Stephen E. Quinn
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 734-5340

December 29, 1995

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

SUBJECT: 10 CFR §50.59(b) Report for Indian Point Unit No. 2

Pursuant to 10 CFR §50.59(b)(2), enclosed please find a report of the changes, tests and experiments conducted at Indian Point Unit No. 2 during the period from January 1, 1994 through June 30, 1995.

Should you or your staff have any questions regarding this matter, please contact Mr. Charles W. Jackson, Manager, Nuclear Safety and Licensing.

Very truly yours,

Enclosure

cc: Mr. Thomas T. Martin
Regional Administrator - Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Francis J. Williams Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
US Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
Buchanan, NY 10511

50-247 INDIAN POINT 2 CECONY

CHANGES, TESTS AND EXPERIMENTS FOR THE PERIOD FROM JAN. 1, 1994 THRU JUNE 30, 1995

REC'D W/LTR DTD 12/29/95...9601160029

ATTACHMENT

10 CFR 50.59 (b) REPORT

CHANGES, TESTS AND EXPERIMENTS FOR
THE PERIOD FROM
JANUARY 1, 1994 THROUGH JUNE 30, 1995

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
DECEMBER, 1995

DESCRIPTION OF CHANGES, TEST OR EXPERIMENTS
MADE PURSUANT TO 10CFR50.59
January 1, 1994 - June 30, 1995
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**DESCRIPTION OF CHANGES, TEST OR EXPERIMENTS
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1. Facility Organization Change: Fire Protection, SE-94-051-EV

This administrative change resulted in a transfer of the reporting relationship of the Fire Protection Administrator from the Manager, Site Protection to the Manager, Plant Engineering. In addition, the title of the Fire Protection Administrator was changed to Fire Protection Specialist. This change was consistent with the Con Edison's System Engineering program implementation, which involved consolidation of system engineering matters in one organization. The change did not result in a decrease in the effectiveness of the Fire Protection Program. A review of the licensing basis for the Fire Protection Program verified that the change to the organization continued to meet applicable NRC guidelines. All previous functions affected by this organization change continue to be performed. The operation and reliability of systems and equipment required to prevent and/or mitigate accidents, fires or malfunctions were not affected as these systems and equipment and the controls on their use were not changed. This change did not involve an unreviewed safety question.

2. Organization Chart Change, SE-94-257-EV Rev. 1

This administrative change realigned support organizations and re-titled some positions for the Indian Point Station. The position of General Manager, Nuclear Power Generation (NPG) was retitled to Plant Manager. All the functions which previously reported to the General Manager, NPG continue to report to the Plant Manager. The functions performed by the organizations formerly designated to be under the General Manager of Technical Services were reorganized into three departments reporting directly to the Vice President of Nuclear Power: System Engineering and Analysis, Site Services and Independent Safety Review. The Plant Engineering organizations were realigned in two sections: Systems Engineering and Engineering Analysis. This reorganization did not impact the individual functions of the sections. The reporting of the Independent Safety Review (ISR) department directly to the Vice President, Nuclear Power maintained the independence crucial to that organization's activities. The Administrative Services and Staff organization was replaced by the Site Services organization and the Nuclear Training organization. Non-safety related functions of Administrative Services were realigned to the Site Services organization. Nuclear Training, which formerly reported to the Manager of Administrative Services was realigned to report directly to the Vice President, Nuclear Power. This change did not represent a reduction in commitments or decrease in effectiveness with regard to the considerations contained in 10 CFR 50.54, since the plans referenced in that regulation (e.g., Emergency Plan, Security Plan, Quality Assurance Program, etc.) were maintained by procedures and processes which were either unaffected or merely adjusted to reflect the new organizations. The changes did involve an increase in the span of control for the Vice President which was well within normal manager capabilities. The reliability of systems which are relied upon to prevent and/or mitigate accidents or malfunctions were not affected. The management controls in place for the implementation of programs and procedures were maintained. This change did not involve an unreviewed safety question.

3. Modification of Control Rod Drive Fan Cooler Motor Bearing Lubrication and Bearing Seal, NS-2-83-111

This change involved a modification associated with the Control Rod Drive (CRD) fan cooler motor inboard bearings. The modification provided for a grease tube to carry grease directly to the area of the inboard bearings and a slotted bearing end cap to facilitate even grease distribution. A new outboard bearing cap with a grease retention seal was also provided. This modification improved the performance and reliability of the CRD system by providing an effective means for retaining grease in the outboard bearings for longer periods and reducing the potential for contaminants in the outboard bearings. There was no effect on the site fire protection plan from this modification. This change did not involve an unreviewed safety question.

