluation Summary:

The temporary setpoint increase provided a means to return TR-802 recorder points 1,4,7, and 11, to service such that the alarm input is active. This was a temporary setpoint change to one (1) of two (2) indicators for each penetration and both remain functional and operational. Previously, the points did not provide alarm input since they cycled on and off at the 210°F setpoint which was a distraction and nuisance for operators. No conclusive data has been obtained that would provide indication of what concrete temperature profile is relative to air temperature exiting the penetration. A one time test revealed concrete at 185°F and air temperature at 200 to 205°F. This suggested a ΔT of 15 to 20° between actual concrete temperature and penetration air exit temperature. Thus, an alarm setpoint of 235° would lead to a possible concrete temperature of 215-220° which exceeds the FSAR limit of 200°F. From a materials standpoint, if the local concrete temperature would approach 220° there is not a structural or strength concern. Thus, a temporary setpoint increase does not cause operator distraction, provides a meaningful alarm indication for unrepresentative recorder points, and does not compromise the structural integrity of the concrete. Alternate temperature indication setpoint was not changed and continues to provide temperature alarm at original values. Therefore, nuclear safety is not adversely affected by this change.

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Procedure: PM M-155 Main Steam (MS) Relief Support System Clearance Readings (PCR 1-MT-90-1003)

Description of Change:

This procedure change eliminated the taking of clearance readings during hot plant conditions due to personnel safety. Gap readings for all the posts except MS-V21B were previously eliminated during hot conditions based on consistent measurements "hot" and "cold".

Safety Evaluation Summary:

Historical data has shown that gap measurements that are acceptable during cold plant conditions are acceptable also during hot plant conditions. Therefore, gap settings are being monitored during cold plant conditions. The heated post for MS-V-21B was adjusted during the [illegible] outage so that the gap is within the acceptance criteria during hot and cold conditions. Measurements of the gap have shown this to be the case. Therefore, elimination of this surveillance during hot conditions for personnel safety did not jeopardize the gap clearance or nuclear safety. This change is further addressed in detail in Safety Evaluation SE-000415-001.

III. Modifications

Modification: Modification to Waste Level Controllers LC-40A/B (WDL-649A/B) [BA 123212 (123216)]

Description of Modification:

This modification improved the performance of the Waste Evaporator System Level Controllers by installing an improved model float type regulating valve, float adjustment mechanism, sightglass/access port, and isolation valves.

Safety Evaluation Summary:

This modification serves no safety function other than a pressure boundary for a radwaste system. The modification does not compromise Waste Disposal (WDL) piping or the operability of the radwaste evaporators or any other system or component. The modification improves the reliability of the Radwaste Evaporator System. The evaporators are not nuclear safety related components and no nuclear safety related components are affected by this change.

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Modification: Installation of Air filters on Critical Plant Control Valves (BA 123150)

Description of Modification:

To provide a clean, reliable source of instrument air to critical plant control valves, instrument air filters or filters/regulators (with bypass and isolation capability) were installed at the point of use for the following valves: EF-V30 A-D, FW-V16A/B, FW-V17A/B, MS-V3 A-F, MS-V4A/B, MS-V6, MU-V5, MU-V17, MU-V32, AND RR-V6.

Safety Evaluation Summary:

Installation of local air filters or addition of a parallel filter/regulator in the supply air lines to the above valves does not affect the safety function of the Instrument Air (IA), Backup Instrument Air (BUIA), or Two Hour Backup Air Supply (2HBUIA) Systems. Filters and filter/regulators, with access for PM activities, ensured that a clean, reliable source of air is provided to critical control valves. Filters and filter/regulators are adequately sized to prevent excessive loading of the filter elements within the PM period. This modification serves to improve the quality of instrument air supplied to the specified plant control valves and increases control valve reliability.

Modification: Intermediate Closed Cooling Water (ICCW) Drain Valve Modification (BA 128087/WA 51145310)

Description of Modification:

The purpose of this modification was to provide a means with which to drain the ICCW line supplying the Control Rod Drive (CRD) motor cooling coils. The line is required to be drained when it is disconnected due to RV head removal. The scope of this modification involved the installation of a 3/8" drain valve and a short piece of tubing and tubing cap on the ICCW line in order to facilitate the draining of the line prior to disconnection for RV head removal.

Safety Evaluation Summary:

The installation of the drain line did not affect the function or operation of the ICCW system. The valve is only used during a plant shutdown prior to removal of the RV head. The valve is normally closed and capped. Thus, this modification did not adversely affect nuclear safety or safe plant operations. The new valve is used only as an aid to draining the ICCW system during plant shutdown. The modification did not decrease the margin of safety as described in the SAR or the Technical Specifications since no system cooling functions or containment isolation functions were altered by this modification.

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Modification: Secondary Plant Performance Monitoring Modification (BA 41227)

Description of Modification:

The purpose of this modification was to provide additional instrumentation to enhance the secondary plant performance monitoring. The scope of this modification was to install the necessary pressure and temperature devices to allow for trending ability and the identification of degraded equipment. The additional instrumentation was for the Main Turbine L.P. Section Pressure, Condenser Vacuum and L.P. Feedwater Heater Temperature, and various valves in the secondary plant.

Safety Evaluation Summary:

This modification did not reduce a margin of safety because there was no adverse affect on the safety features described in the applicable sections of the FSAR for the affected safety systems (i.e., Extraction Steam System and the Main Condensate System). Additionally, the new instrumentation installed by this task is not governed by the Technical Specifications; thus, this modification did not result in an unreviewed safety question.

Modification: Demolition of Bailey and Mod Comp Computer (BA 412281)

Description of Modification:

The Mod Comp and Bailey Computer System were replaced with a state-of-the-art Gould SEL computer system as part of the plant process computer upgrade. Following removal, the remaining Bailey cabinets in the Relay Room and Control Room function only as termination cabinets. The "B" SEL computer was connected to the T-Bar cabinet in the Control Tower where the Mod Comp Computer had been previously connected. This modification also removed a fire detector from the section of the raised wireway in the Control Room which was removed.

Safety Evaluation Summary:

This modification did not impact equipment and electrical circuits which are required for 10 CFR 50 Appendix R safe shutdown as described in the TMI-1 FHAR. The fire detection system was not adversely affected since the volume protected by the removed detector did not impact the performance of the halon system since it has a priority matrix detection system, whereby any two remaining detectors will initiate the halon system. Interfacing systems were reviewed to identify any pertinent nuclear safety related functions or safe plant operations. To ensure that these nuclear safety related functions or safe plant operation requirements were not adversely affected, the following provisions were made:

a) Conduit was added in accordance with anti-falldown support requirements for Class I Structures.

b) Cable routed for this modification was entirely within seismic trays within Class I Structures.

The SAR was reviewed to identify any accident scenarios previously evaluated that could be affected by this modification. The results of this review were that no previously evaluated accident situation could be found that would directly or indirectly be affected by this modification.

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Modification: Regulatory Guide (R.G.) 1.97 Modifications (BA 412491)

Description of Modification:

R.G. 1.97 required that certain plant parameters be connected to the plant computer. The cables for these computer points were routed during 7R Outage and terminated at the signal conditioning cabinets. These cables are now terminated at the computer I/O cabinet.

Safety Evaluation Summary:

This modification did not adversely affect nuclear safety or safe plant operations. The computer is not a nuclear safety related system. The computer input signals were verified by performing a loop calibration. The margin of safety as defined in the basis for any Technical Specification was not reduced. Availability of the new computer points is not assumed in the basis of any Technical Specification.

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Modification: Radwaste Solidification System/Building Upgrade (BA 412521/WA 37521A10)

Description of Modification:

This modification involved the upgrading of both the Hittman processing system and the Hittman Building. Modifications included:

- o Change over of electrical power from Unit 2 to Unit 1 for increased reliability and availability.
- o Permanent, "hard piped" connection of process vent and dewatering return line to the Auxiliary Building (Aux. Bldg.) HVAC exhaust duct and Aux. Bldg. floor drains, respectively.
- o Conversion of the walls of the structure to concrete block for increased strength and durability, increased weather protection, added radiological protection, elimination of combustible materials, and improved decontaminability of building surfaces.
- o Relocation of radwaste supply lines inside the Hittman Building to improve interface with the radwaste solidification liners and to provide increased clearance between piping and operations personnel when access to the solidification process is required.

Safety Evaluation Summary:

These modifications improve reliability and availability of the system, eliminate reliance on Unit 2 for any support services, reduce the potential for unplanned releases of radioactivity, reduce radiation exposure to operating personnel and to the general public, improve ALARA conditions, and simplify equipment setup and material handling procedures. Connection of the process vent line to the decontamination facility ventilation exhaust duct has negligible impact on the ability of the decontamination facility ventilation system to perform its designed function. The Hittman system is non-nuclear safety and is not part of the engineered safeguards system. All modifications were performed in accordance with established quality requirements and seismic classifications. The modified

Safety Evaluation Summary (Cont'd):

system and structure comply with applicable regulations, Technical Specifications, and the station's Process Control Plan (PCP) for solidification of radioactive waste. The modifications did not affect the safety functions of any interfacing system or structures.

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Modification: Waste Oil Storage System Modification (BA 412521/WA 37521B10)

Description of Modification:

The purpose of this modification was to install a storage system for handling waste oil produced by plant components. The new system provides a clean and efficient means for storing, transferring, and sampling the waste oil prior to ultimate disposal. The new system replaced the previous method which utilized a portable pump and cask liners for transfer and temporary storage of waste oil.

Safety Evaluation Summary:

The installation of the Waste Oil Storage System did not affect the operation of the Radioactive Waste (Radwaste) Solidification System, including the Hittman facility. The modification installed a permanent storage system to replace the temporary equipment. The modification did not increase the probability of occurrence or consequences of an accident since the system replaced a temporary, makeshift oil storage system with a dedicated, permanent system. The new system provides better control and handling of waste oil transfer and storage. This modification did not adversely affect nuclear safety or safe plant operations since the safety functions of Radwaste systems, the Auxiliary Building Structure, and plant electrical systems were not adversely affected.

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Modification: Bailey BY Transmitter Replacement (BA 412522)

Description of Modification:

The purpose of this modification was to replace six (6) Bailey BY transmitters with new Rosemount differential pressure instruments. One BY transmitter, RC1-LT2 was deleted and not replaced. The transmitters affected by this modification were the pressurizer level instruments RC1-LT1, 2, and 3, and four (4) MFW flow instruments SP8A/B-DPT1/2.

Safety Evaluation Summary:

The pressurizer level transmitters provide information to the plant computer system, indication to the control room operators, and provide a pressurizer level signal to the ICS system. The feedwater flow transmitters provide an input

Safety Evaluation Summary (Cont'd):

signal to the ICS system and provide indication of feedwater flow to the control room. This modification does not decrease any margin of safety since this activity does not impact any systems' safety functions, the work is performed in accordance with existing seismic requirements, and electrical/physical separation was provided. New tubing and associated fittings for the pressurizer instrument impulse lines were installed and inspected in accordance with existing design criteria so as not to provide a leakage path for reactor coolant.
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Modification: MW Digital Meter/Transducer Upgrade (BA 412528/WA 30128310)

Description of Modification:

In order to provide the operator with more accurate indication of main generator MW the analog meter was replaced with a digital meter. The previously existing MW transducers used for ICS input are no longer available. These transducers (2) were replaced by more state-of-the-art type transducers.

Safety Evaluation Summary:

The margin of safety as defined in the basis for any Technical Specification was not reduced. The MW signal is not defined in any Technical Specification safety limit basis. Safe plant operations were not adversely affected since the operators are still provided with an indication of MW when loading and unloading the main generator. The availability of generated MW signals is not taken credit for in any safety analysis of Chapter 14 of the SAR. There are no safety related interlocks associated with MW signal loops.
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Modification: Installation of Main Steam to Heated Post Isolation Valves (BA 412528/WA 30129310)

Description of Modification:

A means shall be provided to maintain the heated post system while the plant is at power. Previously, all four (4) lines were required to be isolated to permit maintenance. Isolation of all four trains is not permitted by the Technical Specifications. This modification installed four isolation valves in the steam return line downstream of the drip leg. This permits short term isolation of individual trains for maintenance of related piping.

Safety Evaluation Summary:

The main steam to heated posts supports system is designed to maintain the supports for the Main Steam Safety Valve discharge line such that the main steam

Safety Evaluation Summary (Cont'd):

piping is not overstressed when the valves open. This modification did not increase the probability of occurrence or consequences of an accident, since the normally open valves will not affect system performance and the safety function of the heated post supports system. This modification did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, since the integrity of the pressure boundary and the seismic boundary of the main steam to heated post supports system are still maintained. This modification permits on line maintenance of individual trains of the main steam to heated post supports and related piping.

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Modification: Deletion of Water Detector AH-LS-249 A/B/C for the RB Ventilation Fan Motors (BA 412528/WA 30149310)

Description of Modification:

This modification deletes the water detector, AH-LS-249 A/B/C for the RB Ventilation fan motors, AH-E-1A/B/C, and spares the associated Control Room alarm and computer points. The water seals on the fan motor shafts were replaced with a dry seal.

Safety Evaluation Summary:

The margin of safety defined in the FSAR Chapter 6 was not reduced, and nuclear safety and safe plant operations were not adversely affected. The AH-E-1 motor design with dry shaft seals and continuous draining with backflow prevention from check valves protects the motor against motor water induction. Therefore, AH-L-249A/B/C are no longer required to assess the operability of the motors. Removal of the water detectors eliminates nuisance alarms due to repeated failures of the moisture detectors. Deletion of the moisture detectors did not introduce any new failure mechanisms for the RB fan motors. A pipe plug was installed in place of the moisture detector to prevent a possible new pathway for moisture leakage to the motor cavity.

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Modification: Heat Sink Protection System (HSPS) Additions (BA 412538)

Description of Modification:

GPU Nuclear committed to install, during the 8R refueling outage, different control room indication for the Main Steam Rupture Detection (MSRD) portion of the HSPS. TMI-1 Operations, Maintenance, and Plant Engineering had identified other concerns with the HSPS configuration, particularly as it related to the man-machine interface. A summary of each of the individual tasks associated with

Description of Modification (Cont'd):

this project is described below:

1. Provided new design to resolve the NRC's SER comment on MSRD control room indication which also includes new provisions for the CRO to isolate the OTSG independently.

2. Provided new channel-check provisions via the new computer system that would allow the CRO to read and compare all analog input parameters to HSPS.

3. Installed additional instrument tubing isolation valves that would allow transmitter testing without degrading HSPS logic from 2 of 4 to 1 of 2.

4. Installed new and reconfigured existing control room alarms concerning Emergency Feedwater (EFW) Actuated, EFW Defeated, and Main Feedwater (MFW) Isolated to provide clearer and more consistent annunciation of actuation/defeat status.

5. Installed new control room valve position indication for MFW control valves FW-V16/17; no direct position indication had previously existed.

6. Installed new capability to determine status of back-up power supply to HSPS cabinets.

7. Installed new capability to determine status of isolation fuses for MFW isolation valve solenoids.

8. Installed additional provisions to allow V/I module testing and Resistance Temperature Detector (RTD) test input without lifting leads in HSPS cabinets.

9. Installed new capability to test HSPS status lights in the control room and on the HSPS cabinets.

10. Provided a rescaled startup OTSG level range to control room recorder for enhanced readability at low levels.

11. Changed the default signal to the EFW control valves such that the controller will revert to the OTSG startup level signal rather than the OTSG operate range level signal.

Description of Modification (Cont'd):

12. During testing of this modification, an as-found condition was discovered which required a change to be engineered and installed prior to restart. It was found that partial losses of power within the HSPS cabinets could cause MFW isolation. Blocking diodes were installed in the HSPS cabinets, and additional confirmatory testing was performed.

Safety Evaluation Summary:

The installation of this modification improved the capability of maintaining the OTSGs as the primary heat sink and provides additional information to the operator in other related areas associated with the HSPS. This modification did not introduce any accident or malfunction not previously evaluated, nor did it increase the likelihood of occurrence or consequences of any accident as analyzed in the SAR. No safety margins were reduced as a result of this modification. Therefore, this modification did not adversely impact nuclear safety and does not represent an unreviewed safety question.
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Modification: MU-P-1B Lube Oil Pump Power Supply Changeover (BA 412543)

Description of Modification:

TMI-1 Plant Operations and Probabilistic Risk Assessment (PRA) identified a condition which could cause a total loss of High Pressure Injection (HPI) due to a single failure. This condition arose when either MU-P-1A or 1C is unavailable, MU-P-1B is selected for ES operation, and the 1C ES Valve MCC supplies power to the auxiliary and main lube oil pump (MU-P-2B/3B) either of which is required for operation of MU-P-1B. The modification changed the power supplies for MU-P-2B/3B from the previous source, 1C ES Valve MCC to the following:

o MU-P-2B -- 1A ES Valve MCC

o MU-P-3B -- 1B ES Valve MCC

This new power supply configuration ensures that MU-P-1B always has an energized lube oil pump to support operation.

Safety Evaluation Summary:

This modification does not adversely affect nuclear safety or safe plant operations and does not reduce a margin of safety because the purpose of this modification was to ensure that MU-P-1B always has an energized lube oil pump to support its operation. Reconfiguring the power supply of the lube oil pumps ensures that MU-P-2B is available to support MU-P-1B when powered from the 1D

Modification: 8R Underprotected Cable Fix Modification (BA 412561)

Description of Modification:

This task implemented during the 8R outage the modifications recommended in the Underprotected Cable Study Interim Report, Revision 1, dated June 5, 1989. This report identified the underprotected cables which represented the greatest risk to safe plant operations and/or premature cable degradation. This modification encompasses the following cable, breaker, and overload heater replacements.

Replacement Item	Circuit	Change From:	Change To:	Powered From:	Load
Cable	CB4	#6 AWG	#2 AWG	1B Turbine Plant - Unit 1D	B Generator Bus Duct Fan
Breaker	CG54	70 AMP	50 AMP	1A ES MCC-Unit 5DL	Heating Coil AH-E-29A
Breaker	CS63	30 AMP	20 AMP	1C ESV MCC-Unit 9BR	Instr. Air Dryer IA-Q-1
Breaker	EA371	30 AMP	20 AMP	120 VAC-PNL VBB-#20	HSPS Cabinet B
Breaker	EA771	30 AMP	20 AMP	120 VAC-PNL VBD-#20	HSPS Cabinet D
Overload Heater	CL41	H51	FH50	1A Reac. Plant H&V MCC-Unit 5A	Reactor Operating Floor Supply Fan AH-E-3A

Safety Evaluation Summary:

The margin of safety defined in the FSAR was not reduced because this modification was designed and installed in accordance with established procedures and criteria such that the safety function of associated systems were not adversely compromised. This modification did not decrease the margin of safety as described in the bases of the Technical Specifications because this modification was a replacement-in-kind. Class 1E circuit breakers powering HSPS cabinets were replaced with Class 1E breakers with lower trip ratings to properly protect the power cable in accordance with its design capacity.

Safety Evaluation Summary:

The margin of safety defined in the FSAR was not reduced because this modification was designed and installed in accordance with established procedures and criteria such that the safety function of associated systems were not adversely compromised. This modification did not decrease the margin of safety as described in the bases of the Technical Specifications because this modification was a replacement-in-kind. Class 1E circuit breakers powering HSPS cabinets were replaced with Class 1E breakers with lower trip ratings to properly protect the power cable in accordance with its design capacity.

****

Modification: Main Generator Relay Protection (BA 412563)

Description of Modification:

This modification replaced the underfrequency relay scheme with a new two-out-of-three relay matrix. The new matrix provides the same trip and alarm functions as the previous relays, and also includes the breakers for the incoming line from the 500KV substation. The primary function of these relays is to protect the turbine generator under low frequency conditions.