4. Modernize Aging Radiation Monitor System, SE-93-288-MD

This change involved the removal of iodine monitors and the replacement of radiation monitoring equipment at Indian Point Unit No. 1 including the sphere foundation drain sump monitor. The purpose of the sphere foundation sump is to detect radiation in the sump and alert the operators in the Central Control Room (CCR) to take corrective action. The replacement monitor included monitoring skids, annunciator alarms, recorders, cable and conduit. The iodine monitors were no longer needed and were therefore removed. None of these monitors were related to any accident evaluations or affect any equipment that has been previously evaluated. Materials used are the same as those used in the previous system. The additional electrical loads were added to non-vital electrical busses and did not impact the Emergency Diesel or Battery loading. Cables were installed according to separation and tray loading standards. These monitors are not required to be functional following a seismic event although some conduit runs to the Radiation Monitoring Computer were seismically restrained. The Environmental Qualification (EQ) program was not impacted by this change. Fire Protection Program Plan requirements were met. This change did not involve an unreviewed safety question.

5. Installation of Nitrogen-16 (N-16) Radiation Monitors, SE-92-202-MD

This change involved the installation of Nitrogen-16 radiation monitors near each of the four main steam pipes in the pipe bridge area between the auxiliary boiler feedwater pump building and the turbine building. Other associated equipment also installed included fuse disconnects, a N-16 analyzer cabinet, reflash alarm module, air conditioner for the cabinet, an annunciator window in the CCR and interconnecting cables and conduits. The purpose of the N-16 monitors is to detect the presence of a small tube leak in any of the steam generators. These monitors provide leak rate indication and alarm to alert the operator to take immediate and long term action in the event of a small tube leak in a steam generator. These monitors do not perform any control function and are only intended as backup indication to existing leak detection monitors. No new safety-related failure modes are introduced and no accident analyses in the UFSAR are affected. Fire protection, environmental qualification, seismic integrity and emergency planning are not adversely affected by this modification. This change did not involve an unreviewed safety question.

6. Residual Heat Removal (RHR) Pump Surveillance Instrumentation, NS-2-87-008

This change involved a modification to install instrumentation to monitor suction and discharge pressure of the RHR pumps. The purpose of this modification was to provide information to CCR personnel regarding the operating conditions of the RHR pumps during plant cooldown and draindown modes. Indicators and computer display screens were provided for Central Control Room (CCR) indication of decay heat removal system performance pursuant to NRC's Generic Letter No. 88-17. Bearing temperature, vibration, and key phasor sensors and associated cable routings were retired in place since these functions were determined to be no longer necessary due to alternate capability to monitor pump performance. CCR alarms were changed to reflect this modification. Since the equipment installed in this modification is not designed or qualified to function during a loss-of-coolant accident (LOCA) measures have been taken to preclude its use during LOCA mitigation activities to avoid the possibility of providing an operator with potentially misleading information. This modification does not adversely affect safety systems through physical or electrical interactions nor does it decrease the existing level of fire protection. The changes to the CCR are consistent with human factor engineering requirements. This change did not involve an unreviewed safety question.

7. Filling and Venting of the Safety Injection System, SE-91-148-PR Rev. 1

   This change involved a revision to a temporary operating instruction (TOI) used to control the filling and venting of the residual heat removal (RHR) and safety injection systems. This revision added a new section to the TOI on venting safety injection piping and pumps using gravity fill from the refueling water storage tank (RWST) and flow from the RHR pumps and provided for a ten second operation of the SI pumps for additional venting. The venting evolution controlled by this TOI was limited to periods when the reactor was in a defueled, cold-shutdown condition. All equipment and systems were maintained within their design envelopes and there was no impact on the reliability or operation of equipment. Seismic qualification of equipment was maintained. With the reactor in a defueled, cold-shutdown condition, there was no accident analysis impacted by the equipment affected by this TOI. The valve manipulations and pump operations resulting from the TOI did not have any adverse impact on any plant systems. The enhancements provided by the revision to the TOI, by filling and venting, ensure those system will function as designed. This change did not involve an unreviewed safety question.