Safety Evaluation Summary:

The underfrequency relays perform no safety related functions or affect the operation of any safety related equipment or systems. The two-out-of-three matrix enhances reliability, and eliminates the possibility of a malfunction of the single underfrequency relay causing a turbine and reactor trip. By tripping breakers to separate the turbine generator from the grid under low frequency conditions, which is consistent with original design philosophy, the underfrequency relays prevent damage to the low pressure turbine blades. Thus, there is no increase in the probability of occurrence or the consequences of an accident previously evaluated in the SAR.

****

Modification: Relocation of Decay Heat Closed Cooling Water (DC) Valves DCV-2A/B & 65A/B Controls (BA 412569)

Description of Modification:

The purpose of this modification was to alleviate operator burden for a task that was necessary during all normal plant shutdowns. Previously, valves DC-V-2A/B and DC-V-65A/B are controlled during cooldown using local manual pneumatic controllers located above the decay heat vaults. This required stationing an auxiliary operator at the controllers and calling down from the control room the necessary adjustments. The pneumatic controllers were replaced by current to pneumatic converters driven from manual loading stations located on main control

Description of Modification (Cont'd):

consoles CC and CR. Nuclear safety related solenoid valves and key locked control switches were added to disable the control loops during normal operation.

Safety Evaluation Summary:

There are no accidents described in Chapter 14 of the FSAR which are applicable to this modification. The probability of occurrence of an accident previously evaluated or malfunction of a different type than any previously identified in the SAR was not introduced. Failure of a control component in any one train (A or B) of the new system configuration creates the same condition as it would be created by a failure of a component in any one channel of the existing control system. The consequences of a control component failure would be the inability to remotely adjust the DC water flow and therefore the cooling rate. However, for the LPI operation (i.e., an accident condition), maximum cooling rate would be achieved by de-energized solenoid valves that vent the valve operators. Required cooling rates can also be achieved during control component failure by positioning the valve operators using the handwheel at the valves.
****

Modification: Modification of Penetrations 414, 415, and 416 (BA 418736)

Description of Modification:

The purpose of this modification was to install blank flanges on penetration 414 both inside and outside containment and to replace the existing elbows with flanged end elbows in penetrations 415 and 416. The flanged end elbows can be removed to provide accessible openings for the temporary routing of electrical wires, cables, and hoses required to support outage related work. The modification of penetration 414 also eliminated Leak Rate System and Penetration Pressurization System piping and valves which are no longer used for Integrated Leak Rate Testing (ILRT).

Safety Evaluation Summary:

The blank flanges outside and inside containment are the new isolation barriers for penetration 414 and eliminate valves LR-V-4, 5, and 6 as containment isolation valves. The new flanged spool pieces at penetrations 415 and 416 do not degrade their containment isolation barriers. Containment isolation requirements per the Technical Specifications were met prior to operation following this modification. 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 SAR since the integrity of pressure boundary and seismic boundary of the RB ILRT system are still maintained. Therefore, this modification did not adversely affect nuclear safety or safe plant operation.

Modification: Control Room (CR) Watch Station Upgrade (BA 418746)

Description of Modification:

This modification upgraded the TMI-1 CR Watch Station area located in the north end of the CR. This area was redesigned so that the Shift Supervisor, Shift Foreman, and Shift Technical Advisor can efficiently and eff ctively perform their functions during normal and accident conditions. The Watch Station was redesigned to incorporate new technical equipment such as personal computers, printers, storage devices, and keyboards, in addition to the normal desks, cabinets, and telephones. The Watch Station was also redesigned so that the CR Supervisory team has greater visibility of the operations in the CR by providing the desk area on an 8 inch raised platform. A study area consisting of three (3) cubicles was installed in the CR behind the main panels which is utilized by off-shift CROs.

Safety Evaluation Summary:

The CR Watch Station raised floor configuration was evaluated to ensure that the design maintains it integrity during a seismic event. The raised floor was supplied with a seismic brace assembly for approximately one-third of the internal pedestal supports for additional lateral stability. This modification did not impact the accident scenarios in Chapter 14 o. SAR nor did it adversely nuclear safety or safe plant operations.

****

Modification: Fire Service Water System Connection for the North Office Building (NOB) (BA 419648)

Description of Modification:

This modification provided the fire service water connection to the NOB. It provided a water supply for the fire suppression systems in this building and surrounding yard area. The scope of this modification included the tie-in to the existing 12" underground fire main and extending a branch line form the new Post-Indicator Valve (PIV) at the fire main to the second new PIV located outside the NOB. Two (2) new hydrants were added to the yard north of the building.

Safety Evaluation Summary:

This modification did not impact the safety function of any affected systems. This modification did not increase the probability of occurrence or the consequences of a malfunction of equipment since the installation of the PIV allowed the isolation of the branch line in the event of a line break without loss of water to the remainder of the fire main.

Modification: Fire Service (FS), Service Air (SA), and Domestic Water Tie-Ins for the Outage Support Fabrication Shop (OSFS) (BA 419675)

Description of Modification:

This modification provided the FS water, SA and Domestic Water System connections for the OSFS located on the north side of Fuel Oil Storage Tank FO-T-1 within the protected area fence. The FS water connection provided a supply line for the sprinkler system installed in the building. The SA supply line feeds a header within the OSFS for operation of air tools and other compressed air services. The Domestic Water connection provided a supply line for a safety shower/eye wash station and cold water supply source for fabrication processes.

Safety Evaluation Summary:

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in Chapter 14 of the SAR is not increased. The tie-in to the fire service yard main to supply the OSFS did not impact the design capability of the yard fire main. This modification did not decrease the margin of safety of the affected systems since the system performance requirements of the FS water system are unaffected.

****

Modification: CA-V-2 Fluid Block Piping (CMR-88-074 Revision 1)

Description of Modification:

This modification cut and capped the previously abandoned Fluid Block (FB) connection on CA-V-2. The FB System was placed out of commission and the two (2) FB head tanks were removed and discarded per a previous modification. This modification also provided a potential fluid out leakage path and, in the case of CA-V2, a suspected air in-leakage path. Thus, it was desirable to disconnect the abandoned piping from operational systems and remove any abandoned piping and equipment which could cause access problems.

Safety Evaluation Summary:

The elimination of the FB function from the containment isolation valves was evaluated by SE-412464-001. The safety evaluation for this modification had the limited scope of considering the effect of cutting and capping the abandoned fluid blocking connection on any of the previously fluid blocked containment isolation valves and the removal of sections of the FB piping. The abandoned FB system piping is NITS; thus, the installation of the new cap and the elimination of the piping sections has no safety significance. There are no adverse effects on safety related systems or equipment as a result of this modification.

III. Temporary Jumpers/Modifications

Modification: MS-V1065 Temporary Bonnet Installation (TMM 1)

Description of Modification:

This temporary modification installed a bonnet assembly and new spiral wound gasket in place of the existing bonnet. The purpose of this modification was to provide a leak tight joint between the body and bonnet of MS-V-1065 prior to the Integrated Leak Rate Test (ILRT).

Safety Evaluation Summary:

The temporary installation of the spare non-QC valve bonnet had no affect on nuclear or safe plant operations. The spare valve bonnet assembly was equivalent to the assembly being removed. Following the completion of the ILRT, MS-V1065 was replaced with a new valve under CMR 88-138.

****

Modification: Temporary Bypass of Powdex Sample Control Valves CO-V-80C&F (TMM 2 & 5)

Description of Modification:

This modification allowed on-line cation conductivity monitoring of the effluent from the "C&F" powdex units to continue while replacement valves were procured.

Safety Evaluation Summary:

The system and other components affected by this temporary modification are classified as "Other". Nuclear safety or safe plant operations were not adversely impacted by this temporary modification. The system was returned to its normal configuration.

****

Modification: Temporary Modification to FW-V-95A (TMM 4)

Description of Modification:

The purpose of this temporary modification was to cap-off FW-V-95A (OTSG Drain Cooler Drain) to isolate steam leakage. This valve could not be completely closed due to the valve stem being sheared.

Safety Evaluation Summary:

This modification had no effect on nuclear safety or any adverse affect on plant operations. The steam leak was a personnel safety hazard since it was located

Safety Evaluation Summary (Cont'd):

on the 281' elevation of the RB near the elevator. Thus, this modification enhanced personnel safety. The valve was restored to its original configuration.
****

Modification: Temporary Modification to Provide Alternate Make-up Flow to Industrial Coolers from the Fire Service System. (TMM 6)

Description of Modification:

This temporary modification provided continued water make-up to the industrial coolers (closed and open loops) while the normal source was isolated.

Safety Evaluation Summary:

The equipment/components affected by this temporary modification are not safety related equipment. The potential for loss of spray side of the coolers would not have eliminated all RB cooling. The air side can still maintain RB temperature low enough for several hours to permit repairs of normal make-up path. The system was restored to its original configuration.
****

Modification: Temporary Modification to Fire Sprinkler Pipe (TMM 7)

Description of Modification:

The purpose of this temporary modification was to permit removal of the one inch fire sprinkler pipe from above TG-RV-5 to support overhaul of this component during the 8R outage.

Safety Evaluation Summary:

The affected system was maintained in service after the removal of the sprinkler head. The surrounding heads provided adequate protection for the area. The sprinkler piping in the Turbine Building is classified as "Other." The system was restored to its original configuration.
****

Modification: Temporary Installation of a Larger Suction Line on WT-P-13 to Test Pump (TMM 19)

Description of Modification:

The purpose of this temporary modification was to install a larger suction line on pump WT-P-13 for troubleshooting purposes.

Safety Evaluation Summary:

This modification was temporary to increase the suction pipe to WT-P-13 to troubleshoot the reason for the pump losing suction and becoming air bound. There were no adverse affects on nuclear plant safety or safe plant operations. The system was restored to its original configuration.

****

Modification: Temporary Modification to Repair Vacuum Leak (TMM 23 & 30)

Description of Modification:

A leak was found in the vacuum breaker/auto vent piping for the Intermediate Waterbox. This modification installed a welded plate in the waterbox and a flange outside the condenser to isolate the leak.

Safety Evaluation Summary:

Installation of this TMM has no affect on plant safety. If excessive leakage is identified during plant operation either by loss of vacuum or indication of Circulation Water (CW) inleakage, the loop of the CW system will be removed from service, if required, and the problem corrected. If CW system shutdown is required, vacuum breakers/auto vents are installed on the inlet and outlet waterboxes. The Intermediate Waterbox will be vented through these vents and the tubes in the tube bundle. Proper operation or shutdown of the CW system is not affected. This modification is still in effect.

****

Modification: Provide Temporary Connections to Iron Sampling Rig (TMM 25)

Description of Modification:

The purpose of this temporary modification was to provide connections to iron sampling rig for CE-2, 4, 6, 12, CE-13A-F, CE-15/16, in support of iron transport study.

Safety Evaluation Summary:

The installation of these sample rigs was not important to safety. This installation did not interfere with the ability to sample for secondary coolant activity as required by the Technical Specifications and did not impact any SAR evaluation. The system was restored to its original configuration.

Modification: Temporary Modification to Sluice Duratek Media into the Spent Resin (WDL-T-4) (TMM 26)

Description of Modification:

The purpose of this temporary modification was to provide connections in order to sluice spent duratek media into the spent resin (WDL-T-4) or used precoat tank. The temporary spool piece replaced the spool piece on the suction of WDL-P-10 located on the south wall in the decant/slurry pump room.

Safety Evaluation Summary:

This change had no effect on any nuclear safety related (NSR)component or reduced the margin of nuclear safety on any NSR component. This change did not adversely affect plant operations. The system was restored to its original configuration.
****

Modification: Temporary Modification to the Containment Vessel Integrated Leak Rate Test (ILRT) Air Supply Piping (TMM 40)

Description of Modification:

The purpose of this modification was to temporarily change the Containment Vessel ILRT Air Supply piping service pressure. A low pressure relief valve was blanked and a high pressure relief valve was installed on LR-Q-1 to allow the supply tool air at 100-110 psig.

Safety Evaluation Summary:

This temporary modification did not impact safety related equipment nor did it adversely affect containment integrity. Operating the system during irradiated fuel movements was acceptable since the piping was maintained greater than 100 psig which provided an adequate ventilation barrier from the possibility of RB atmosphere leakage through the leakrate system piping. This system was made a permanent installation.
****

Modification: Temporary Modification to Permit Continuous Instrument Air (IA) Supply Downstream of Air Dryer (IA-Q-1) (TMM 41)

Description of Modification:

The purpose of this temporary modification was to permit continuous IA supply downstream of IA-Q-1 during installation of isolation valves IA-V2104 A&B.

Safety Evaluation Summary:

This temporary modification provided continuous IA supply dow ream of IA-Q-1 and did not change the intended function of the IA System. Th .ttings and hose were designed for pressures greater than the operating pressure of the IA System. The system was restored to its original configuration.

****

Modif cation: Temporary Modification to Install Gagging Devices on Selected Precoat Devices (TMM 58 through 61)

Description of Modification:

The purpose of this temporary modification was to maintain the precoat vessels (PDL-F-1 A&B) operable during a scheduled 1H 480V bus outage. Gagging devices were installed on selected precoat valves to prevent the valves from closing which could have caused loss of precoat.

Safety Evaluation Summary:

This temporary modification did not reduce a margin of safety. The precoat filters are not nuclear safety related and are not required for reactor shutdown. This modification allowed the precoat filters to remain operable or in standby during a brief bus outage which would have otherwise shutdown the cleanup of the BWST and/or the RCBT. The system was restored to its original configuration.

****

Modification: Temporary Modification for Hypochlorite Addition to River Water Systems (TMM 67)

Description of Modification:

The purpose of this temporary modification was to permit biocide application of sodium hypochlorite, in place of gaseous chlorine, to treat the river water systems for microbiological control.

Safety Evaluation Summary:

The river water chlorination system was intended to aid in controlling microbiological growth and subsequent fouling of service water heat exchangers. Hypochlorite is an equivalent method of treatment compared to chlorine gas; therefore, material compatibility with the new chemical was acceptable and its use did not adversely affect safety related equipment. The use of gaseous chlorine has been discontinued

Modification: Temporary Jumper for MDCT Control Panel (EJ 3,8,9,12,16,20,65)

Description of Modification:

This modification installed a temporary RTD converter in the Mechanical Draft Cooling Tower (MDCT) control panel nest 4/slot 9. This change allowed data collection for justification to permanently relocate the River Water (RW) inlet resistance temperature detector (RTD) from its previous location on wing wall to the NR piping.

Safety Evaluation Summary:

The RW inlet temperature instrumentation is classified as NITS. Additionally, test arrangements did not affect the control room station and unit ΔT indication. The system was restored to its original configuration.

****

Modification: Temporary Installation of a Manual Transfer Pushbutton and Lifted Lead for Inverter 1E (EJ 13 and 50; LL 27 and 28)

Description of Modification:

A momentary pushbutton (normally open) was installed between wire 301 on terminal L1 of relay RL8 and terminal 7 of relay RL10 wire 302. Additionally, this modification lifted the 301 wire on terminal L1 of relay RL8. The purpose of this modification is to provide a safe means to retransfer inverter 1E back to ATB from TRB, and to prevent the overheating of resistor R2 on static switch sensing board V082D4.

Safety Evaluation Summary:

Installation of the manual transfer pushbutton on Inverter 1E will not prevent it from transferring to the alternate source (i.e., TRB) in the event of inverter failure or overcurrent/undervoltage condition. The Control Room will be notified that the inverter has transferred to the alternate source by alarm H-1-4 (ICS Power Transfer). The margin of plant safety is not compromised. Inverter 1E does not support a vital bus. This modification is still in effect and is to be completed during a planned outage.

****

Modification: Temporary Jumper for Testing of a Spare MW Transducer (EJ 15)

Description of Modification:

This temporary modification permitted testing of a spare MW transducer to determine the source of spiking of the MW signal.

Safety Evaluation Summary:

This jumper posed no safety concerns for plant operation. ICS-auto power was protected via the 1/4 amp fuse and CA breaker. If CA-auto power was lost, it would not have disturbed normal plant operations. The system was restored to its original configuration.

****

Modification: Provide Temporary Power to Distribution Panel D-22 (EJ 19)

Description of Modification:

The purpose of this modification was to provide temporary power to distribution panel D-22 during the outage of ATB to repair 1E inverter static switch.

Safety Evaluation Summary:

The plant was in a cold shutdown condition during this modification. Temporary loss of the Emergency Notification System (ENS) during installation and removal of this modification did not affect plant safety. The NRC was notified prior to the loss of the ENS phone system. The system was restored to its original configuration.

****

Modification: Temporary Jumper for HSPS Testing (EJ 32)

Description of Modification:

The purpose of this temporary jumper was to allow train A & B modification and subsequent loss of power testing of the HSPS. The jumper maintained control of the MS-V13A/B valves from the Control Room and prevented automatic actuation from the HSPS.

Safety Evaluation Summary:

The HSPS is required to be operable prior to criticality. The temporary jumper was removed and tested prior to criticality. The jumper maintained Control Room operability of MS-V13A/B; if the jumper has failed the automatic actuation would be restored. The system was restored to its original configuration.

****

Modification: Temporary Jumper for Testing of CRD Transfer Switches (EJ 46 and 47)

Description of Modification:

The purpose of this temporary modification was to permit CRD Transfer Switch

Description of Modification (Cont'd):

testing without motor power. CRD Transfer Switches can be operated with the normal and auxiliary power supplies off.

Safety Evaluation Summary:

The plant was in a cold shutdown condition during this modification. The normal and auxiliary power supplies were deenergized, only control power was used for this test. CRD stators were not energized by the testing. Failure of the jumper would have only prevented the relay from being energized which would have prevented any further testing. The system was restored to its original configuration.

************.**

Modification: Temporary Jumper to Maintain Power to RMA-14 (EJ 73)

Description of Modification:

The purpose of this temporary jumper was to ntain power to RMA-14 when the 1CESVMCC was removed from service for preven .ve maintenance. This modification allowed continued compliance with Technical Specification table 3.21-2, item 6.

Safety Evaluation Summary:

RMA 14 was powered from a temporary source when the 1CESVMCC was removed from service for preventive maintenance. The feeder breaker was tagged to prevent inadvertent deenergization. The above actions allowed the plant to be maintained in a normal plant status and not degrade safe operation. The system was restored to its original configuration.

****

Modification: Lifted Lead to provide Electrical Isolation from Exciter Field Breaker (LL 6 & 7)

Description of Modification:

The purpose of this temporary modification was to provide electrical isolation from exciter field breaker during the 8R outage and still permit testing of the field breaker.

Safety Evaluation Summary:

The generator excitation system, which includes the exciter and field breaker, is not a safety related system. This temporary modification was used only during plant shutdown conditions. The system was restored to its original configuration.

Modification: Lifted Leads for Feedwater (FW) Valves (LL 112-116, 118-127)

Description of Modification:

The purpose of this modification was to prevent the closing of FW valves 5A, 5B, 92A, and 92B on an actuation from the HSPS during construction and testing.

Safety Evaluation Summary:

The HSPS is required to be operable prior to criticality. The lifted leads were reterminated and tested prior to criticality. Failure of the lifted leads would have prevented automatic closure of the above FW valves during hot shutdown conditions. The system was restored to its original configuration.

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

Modification: Installation of Elapsed Time Meters [BA 123220 (123240)]

Description of Modification:

This modification involves the installation of elapsed time meters on various Balance of Plant (BOP) 4160V switchgear motors.