8. Replacement of Condensate Storage Tank (CST) Level Instrumentation for Indian Point Unit No. 1 (IP1), MFI-88-01885-M

   The original Unit 1 CST level instrument which was obsolete was replaced with a different manufacturer. Recorders in the CCR and locally were replaced. The new devices weigh less than half the weight of the original ones and the power consumption is much less due to the improved design. The instrument power is taken from the same IP1 source as the original. The IP1 CST's are not required for either safe operation or shut down of Indian Point Unit No. 2 (IP2). These tanks are used only to provide extra storing capacity of treated water/condensate. Failure of the IP1 CST level instrumentation will not cause any equipment or system malfunction associated with IP2 nor to the environment. These tanks are not required by the Technical Specifications for Unit 1 or 2 nor their bases. This change did not involve an unreviewed safety question.

9. Modification of the Condensate Storage Tank (CST), SE-92-210-MD

This change involved the removal of the CST bladder and the sealing of the tank. The sealing of the tank was accomplished by replacing the tank vent with redundant breather valves which are designed and tested to preclude explosion and implosion of the CST. The nitrogen supply rate capacity was increased to allow it to fully compensate for any changes in tank inventory. Level instrumentation was added to avoid the potential of overflowing the tank. The purpose of this modification was to keep the CST water from becoming entrained with oxygen. The modification also eliminates the requirement to periodically inspect the bladder and eliminates the possibility of the auxiliary feedwater pump suction piping from being blocked by a torn bladder. The breather valves were seismically designed and mounted and were fully tested. All safety related components were designed and supported to function through a seismic event. Fire protection, security, environmental qualification, seismic adequacy, and emergency planning were unaffected by this modification. This modification introduces no adverse interaction with plant operations and sufficient redundancy was provided to assure a negligible probability of malfunction of the breathing capability of the CST. This change did not involve an unreviewed safety question.

10. Hydrazine Addition to Unit 2 Condensate Storage Tank (CST) and Feed Water, SE-93-168-MD

This change involved the installation of a permanent hydrazine addition system to provide hydrazine to the condensate and feedwater systems in an effort to further reduce steam generator corrosion. This installation included a chemical feed pump with a 90 gallon storage tank, stainless steel tubing, valves, instrumentation, supports and associated cable and conduit. No new failure modes were introduced and no accident analyses in the UFSAR were affected by this installation. Failure of the system would not adversely affect the CST's safety function. The addition of hydrazine for such use is consistent with the Electric Power Research Institute's (EPRI) "PWR Secondary Water Chemistry Guidelines", Rev. 3, dated May 1993. In addition, the New York State Department of Environmental Conservation has granted approval for its use at Indian Point Unit No. 2. Security, EQ, seismic design and emergency planning are unaffected by this modification. This change did not involve an unreviewed safety question.

11. Installation of a vent in Chemical Mixing Tank No. 21, MFI-88-01482

A new vent line and a vent valve were installed in Chemical Mixing Tank No. 21 to eliminate splashing when chemicals are manually added to the tank. The purpose of the chemical mixing tank is to provide the addition and mixing of boron for the Chemical and Volume Control System (CVCS). The modification consisted of pipe, fittings, a valve and supports. The new vent was seismically designed so as not to compromise the seismic integrity of the tank. The materials used were corrosion resistant and identical to the tank material. The chemical mixing tank is not described or required by the IP2 Technical Specifications or its bases. This change did not involve an unreviewed safety question.

12. Construction of New IP2 Simulator Building, SE-89-240-MD

This change involved the installation of a prefabricated structure to house the new simulator facilities. This building is not related to plant operations but functions as a training facility outside the plant's protected area. The only interface with the plant systems is the tie-in to the high pressure fire protection water supply. Valves were installed to provide the capability to isolate postulated leaks or ruptures in the fire protection water piping in the new simulator building. Additionally, the flow requirements for the simulator building hydrant, sprinklers and hose stations do not exceed the system flow requirements for the station's high pressure fire pumps. This change did not involve an unreviewed safety question.