Safety Evaluation Summary:

The components affected by this modification do not perform a nuclear safety related function. The installation of the elapsed time meters will allow the Maintenance Department to accurately track equipment run time. Thus, the scheduling of Preventive Maintenance (PM) and balancing of equipment operating time is enhanced by this modification. Since the elapsed time meters only track time equipment run time, their failure to operate will not adversely affect equipment operation.

****

Modification: Use of B&W Rolled and Ribbed Plugs in TMI-1 OTSGs (BA 123253)

Description of Modification:

The purpose of this tube plugging was to remove from service certain TMI-1 OTSG tubes using B&W rolled and ribbed plugs. The plugs are utilized in defective tubes to establish an adequately leak tight pressure boundary between the OTSG primary and secondary side.

Safety Evaluation Summary:

The plug qualification test results demonstrated acceptable primary to secondary leak rates and plug retention capability under normal as well as postulated accident conditions. A pressure test of greater than 10,000 psi performed on both ribbed and rolled plugs indicated no significant plug movement. This demonstrated that a plug is unlikely to be ejected under any adverse conditions since 10,000 psi pressure is greater than the calculated tube rupture pressure for a nominal thickness tube.

Since the plugs meet the same criteria as the remainder of the RCS pressure boundary, the probability of any analyzed accident occurring has not been increased. The design maximum allowable primary to secondary leakage, if assumed to occur for all current and installed plugs, is a small fraction of the Technical Specification limit on total leakage through the OTSG tubes during

Safety Evaluation Summary (Cont'd):

plant operation. Additionally, since the plugs are designed to maintain their integrity under normal, transient, and accident conditions to the same criteria as the rest of the RCS, the margin of safety associated with the pressure boundary is unchanged.

****

Modification: Mini-Mod for Support of Pneumatic Tubing (BA 128086/WA 5573-51130)

Description of Modification:

The purpose of this modification was to upgrade the support configuration of the instrument and control tubing in the Control Building HVAC System located in the A and B Ventilation Equipment rooms.

Safety Evaluation Summary:

This modification resulted in improved reliability of the pneumatic copper tubing and provided a verification that the support configuration conforms with the functional and seismic requirements of the associated instruments and controls. The tubing support attachment upgrade does not directly affect system operations or performance except for the corresponding increase in the reliability of the associated instruments and controls. No changes to the existing instruments or control devices were required.

****

Modification: Makeup (MU) Valves MU-V-11A/B Operator Supports (BA 128086/WA 51137310)

Description of Modification:

The purpose of this modification was to install supports for the valve operators for pneumatic-operated valves MU-V-11A/B. The supports eliminate the extra loading placed on the valve stem by the weight of the operator, enabling the stem to travel without added resistance. This modification allows more reliable and precise operation of the valves.

Safety Evaluation Summary:

The function of valves MU-V-11A/B is to control the flow path of the MU System flow through the filters MU-F-1A/B to tank MU-Y-1 in order to control system chemistry and inventory. This modification improves the reliability and control of valves MU-V-11A/B by reducing the resistance to stem movement. The margin of safety for the affected system was not reduced since the function of valves MU-V-11A/B was not adversely affected. Nuclear safety and safe plant operations were

Safety Evaluation Summary (Cont'd):

not adversely affected by this modification since the function of the system as described in the FSAR was not impacted.

****

Modification: MU-V-36 & 37 Controls Relocation (BA 128087/WA 51142310)

Description of Modification:

Following an ES actuation, Makeup Pump (MU) Recirculation valves MU-V-36 & 37 go closed. To reopen these valves, the operator had to go behind the console to the panel switches. Failure to open these valves could result in damaging the running MU pumps. In order to improve operator response and reduce the possibility of MU pump failure, the control switches for Mu-V-36 & 37 were replaced and relocated to the Control Room console center.

Safety Evaluation Summary:

Nuclear safety was not reduced by this modification since the control logic of the valves was not changed. Safe plant operation is assured since the operator is able to more quickly respond to opening the affected valves as required after ESAS actuation. Relocation of the control switches does not change the existing ESAS logic for these valves. These valves will still close on ES actuation and are still operable from the Remote Shutdown panel.

****

Modification: Inverter Radio Frequency Interference (RFI) Choke Bypass Installation (BA 128087/WA 51143310)

Description of Modification:

This modification added wire jumpers to inverters 1A-E to bypass e RFI inductors (RFI chokes) to improve the voltage at the output of the inverters. The chokes were determined to not be required and caused an undesirable voltage drop.

Safety Evaluation Summary:

Since the RFI choke bypass installation did not reduce the ability of the inverters to supply reliable power to safety related loads, no margin of safety defined in the SAR or Technical Specifications was reduced. The RFI choke bypass installation did not reduce the ability of the inverters to supply reliable power to safety related loads; therefore, there was no potential for nuclear safety or safe plant operations to be adversely affected.

Modification: Emergency Diesel Generator (EDG) Up-to-Frequency Relay Replacement and Air Start Valve Test Switches (BA 128087/WA 51150310)

Description of Modification:

To increase EDG reliability, the obsolete Westinghouse type CF-1 electromechanical up-to-frequency relays were replaced with Brown-Boveri type 81 solid-state relays. Additionally, a test switch was installed to allow separate testing of the air start valves.

Safety Evaluation Summary:

The replacement of the up-to-frequency relays did not alter the deigned operation of the EDGs as described in the SAR. The air start valve test switches are administratively controlled and are used only periodically to test the opening of the air start valves. This modification does not adversely nuclear safety or safe plant operations since it eliminates the contact burning problem and provides a means for testing air start valves.

****

Modification: Test Lights for ES Testing (BA 128087/WA 51152310)

Description of Modification:

IE Notice 88-03 addressed the potential for inadequate testing of relay contacts during safety related logic system functional testing. A review of ESAS surveillance procedures revealed that there were some relay contacts not specifically addressed by Technical Specification surveillance requirements which were either partially tested or not tested. Only one 2 out of 3 ES contact logic combination for the 480V bus 1P and 1S Auto-Start lockout scheme was tested in a refueling interval by the Engineered Safeguards (ES) System Emergency Sequence and Power Transfer test. This modification connected an amber test light in series with the normally open ES auto start lockout contact at bus 1P and 1S. For ES testing, this light is illuminated when the ES contact is closed. Previously, the ES blocking contact to the stop pushbuttons on the skid-mounted control panel was not included in a surveillance procedure. The contact is currently tested by a surveillance procedure and, in order to verify contact operation, an amber test light was connected in series with the normally closed contact.

Safety Evaluation Summary:

The margin of safety as defined in the basis for any Technical Specification was not reduced since these test lights did not alter the designed normal or emergency operation of their respective circuits. Nuclear safety was not adversely affected because these test lights do not perform a safety related

Safety Evaluation Summary (Cont'd):

function and are passive elements in their associated circuits. Safe plant operations were not adversely affected because these test lights provide verification of ES contact operation during ES testing.

****

Modification: Vertical Bus Bars (B/B) Support Upgrade (BA 128087/WA 51157310)

Description of Modification:

The Appendix R GAI Electrical System Studies Report Vol. 1, G/C 2734 had revealed that the short circuit currents on the ES motor control centers (MCC) 1A-ES, 1B-ES, 1A-SHES, and 1B-SHES could exceed the short circuit rating of their vertical B/B. The fault duty rating of these B/B is 22,000A (symmetrical), while the fault current in a worst case scenario could be 24,318A (symmetrical). As a result PSC 87-013 was issued to evaluate the impact on an interim plant operation. The PSC was closed out per JCO-5350-88-310, and LAI-89-9038 was issued to correct this problem. This modification was designed to correct this problem. This modification installed additional B/B insulating supports on the vertical buses of these MCC and rearranged the spacing of the insulators, such that these buses are able to withstand up to 42,000A (symmetrical) short circuit currents.

Safety Evaluation Summary:

The safety function of the MCCs is to supply the ES loads during a Design Basis Accident (DBA). This function was unaffected by this modification since the size of the MCC buses and their normal current carrying capacity were not altered. The additional insulators were seismically mounted and improved the electrical capability of the MCCs to handle electrical fault currents of higher magnitude than before. Thus, this modification enhanced the ability of the MCCs to maintain the electrical power to the ES loads.

****

Modification: OTSG Main Feedwater (MFW) Nozzle Replacement (BA 128135)

Description of Modification:

Main Feedwater (MFW) nozzle plates, made of carbon steel, at other B&W plants have experienced severe corrosion which affects the OTSG performance and could impose risk of damaging the OTSG internal components. Visual inspection of the TMI-1 MFW nozzles also found indication of erosion. The workscope of this modification was to replace the MFW nozzle plates with B&W new design nozzle head which improves the reliability of the MFW nozzles.

Safety Evaluation Summary:

Replacement of the MFW nozzle spray plate with Inconel 600 material and the new configuration which increases the size and reduces the number of holes does not impair the integrity of the nozzle. The new design nozzle improves the corrosion and erosion life of this component. The margin of safety as defined in the basis of the Technical Specifications is not reduced since this modification is considered a replacement in kind which improves the life of the MFW nozzle and does not affect the performance and safety function of the OTSG MFW System. This activity improves the feedwater injection momentum which, in turn, allows a higher maximum level and increases the operational margin of safety.

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Modification: Containment Hydrogen Monitoring System - Valve Relocation (BA 128140)

Description of Modification:

Test connections (valves HM-V05A and HM-V08A) were not readily accessible for Inservice testing (IST) on inner containment isolation valves HM V03A and HM-V04A. This modification rerouted the test connections to a location accessible from platforms that were installed for the inspection of containment ventilation system purge air valve AH-V1C.

Safety Evaluation Summary:

This modification consisted of re-routing inner containment IST test connections for the Containment Hydrogen Monitoring System. The work was performed in accordance with original design codes and classifications and with identical material specifications. This modification did not decrease the margin of safety as described in the applicable Technical Specifications since system functions were unchanged and containment leak tightness was maintained. The modification did not increase the probability of occurrence or consequence of a malfunction of equipment important to safety because the re-routing of the test connections did not affect the safety function of any systems.