13. Installation of Auxiliary Boiler Feedwater Pump (ABFP) Building Temperature Control, SE-89-279-MD

This change involved the installation of intake damper motor drives and thermostats for the ABFP building exhaust fans. The thermostats and damper drives were seismically restrained to prevent damage to ABFP during a postulated seismic event. The ventilation system is not taken credit for in the ABFP high energy line break analysis or Appendix R fire protection analysis which ensures motor driven pump operability. This installation does not introduce new failure modes that have an adverse effect on plant safety. Existing wiring and cables are used and there are no changes to cable separation from this installation. This electrical power load has been accounted for in the IP2 load study. The ABFP building ventilation system is not specifically mentioned in the IP2 Technical Specifications. This change did not involve an unreviewed safety question.

14. Installation of Travelling Screens for Service Water Pumps, SE-91-274-MD Rev. 1, SE-91-284-MD

This modification involved the installation of two travelling water screens to improve debris removal capability, improve hydraulic conditions in the service water bay and provide fish loss mitigating features to the service water system. These two screens replace the single screen which had formerly been in operation. A frame was installed downstream of each travelling screen to house stilling wells and hypochlorite diffusers. A bar rack was installed upstream of each travelling screen to prevent large objects from entering the service water bay. Weather proof control panels were installed for each screen. These panels included wiring of accessories, wiring to the turbine building common local alarm panel and wiring to the CCR for category alarming. Elapse time meters were installed in the control circuit for each service water pump and 1E fuses were installed to separate these meters from the 1E circuits associated with the service water pumps. A chlorine monitor was also installed. Each travelling water screen is capable of handling the full service water flow. The screens were designed to maintain structural integrity and be capable of starting with a maximum 12 foot water level differential between the upstream and downstream sides of the screens. Required service water flow is assured without the operation of the screens since an open sluice gate from an adjoining circulating water bay would admit all of the required flow should a one foot differential develop between the service water bay and the circulating water bay. The operation of the service water screens has no effect on the accident analyses in the Updated Final Safety Analysis Report (UFSAR). This modification does not adversely impact security, emergency planning, fire protection, environmental qualification or seismic capability. This change did not involve an unreviewed safety question.

15. Replace Condenser No. 23, SE-94-054-MD

This change involved the replacement of Condenser No. 23. The scope of the replacement included the removal of the existing condenser water boxes, tube sheets, tubes and support plates and the installation of four modular titanium tube bundles with new rubber lined water boxes. No new failure modes were introduced by this replacement. The modified system operates in the same manner as the existing system. The safety related functions of the system (air ejector & radiation monitoring) were not affected. The titanium material upgrade substantially increased the equipment's corrosion protection and augmented the protection of the steam generators by decreasing the existance of trace elements of copper in the system . Additional condenser shell hold down bolts were installed since the new condenser modules were lighter than those replaced. This change did not involve an unreviewed safety question.

16. Disconnect Strainer Blowdown Valve Motor Operator for Service Water Pump (SWP) No. 24, SE-94-247-TR

This temporary change involved disconnecting of the motor operator for the SWP No. 24 strainer blowdown valve and the positioning of this valve in the open position. This change allowed SWP No. 24 to be restored to service with strainer blowdown in the continuous mode while the motor operator was undergoing maintenance. This change did not affect the safety function of the strainer blowdown system. No new failure modes resulted from this change. The removal of the motor operator on the strainer blowdown valve did not affect pump or strainer performance with the valve left in the open position. The accident response of the service water system and all other systems was unaffected by this change. The operability of the system and all of the system parameters such as pressure and flow rate are unaffected by this change since the system is normally maintained in the continuous blowdown mode. Following the completion of maintenance on the motor operator, it was reconnected and made available for service. This change did not involve an unreviewed safety question.

17. Wash Water To Service Water (SW) Inter-Tie, SE-95-122-PR

This change involved the implementation of a new temporary operating instruction (TOI) to provide directions for cross-connecting the Unit No. 1 wash water system to the Unit No. 2 service water system. This cross-connection enabled the SW system to be shutdown to allow divers to enter the Service Water Bay to perform a search and retrieval effort. In addition, as a backup cooling source, high pressure fire water was cross-connected to the portion of the service water system which supplies cooling to the emergency diesel generators (EDG's). This backup cooling source was provided to maintain a cooling source to the EDG's in the event of a loss of all offsite power, including Unit 1 power. The flow which would have been diverted if needed would not have adversely affected the supply of high pressure fire water for fire protection uses. The implementation of this TOI was limited to periods when the reactor is in a defueled, cold-shutdown condition. Provisions were taken to ensure that sufficient time was available to restore SFP heat removal to prevent the maximum operating temperature of the SFP from being exceeded. No operating limits for refueling or technical specification requirements were exceeded. All safety equipment functions were maintained. This change did not involve an unreviewed safety question.

18. Temporary Single Service Water Header Operation, SE-95-219-PR, 95-308-TM

This temporary change involved the implementation of a Temporary Operating Instruction (TOI) directing the operation of the service water system with one header while maintaining the reactor in the hot-shutdown condition and reactor coolant system temperature greater than 350 degrees Fahrenheit. This evolution was the subject of NRC approved enforcement discretion and allowed for the repair of the SWP discharge isolation valves. The evolution involved the transfer of all non-essential service water loads from the non-essential service water header to either the Unit 1 wash water supply or the essential service water header. The three non-essential service water pumps were then removed from service and the non-essential service water header drained. Upon completion of the valve repairs, the system was returned to normal and the enforcement discretion terminated. Compensatory actions were established to expeditiously return the non-essential service water header to service if necessary. Full essential service water capability continued to be maintained thoughout the duration of the evolution providing an adequate supply to all safety-related loads normally supplied by both headers. Provisions were made to ensure that an effective isolation boundary existed between the drained header and the operating header. A temporary modification of the low pressure category alarm circuitry was implemented to ensure that the low pressure in the out-of-service header did not mask a postulated low pressure condition in the in-service header. None of the transients or accidents described in the Updated Final Safety Analysis Report rely on the non-essential service water pumps when the reactor is in the hot-shutdown condition. This change did not involve an unreviewed safety question.

19. Temporary Fire Protection to the Temporary Turbine Tool Crib, SE-93-033-TM

This change involved the temporary addition of pipe to the existing high-pressure water fire protection system to provide fire water supply to the sprinkler system in a temporary trailer located on the 53 foot elevation of the Turbine Hall. The piping supplied two sprinkler heads from a hose station through a wye fitting. The ability of the high-pressure water fire protection system to supply the highest demand was not impaired by the addition of the small diameter piping. This temporary modification did not introduce any functional changes to any plant system or component. No new failure modes were introduced and no accident analyses in the UFSAR were affected. None of the equipment in the area affected by this temporary modification was safety-related or seismic nor was there any impact on any environmentally qualified equipment. This change did not involve an unreviewed safety question.

20. Removal of Thermo-Lag Barrier from Residual Heat Removal RHR Pump No. 22 Feeder, SE-93-234-MD

This change involved the removal of the Thermo-Lag 330-1 fire barrier from the conduit in RHR Pump No. 21 room that contains the power cables for RHR Pump No. 22. This Thermo-Lag had been installed to isolate the fire effects in pump No. 21 room from damaging pump No. 22 power cable. This change also included the removal of the Thermo-Lag which was installed on a portion of the conduit for pump No. 21. This Thermo-Lag had been installed to prevent damage to pump No. 22 cables due to thermal shorts. Provisions, consisting of the installation of a reel for the storage of a pre-lugged casualty cable and associated procedural guidance on the use of the cable, were implemented to ensure that one RHR pump would be available in the event of a fire in the pump No. 21 room. Therefore, the fire barrier is no longer required. Removal of the fire barrier also eliminated the ampacity derating of the power cables for both RHR pumps and thereby extends the life of the cables. This change did not increase the fire loading in the RHR pump room and did not create a hazardous condition that could cause a fire. The change did not alter the design, function, configuration or use of the RHR system. The installed equipment was seismically restrained to prevent interaction with any safety-related equipment. This change did not involve an unreviewed safety question.

21. Containment Fire Extinguishers, SE-95-174-PR

This change involved the removal of fire extinguishers from containment during plant heatup and plant operation. This change was made because the fire extinguishers are not qualified for accident conditions or seismic events and are only required to be permanently installed during refueling and maintenance outages. Administrative controls were instituted to ensure that the fire extinguishers are reinstalled in containment during a plant cooldown and subsequently removed during a plant heatup. Removal of the fire extinguishers did not adversely impact their ability to function, did not contribute to the occurrence of a fire, did not adversely impact safety equipment, did not contribute to the severity of a postulated fire and did not degrade fire suppression capability in containment. For any maintenance work that would take place when the reactor coolant system average temperature is above 350 degrees Fahrenheit which involves cutting, grinding, welding or open flame, existing administrative controls require a dedicated fire watch with fire extinguishers be provided. If a postulated fire could not be extinguished by the work crew, then the fire brigade would respond with extinguishers obtained from outside containment. Removal of the fire extinguishers during plant heatup above a reactor coolant temperature of 350 degrees Fahrenheit will eliminate the potential for an extinguisher to damage safety related equipment during accident conditions or a seismic event. This change did not involve an unreviewed safety question.