****

Modification: Pressurizer Heater Connector Repair (BA 128141)

Description of Modification:

Due to pressurizer heater connector failures, it was necessary to replace the failed connectors. The new connectors are a permanent connection as opposed to the previous removable type connector.

Safety Evaluation Summary:

The pressurizer heaters serve no safety function; therefore, this modification did not increase the probability an accident or malfunction of safety related equipment. The new connectors did not alter the designed operation of the pressurizer heaters. Since the pressurizer heaters perform no safety related function nor have any effect on the operation of safety related equipment, this modification does not adversely impact nuclear safety nor reduce any margin of safety.

****

Modification: Insulation of Industrial Cooler Systems Piping in the RB (Job Order 16581)

Description of Modification:

The purpose of this modification was to install insulation on a portion of the Industrial Cooler system piping in the RB in order to eliminate problems caused by pipe sweating. The insulation reduces the pipe sweating and the resultant pipe coating degradation and exterior corrosion. Thus, this modification minimizes pipe repainting requirements and extends component service life.

Safety Evaluation Summary:

The performance of the Industrial Cooler system was unaffected by this change since no system configuration or operational changes occurred. This modification did not increase the probability of occurrence or consequences of an accident since the seismic classification of the Industrial Cooler system was unaffected, and since the insulation was designed and installed to remain attached to piping during seismic events.

****

Modification: Plant Process Computer Reactimeter (BA 412281)

Description of Modification:

As part of the overall plant process computer upgrade and replacement modifications, the Mod Comp computer with reactimeter was removed. This modification provided installed test connection points and a mobile console for mounting test equipment. The test equipment is used temporarily following refueling and is connected to the plant via temporary plug-in test cables.

Safety Evaluation Summary:

The Plant Computer System is not safe shutdown related, therefore, it does not have a safety function. The Plant Computer System is used by plant operators for alarm monitoring, performance monitoring CRT display, and data logging

Safety Evaluation Summary (Cont'd):

activities. This modification did not alter the safety features of any affected systems. The interface of this modification with the Nuclear Instrumentation System was evaluated in FMEA-TI-602-001 and found to have no impact on safety related functions. Therefore, this modification did not increase the probability of occurrence or consequence of a malfunction of equipment important to safety. This modification did not decrease the margin of safety as defined in the bases of the Technical Specifications because the work did not impact any systems' safety functions.

****

Modification: Intermediate Closed Cooling (ICC) Pump Control Circuit (IC-5-FS) (BA 412511)

Description of Modification:

This modification installed time dela elays to allow sufficient time for the standby ICC pump to re-establish ICC flow before sending a loss of ICC flow signal in the RC pump trip circuit. Time delay relays were also added to the IC-P-1A/B pump control circuits to prevent tripping of the pump when the indicating flags were matched with the control switch following a pump auto start.

Safety Evaluation Summary:

The function of the ICC system is to provide cooling water for various plant components (e.g., letdown coolers, RC drain tank cooler, control rod drive cooling coils). The installation of this modification improves the ability of the ICC system to provide ICC flow to system components and eliminate an erroneous trip signal for the RC pumps. The ICC system is not an Engineered Safety Actuation System (ESAS). The only safety consideration of the ICC is that its Reactor Building isolation valves close as required on an ESAS signal. This modification does not affect the ICC containment isolation valves or the remote shutdown function of the ICC pumps (i.e., IC-P-1A and IC-P-1B). Safe plant operation is enhanced by assuring auto start of the standby ICC pump and eliminating the possibility of a false ICC loss of flow signal to the RC pump trip logic

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Modification: Computer Alarms for Electro H raulic Control ((BA 412512/WA 34512010)

Description of Modification:

This modification installed a new digital computer point to provide the control room with an audible computer alarm when an EHC electrical malfunction signal is generated.

Safety Evaluation Summary:

The safety function of the plant computer is to display and log data as required by R.G. 1.97. This modification affects the plant computer system by adding a digital alarm input to log. The EHC has no safety related function. The implementation of this modification does not adversely affect nuclear safety or safe plant operations. The current and new alarms do not interlock with any safety related equipment.

****

Modification: Main Feedwater Pump (MFWP) Bearing Oil Pressure Trip to Plant Computer (BA 412512/WA 34512011)

Description of Modification:

Safety Performance and Improvement Program (SPIP) Recommendation TR-014-MFW addressed the installation of a monitoring system for the MFWP trips. Per the referenced recommendation, this modification connected the MFWP bearing oil pressure trip to the plant computer.

Safety Evaluation Summary:

The margin of safety as defined in the basis for any Technical Specification was not reduced since the plant computer and the MFWPs are not in the basis of any Technical Specification. Safe plant operation was not affected since a set of spare contacts was used. The setpoint of the pressure switch was not changed by this modification. The pressure switches connected to the computer do not interlock with any safety related equipment that is addressed in Chapter 14 of the SAR.

****

Modification: Class 1E Diesel Genera_ Synchronous Check Relay (BA 412512/WA 34512012)

Description of Modification:

The B&W Owners Group Safety and Performance Improvement Program determined that in order to reduce the risk of damage to the Class 1E diesel generators due to operator error, synchronism check relays should be installed. These relays would prevent an interlock in the EG-Y-1A and 1B Class 1E diesel generator breaker close circuits, which would automatically prevent breaker closing in an out-of-sync condition. This modification installed two (2) Brown Boveri Type 25S sync check relays in control room panel CR.

Safety Evaluation Summary:

The margin of safety as defined in the basis for any Technical Specification was not reduced because these relays eliminated the potential risk of damage to EDGs due to operator error when synchronizing and closing generator breakers. 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 because the relays are seismically qualified and mounted, and existing wiring was used to connect the permissive contacts to the manual control circuit of the breakers.

****

Modification: Vent Valve Installation for Testing Air Accumulators (BA 412512/WA 34512013)

Description of Modification:

The scope of this modification was to install a vent valve on the air supply line between the check and isolation valves for those air operated valves that are required to be tested. The Instrument Air (IA) lines for the following valves were modified to allow accumulator testing: IC-V3,4; IC-V6; and MU-V20. The accumulators for HD-V4, FW-V7A/B, and MS-V4A/B were also tested; however, no modifications were required for these valves.

Safety Evaluation Summary:

The loss of IA would not adversely affect any safety related system, because all the safety related air operated valves would fail in a safe position or have accumulators that would maintain the valve in a safe position. The accumulator and associated tubing, required to provide containment isolation, is safety related. This modification installed vent valves which allow testing of these components. This modification was outside the safety related boundary. Failure of the modified tubing, therefore, would not have increased the probability of occurrence or the consequences of an accident previously evaluated in the SAR.

****

Modification: Reactor Coolant Pump (RCP) Lube Oil System Upgrade (BA 412512/WA 34512014)

Description of Modification:

The RCP motor lube oil system experienced oil leakage and difficulties when filling the oil reservoirs (backup and spillage of oil, slow fill rate, and lack of easy-to-read normal level reference marks). As a result , the following modifications were performed:

Description of Modification (Cont'd):

- o Upgraded the piping/tubing to welded joints as much as possible to minimize leakage.
- o Redesigned the upper and lower reservoir local fill stations for easier and more rapid filling capability during refueling outages for oil addition capability during forced shutdowns.
- o Installed remote fill stations for on-line oil addition capability during plant operation.
- o Replaced existing oil level sight gauges with ones that are easier to read.
- o Replaced level switches with new upgraded design.
- o Added capped drain on RCP oil shield drain tank.
- o Installed new high pressure hoses for the lube oil lift piping.
- o Enlarged/relocated oil drip pans where required as a result of piping/tubing modifications.

Safety Evaluation Summary:

The purpose of this modification was to improve the RCP motor lube oil system reliability and to avoid, to the extent possible, plant trips and transient conditions that could be initiated by excessive oil leakage. This modification did not affect the safety function of any associated systems; therefore, nuclear safety or safe plant operations were not adversely affected. The seismic design of the lube oil system and the ability to collect and contain any leakage via seismically designed drip pans and leakage collection piping were maintained.
****

Modification: Installation of Letdown Piping Shielding (BA 412528/WA 30121310)

Description of Modification:

The purpose of this modification was to provide permanent shielding around the Reactor Coolant Letdown Piping above the Makeup (MU) Filter 2A/B cubicle at elevation 281' of the TMI-1 Fuel Handling Building (FHB). This shielding reduces personnel exposure during filter changeout operations which require maintenance personnel to work close to the piping.

Safety Evaluation Summary:

The added support structure and lead shielding added a relatively small mass to the reinforced concrete wall. Attachment of the support structure to the wall utilizing concrete expansion anchors did not degrade the structural integrity of the wall. The support structure and lead shielding have no safety function and are not required to function during or after an SSE, but are required to maintain passive integrity. Therefore, they are classified as Seismic Class II.

****

Modification: Installation of Fire Damper FD-66 Access Ladder (BA 412528/WA 30162310)

Description of Modification:

This modification provided a permanent means of access to FD-66 located in the MU-P-1C cubicle. This access improves access to FD-66 for purposes of surveillance/maintenance activities and reduces exposure to potential safety hazards by providing permanent provisions for climbing up to the damper.

Safety Evaluation Summary:

The ladder and supports have no nuclear safety function and are not required to function during or after a Safe Shutdown Earthquake, but they are required to maintain passive integrity. Therefore, they are classified as Seismic Class II, anti-falldown. This modification does not impact nuclear safety or safe plant operations.

****

Modification: Installation of Protective Covers for Cable Trays 4.1 ..d 496 (BA 412528/WA 31093310)

Description of Modification:

This modification provided physical protection for the pressurizer heater power cables located near the heater bundles, elevation 312'-7". The previous configuration exposed the cables to potential damage and general wear-and-tear during maintenance and inspection traffic around the pressurizer.