22. Retirement of Equipment Associated with Boron Monitor Tanks, SE-93-200-MD

This change disconnected and retired in-place level transmitters, alarms, pumps, heaters and piping associated with the Boron Monitor Tanks. These tanks were part of the boron recycle system which is no longer used. The purpose of this modification was to remove the low-level alarm and to eliminate the need for calibration of the level instrument. This change was completed following approval of a Technical Specification change to remove the requirement to maintain and calibrate the level instrument associated with the Boron Monitor Tanks. The permanent isolation of these empty tanks eliminated the possibility of release or leakage from or to the tanks. The change did not result in any adverse effect on formerly interfacing systems and the seismic adequacy of the remaining safety-related systems was maintained. This change did not involve an unreviewed safety question, however an NRC approved Technical Specification change was obtained.

23. Installation of Source Range Detector High Voltage Cutout Switch, SE-93-331-TM, 94-085-MM

This change involved the installation of two switches to enable the source range detector high voltage to be administratively controlled. These switches were seismically installed. The source range detectors provide three primary functions: providing input signals to low power trip protection for power excursions in the source range; providing indication of core power for source range operations (including approach to criticality); and providing monitoring capability of core conditions during core alterations and refueling. During power operations, the source range detectors are normally de-energized. Inadvertent energization of the high voltage circuit to the source range detectors could cause detector damage at full power. This change allows the detector to be readily available on a unit trip while ensuring that it is not inadvertently energized and consequently damaged. The use of these switches is procedurally controlled and will be prohibited while the plant is at hot shutdown. If the source range trip were to fail or the switches inadvertently mispositioned when reactor is in the source range, a power excursion would be limited by the intermediate range high flux trip and the power range low power high flux trips. This change did not adversely effect equipment reliability but rather enhances the reliability of the source range detector. This change did not involve an unreviewed safety question.

24. Replacement of Power Range Meter, SE-94-331-MM

This change involved an upgrade of the power range drawers in the nuclear instrumentation system (NIS). This upgrade consisted of replacing the analog meters with digital meters and associated switches and wiring. The digital meters provide greater resolution for the reactor percent power and detector current. This upgrade did not affect the source or intermediate range drawers. The NIS meters perform no safety-related function but do provide the control room operators with a visual indication of reactor core performance. The structural integrity and functional operability of the digital meters, associated switches and mounting hardware were verified. The dynamic response characteristics of the NIS were not affected by this change. The functional design and operation of the NIS did not change as a result of this change. In addition, the physical independence and separation criteria for the NIS was maintained. The common mode failures and system malfunctions associated with analog-to-digital equipment replacement were evaluated and addressed. The equipment associated with the digital meter upgrade was qualified for its environment. This change did not involve an unreviewed safety question.

25. Defeat of IR Trips Due to Instrument Drift, SE-95-006-PR

This procedure change involved the defeat of the Intermediate Range (IR) NIS reactor trips using the installed trip bypass switches. This procedure change was written to prevent the automatic re-instatement of the reactor trips when reactor power was less than 15 percent but still above P-10 (10 percent) until the trip bistables were verified to be reset. The purpose of this change was to prevent an inadvertent reactor trip from IR instrumentation which may have drifted with the implementation of longer fuel cycles. The change also provided procedural steps to ensure that the trips were re-instated when they were required by Technical Specifications. The IR trips which are normally automatically reinstated below P-10 serve as a backup for the low power power range reactor trips which are also automatically reinstated below P-10. The current from the IR detectors continued to be indicated to the operators for any sign of a reactivity anomaly. The accident analysis assumptions were not violated since the power range channels would continue to protect the core even with a single failure. Bypassing the IR trips did not affect any other equipment. This change did not involve an unreviewed safety question.