Safety Evaluation Summary:

This modification did not directly affect any plant systems, subsystems, or components. It does indirectly affect the RCS in that the power cables to the pressurizer heaters receive the benefit of physical protection and therefore an increase in reliability. The tray covers and supports have no safety function and are not required to function during or after the SSE, but are required to

Safety Evaluation Summary (Cont'd):

maintain passive integrity. Therefore, they are classified as Seismic Class II Anti-Falldown.

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Modification: Pump Pecirculation Flow Meters for Inservice Testing (BA 412531)

Description of Modification:

The purpose of this modification was to provide diesel fuel local flow measurement for pumps, DF-P-1A-D in the Emergency Diesel Generator (EDG) fuel system (phase 1) and to install in-line flow meters for the Boric Acid Recycle Pumps, WDL-P-13A/B, and the Boric Acid Pumps CA-P-1A/B (phase 2). This modification resolved a commitment to the NRC regarding in-service testing requirements for the diesel fuel transfer pumps.

Safety Evaluation Summary:

The installation of the local flow elements and flow indicators did not affect the performance of the EDG fuel system as it only provided indication and had no control function. The flow indicators and elements provide local flow indications to the IST program requirements in accordance with the ASME Boiler and Pressure Vessel Section XI code. This modification did not adversely affect nuclear safety or safe plant operations, or decrease a margin of safety since operation and function of the diesel generators was not affected.

****

Modification: Reactor Building Maintenance Platforms (BA 412540)

Description of Modification:

This modification installed platforms at the Reactor Coolant Pumps RCP-1B&C and at the Reactor Coolant Drain Tank (RCDT) to provide safe, permanent work areas of sufficient size to support maintenance activities for these components.

Safety Evaluation Summary:

This modification affected the secondary shield walls and their attached compartment walls. No other systems, sub-systems, structures, or components were directly affected by this modification. The additional load of the platforms plus equipment and personnel loads on the secondary shield wall and the RB structure were analyzed and found to be acceptable. The platforms and their supports are passive during operations of the plant and did not inhibit the ability to achieve containment integrity. The additional structure loading due

Safety Evaluation Summary (Cont'd):

to these platforms did not reduce the margin of safety of these supporting structures.

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Modification: OTSG A&B Upper Manway and AH-V1C Platforms (BA 412546)

Description of Modification:

This modification installed platforms at the AH-V1C rge valve and extensions to the existing platforms for the OTSG upper manways to provide safe, permanent work areas of sufficient size to support maintenance activities for these components.

Safety Evaluation Summary:

This modification affected the secondary shield walls and their attached compartment walls. No other systems, sub-systems, structures, or components were directly affected by this modification. The additional load of the platforms plus equipment and personnel loads on the secondary shield wall and the RB structure were analyzed and found to be acceptable. The platforms and their supports are passive during operations of the plant and did not inhibit the ability to achieve containment integrity. The additional structure loading due to these platforms did not reduce the margin of safety of these supporting structures.

****

Modification: OTSG Skirt Manway Access Enlargement (BA 412562)

Description of Modification:

This modification enlarged a portion of the circular access opening to the OTSG lower manway to improve personnel and material access to the lower head for maintenance and repair operations. This also added four (4) sets of four (4) small holes in the skirt near the access opening to allow the temporary mounting of radiation shields for use during non-power operations.

Safety Evaluation Summary:

The proposed modification has no effect on the safety function of the OTSG since there is no impact on the ability of the component to provide a fission product barrier. Since the modification was wholly external to the OTSG, the system performance was unaffected. This modification did not introduce any new operating mode or subject existing components to new operating challenges outside

Safety Evaluation Summary (Cont'd):

of the original operating, safety, and design bases. No new unanalyzed accidents or malfunctions were introduced, nor made either more probable or of greater consequence.

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Modification: Auxiliary Fuel Handling Bridge (AFHB) Equipment Upgrades (BA 412564)

Description of Modification:

The purpose of these modifications were to:

- o Install an Inching Motor with the control center mounted on the auxiliary bridge tower.
- o Install programmable geared limit switches with the control console mounted on the AFHB tower.
- o Install fixed lighting on the main and auxiliary bridges to assist in reading the ZZ position tapes and position scribe marks on the bridge operating floor next to the rails.
- o Install a reel type manually operated camera cable storage assembly permanently mounted on the auxiliary bridge trolley.
- o Rewire the interlock status lights on the main bridge control console. Past bridge operations had indicated that the lights were working improperly.

Safety Evaluation Summary:

The fuel handling system is designed to provide a safe, effective means of storing and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after post-irradiation cooling. The system is designed to minimize the possibility of mishandling or improper operations that could cause fuel assembly damage, potential fission product release, or both. These modifications did not increase the probability of occurrence or consequence of a malfunction of "Regulatory Required" equipment. The improved upgrades increase fuel handling equipment reliability while decreasing equipment maintenance. These modifications did not create a possibility for an accident or malfunction of a different type than any previously evaluated identified in the SAR because operating modes and system control were unaffected.

Modification: Main Feedwater (MFW) Flow Element Replacement (BA 412565)

Description of Modification:

This modification replaced the MFW flow elements (SP-8A/B-FE) with new flow elements to increase the accuracy of flow measurement and to decrease the degradation of measurement accuracy. This modification precludes the necessity for unnecessary reductions in plant performance caused by artificially elevated estimations of reactor power output.

Safety Evaluation Summary:

Following replacement of the venturis, during power escalation, flow testing was performed to confirm the accuracy of the signals developed by the newly installed venturis (see Lithium Tracer Testing Activity under Section I, "Tests and Experiments"). Nuclear safety or safe plant operations were not adversely affected by this modification since the safety functions of the affected systems were not impacted. This modification did not increase the probability of occurrence or consequence of an accident previously evaluated in the SAR. This determination was based on a review of the SAR with particular emphasis on Sections 5.1, 5.4, 10.5, and 14. Replacement of the flow element did not affect the function of the FW system.

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Modification: Control Room Noise Reduction (BA 412566)

Description of Modification:

This modification installed a control room global silence button in order to reduce control room noise during a major plant transient and improve the ability of the Control Room Operator (CRO) to communicate and respond to a transient. A chime system was also installed to annunciate when any main control room alarm is reset along with a reflash logic system for the RMS interlock switches. The reflash logic system resulted in the alarm window remaining energized when any RMS interlock switch is in defeat and will alarm again if any other interlock switched are placed in defeat.

Safety Evaluation Summary:

The safety function of the annunciator system is to alert the CRO to the status of safety related equipment and/or the potential of exceeding pre-determined limits. A global silence pushbutton was installed that silences all annunciator alarms for a pre-determined time when the pushbutton is activated. Eliminating distracting annunciator noise during the first minutes of a major plant transient significantly improves the CRO's ability to communicate and respond to the transient. The global silence button could introduce the risk of missing an

Safety Evaluation Summary (Cont'd):

alarm. However, the button will silence the alarms only for a pre-determined time period and the administrative controls governing the use of the button during post-trip conditions will effectively limit this risk. Using the button during a major plant transient outweighs the potential of missing an alarm. The operator can determine what alarms are in during and after global silence time frame by visual inspection of flashing alarm windows. The chime system will allow the operator to audibly distinguish between when an alarm is received and when it is reset.

The accidents analyzed in Chapter 14 of the SAR do not take credit for the annunciator or computer systems. Credit is taken for RMS actuated isolation of release paths to mitigate the consequences of an accident. The changes to the RMS interlock defeat alarm did not affect the ability of the RMS to shutdown any equipment used for isolating a release path since work was only performed on the alarm circuits. The global silence pushbutton and associated relays do not interlock with any safety related equipment that would be used to mitigate or monitor the consequences of an accident as described in Chapter 14 of the SAR.
****

Modification: Replacement of Emergency Feedwater (EFW) Flow Elements (BA 412571)

Description of Modification:

Measurement of EFW flow was previously accomplished with type ANR-76-C21-CSS annubars in both the A and B loop. Flow indication from the annubars was considered unreliable due to suspected air entrainment in the sensing lines. This modification replaced the annubars with flow venturis of proven design which provide a 500 inch ΔP at 600 GPM and 680 inch ΔP at 700 GPM which is maximum venturi design flow.

Safety Evaluation Summary:

The margin of safety defined in Licensing Basis Documents was not reduced as a result of this modification. System operating parameters and design were not changed except for replacement of the flow element which improves accuracy of flow sensing and indication. Increased system ΔP does not affect the capability of the system to deliver required flow under all conditions. Nuclear safety or safe plant operations were not adversely affected by this modification because the safety functions of the affected systems were not impacted.

Modification: Kidney Filter System Modification (BA 412575)

Description of Modification:

The purpose of this modification was to provide a more effective RB atmospheric cleanup capability (i.e., iodine removal) at TMI-1 by modifying the configuration of the RB Purge and Kidney Filter System. The scope of the modification involves the separation of the kidney filter inlet duct (to AH-E-101) from the RB purge inlet duct (supplied through AH-V-1C), and the rearrangement of the kidney filter duct work to enhance iodine removal during purging operation. This modification allows concurrent operation of the kidney filter and the purge supply without the systems impeding each other.

Safety Evaluation Summary:

The function of the Kidney Filter System (Atmospheric Cleanup System) is to reduce the airborne radioactivity levels in the RB during normal plant operations, including outages. This modification did not increase the probability of occurrence or consequence of an accident since the Kidney Filter System performs no safety related function and the operation and function of the Purge system following an accident (i.e., containment isolation) is unaffected by this modification. This modification improves the ability of the Purge and Kidney Filter System to reduce airborne radioactivity levels in the RB during an outage.

****

Modification: Provide Separation Between Emergency Diesel Generator (EDG) Engineered Safety (ES) Relays and Non-ES Components (BA 412583)

Description of Modification:

The voltage and frequency relays for both EDGs were previously connected to non-ES components. This modification provided proper separation between ES and non-ES components by:

1) Installing four (4) 6 amp fuses in EDG A control cabinet and three (3) 6 amp fuses in EDG B control cabinet;

2) Circuits RY56 and RY57 were spared and the new circuits RY56A and RY57A were routed in the ES cable trays; and

3) The following circuits were reclassified to ES circuits from non-ES circuits: RY53, RY55, RZ53, RZ53B, RZ55, and RZ 56. These circuits include the differential protection and relay 46G circuits.

Safety Evaluation Summary:

The margin of safety as defined ' . the SAR was not degraded by this modification because the fuses and rerouting circuits RY56A and RY57A provide separation from non-ES circuits, and adequate separation between channel A and channel B is maintained. The failure of the new components in an open circuit condition does not adversely affect the Class 1E function because it affects only the metering function. However, failure of the fuse block in dead short position during loss of offsite power could disable the EDG breaker closing circuitry by blowing the upstream fuse. Therefore, the fuse blocks have been dedicated for NSR use by test/inspection. Thus, this modification improves the system by separating the non-ES components from the ES components.

****

Modification: Installation of a Stop Function on Control Room Panel PL for Fire Pump FS-P-2 (BA 418695)

Description of Modification:

This modification installed a stop function on Control Room panel PL for FS-P-2. This will allow operators to stop the pump remotely from the Control Room after spurious star fire service malfunctions, and surveillance testing.

Safety Evaluation Summary:

The implementation of this activity did not adversely affect nuclear safety or safe plant operations because FS-P-2 is not classified as Nuclear Safety Related and, although it is fed from a safety related power supply, the electrical power configuration was not changed. The new control cable is routed through the safety related tray and is pr rly protected against electrical faults. Remotely stopping the fire pump does not increase the possibility for a different type accident or malfunction.

****

Modification: Replacement of Regulating Transformers for 120VAC Distribution Panel TRA and TRB (BA 418695/WA 39695010)

Description of Modification:

This modification replaced the four (4) 10KVA regulating transformers feeding panels TRA and TRB with two (2) 20KVA regulating transformers. The previous regulating transformers couldn't maintain the required voltage output (120VAC) to the Regulated Voltage Panels TRA and TRB and to the Station Inverters 1A and 1E.

Safety Evaluation Summary:

This modification did not affect the normal function of the 120VAC Regulated Distribution Panels TRA and TRB since the loads supplied by these panels are improved by the increased voltage output of the new 20KVA transformers. The increased voltage output also improves the performance of the inverter static switch. The margin of safety as defined in the SAR was not reduced because the new 20KVA transformers will continue to support instrumentation, control, and power loads from panel TRA and TRB. This modification does not increase the probability of occurrence or consequence of a malfunction of equipment important to safety previously evaluated in the SAR because there was no change in transformer size and no new loads were added to panels TRA and TRB.

****

Modification: Fluid Block (FB) Valves/ Penetration Pressurization (PP) Valves for Purge Interspace Pressurization Electrical Removal & RR-V4C/4D PCR Panel Indication Relocation (BA 418695/WA 30695010)

Description of Modification:

The FB System was functionally removed from service by Technical Functions project BA 412465; however, associated electrical controls and hardware were not disconnected or removed. This modification disconnected the controls and indication for the FB valves and spare/remove associated electrical circuits. Interspace pressurization devices PP-V167, 169, and 170, and pressure switch PS-677A was disconnected with associated circuits spared/removed by this modification. The PCR Panel indication for MU-V3 was modified such that on loss of DC control power the indication will fail to show the valve as open instead of closed to correspond to the failure position for MU-V3 on loss of DC control power.

Safety Evaluation Summary:

The FB System was previously removed from service and performs no safety function as described in the SAR. The electrical removal was accomplished in a manner such that the associated RB isolation valve position indication relays remain functional when the pilot solenoid valve is disconnected. The automatic functions for the purge interspace pressurization were previously removed and evaluated as acceptable by SE 128124-001. This modification electrically disconnected and removed components that were mechanically disabled. The safety function of other PP components was altered by this modification.

Modification: Control Rod Drive (CRD) Test Jack Panel Installation (BA 418695/WA 34695110)

Description of Modification:

This modification installed a jack panel in system logic cabinet SL-6. This panel is used to connect recorders for both Rod Drop Time testing and position indication tube reed stack testing. This modification permits testing of the position indication and 24% zone reference switches to be performed in a reduced period of time.

Safety Evaluation Summary:

This modification did not affect the safety functions of the CRD System since the jack panels are used only for testing and calibration purposes. System performance is enhanced by this modification since these panels reduce the possibility of terminal board and screw failure. The installation of the test jack panel did not reduce a margin of safety as defined in the SAR since the panels are only used for the connection of recorders during testing of the position indicators and 25% zone reference switches. Additionally, the new components were installed in cabinets which do not contain safety related components; thus, there is no potential for the new components to affect the safety related components of the CRD System.

****

Modification: Installation of Monorail Beam and Supports for MU-F-2A/B Letdown Prefilters (BA 418695/WA 38695110)

Description of Modification:

This modification installed a monorail beam and supports to replace the previous jib beam used for changeout of MU-F-2A/B Letdown Prefilters. The monorail will facilitate a motorized hoist and allow handling of the shielding pig from the general access hallway adjacent to the MU-F-2A/B cubicle. This modification improves the efficiency of filter changes and subsequently reduces person-rem exposure.

Safety Evaluation Summary:

This modification did not adversely affect any safety related systems. Indirectly, the monorail affects the Make-Up and Purification System for handling MU-F-2A/B Prefilters and the Auxiliary Building (AB) by virtue of structural attachments. Structural attachment to the AB concrete walls at approximate elevation 296'-8" is designed in accordance with AISC Structural Code. The beam supports were also designed for Seismic II , anti-falldown. For personnel safety considerations a factor of safety of approximately 5.0 against failure was

Safety Evaluation Summary (Cont'd):

attained for the passive lifting components. The overall rated load of 4000 pounds exceeds the heaviest potential lift.

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Modification: Reactor Vessel (RV) Head Fixed Lifting Pendant (BA 418704)

Description of Modification:

This modification installed permanent lifting pendants and supports in the TMI RV head and service structure. These pendants replaced the previous lifting cables installed each time the head was lifted.

Safety Evaluation Summary:

An analysis was performed by B&W that demonstrated that the pendants have no effect on the seismic performance of the RV head and service structure. Nuclear safety or safe plant operation were not adversely affected by this modification. No additional stresses were induced in the reactor coolant pressure boundary in steady state, transient, or seismic conditions. The RB integrity is ensured since there was no potential increase in RB temperature, increase in hydrogen generation due to an accident, or increase in pressure in the RB during a LOCA as a result of this modification. There was no adverse affect on the RB cooling functions during and following a LOCA due to debris generation.

****

Modification: Reactor Coolant Pump (RCP) Articulated Arms Installation (BA 418747)

Description of Modification:

This modification installed an articulated arm assembly on each RCP motor stand. This modification was made as a maintenance enhancement to the RCP motors. The arms provide a more efficient means of horizontally transferring RCP parts in and out of the motor stand during RCP seal work.

Safety Evaluation Summary:

The articulated arms serve to enhance maintenance and perform no safety function. However, the arms are designed to remain secure during seismic events equal to or greater than the original equipment specification loads. The arms are also mounted to the exterior of the RCP motor stand and do not affect the operation of the RCP or other plant components. Thus, this modification does not adversely affect nuclear safety or safe plant operations.

Modification: Relocate Transzorbs (CMR-88-075)

Description of Modification:

Makeup valves MU-V-3 and 18, and Intermediate Cooling valve IC-V-3 had Transzorbs mounted locally; therefore, the Transzorbs were required to be on the Environmental Qualification (EQ) list. The Transzorbs suppress EMF produced by the induction of de-energizing relay and solenoid coils. Also, Transzorbs connected to these valves had to be tested for low leakage rate. In order to remove this requirement and help preclude the possibility of non EQ being installed, the Transzorbs located at MU-V-3 and 18 were removed from the EQ environment and the Transzorb for IC-V-3 was relocated to the Remote Shutdown Transfer Switch.

Safety Evaluation Summary:

The safety function of the affected MU valves is isolation containment. The valves are automatically closed by an ES signal. The safety function of the affected IC valve is isolation containment and is automatically closed on any ESAS signal. The function of the Transzorb is protection of contacts and coils located within the circuit.

The above safety functions of the affected systems were not altered. The isolation containment valves still perform their required safety function. Moving the Transzorb at IC-V-3 did not electrically alter the circuit. Removing the Transzorb on MU-V-3 and 18 did not reduce the protection of the contacts. The remaining configuration is such that the remaining Transzorb protects all circuits.

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Modification: Control Rod Drive Motor (CRDM) Stator Cooling Lines (CMR 90-019)

Description of Modification:

This modification replaced existing Intermediate Closed Cooling Water (ICCW) hard tubing lines to the CRDM stators with flexible hoses and quick connect couplings. The previous cooling water lines between the CRDM stator cooling coils and the CRD service structure consisted of rigid tubing and threaded couplings. This made reconnection of the cooling water lines difficult and tedious. This change was accomplished on only 14 CRDM stators during 8R. The remainder of the cooling lines will be converted during future outages.

Safety Evaluation Summary:

The function of the CRDM stator cooling coils is to cool the CRDM motor stators during CRDM operation. Each stator is individually cooled by a cooling coil

Safety Evaluation Summary (continued):

which has cooling water supplied from the ICCW. The installation of the flexible hose did not alter the cooling water supply, return and isolation functions of the ICCW system. This modification did not adversely affect nuclear safety or safe plant operation since the function of the CRDM motor and the ICCW system was not affected. The new hose has automatic shutoff on both the stem and body assemblies minimizing water spillage during and after uncoupling.

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Modification: RV Head Access Hole (CMR 90-030)

Description of Modification:

This modification installed an access hole in the Service Structure Support Assembly (SSSA) for improved access to the top surface of the Reactor Closure Head.

Safety Evaluation Summary:

The installation of the access hole did not adversely affect the structural adequacy nor alter the function of the SSSA. The hole was covered with a removable bolted cover to re-establish the physical boundary of the support skirt.