26. Increase in the Containment High Pressure Set Point, SE-94-061-EV

This change involved the relaxation of the Containment High Pressure Engineered Safety Feature (ESF) safety analysis limit setpoint from 7.3 psig to 10.0 psig. This change to the safety analysis limit was accomplished to accommodate an increase in the Containment High Pressure ESF Technical Specification setpoint from 2.0 psig to 5.0 psig. This evaluation included an assessment of the impacts of instrument uncertainties associated with extended fuel cycle lengths from 18 months to 24 months. The effects of this setpoint relaxation were evaluated for its impact on containment integrity and LOCA-related analyses/evaluations, including Large Break LOCA, Small Break LOCA, post-LOCA Long-Term Core Cooling, Hot Leg Switchover, and LOCA Hydraulic Forces. This evaluation determined that the peak calculated containment pressure would be less than the containment design and Integrated Leak Rate Test (ILRT) value of 47 psig. The LOCA-related analyses results were unaffected by the setpoint increase. The potential effects on other safety-related components and licensing basis analyses were also reviewed and found not to be affected by the setpoint increase. These areas included: primary component and systems licensing considerations, instrument and controls/equipment qualification considerations, radiological consequences, non-LOCA analyses, steam generator tube rupture, probabilistic risk assessment, emergency operating procedures. Finally, the effects of the setpoint increase on the diverse signal to initiate a reactor trip on Containment High Pressure for small breaks in the primary system were evaluated. This change did not involve an unreviewed safety question.

27. Cycle 12 Setpoint Phase III Part 2, SE-94-268-SP

This change involved the revision of instrumentation setpoints for an extended fuel cycle up to 30 months. These setpoints included main steam line low pressure coincident with high steam flow or low average RCS temperature safety injection, high steam line differential pressure safety injection, and the control rod protection trip. In addition, alarm windows and indicating light legends in the CCR panels were changed. The revised setpoints are within the existing or revised safety analysis limits for the respective function. Sufficient margin was provided to assure that these safety analysis limits would not be exceeded with the increased channel uncertainties which would result from the extended fuel cycle. These setpoints were on existing channel bistables and were within existing instrument spans. There was no impact on response times or on the response characteristic of the instrumentation. The existing safety functions were maintained. The seismic capability of the bistables or any of the channel's instrumentation was unaffected. There were no additional electrical loads on the batteries, emergency diesels or any other power source as a result of the change. This change did not involve an unreviewed safety question.

28. Cycle 13 Core Loading, SE-95-070-EV, SE-95-131-MD

This change involved the movement of fuel (including the VANTAGE+ assemblies) into the cycle 13 reactor core and replacement of some control rods during refueling operations. The VANTAGE+ fuel assemblies feature Zirlo fuel tubes (cladding), Zirlo thimble tubes, modified plenum springs, intermediate flow mixer grids, low pressure drop mid-grids, enriched annular axial blanket pellets keyless/cuspless top nozzle assemblies, modified debris filter bottom nozzles, repositioned fuel rods and enriched integral fuel burnable absorbers. The cycle 13 core loading configuration featured a low-low leakage pattern. During the cycle 12/13 refueling outage, 89 Optimized Fuel Assemblies (OFA's) were replaced with 68 fresh Region 15A VANTAGE+ fuel assemblies, 12 fresh Region 15B VANTAGE+ fuel assemblies, three Region 12A OFA's discharged at the end of cycle 10 of which one was reconstituted, one Region 12A OFA discharged at the end of cycle 11, four Region 12B OFA's discharged at the end of cycle 10 of which two were reconstituted and one Region 13B OFA discharged at the end of cycle 11 which was reconstituted. Handling of the fuel was done with approved procedures and within technical specification parameters. Criticality considerations were evaluated and the subcritical boron concentration determined to maintain a 5% delta k/k subcriticality during core loading so that the Chemical and Volume Control System Malfunction accident during refueling was unaffected. Replacement control rods were evaluated and determined to be equivalent mechanically and neutronically as the replaced control rods. The mechanical changes associated with the VANTAGE+ fuel assemblies had no impact on chemical, physical or mechanical properties and would not cause the core to operate in excess of pertinent design basis operating limits. No new failure modes or limiting single failures have been created with these mechanical changes. The only new material added by this change was Zirlo fuel cladding in the VANTAGE+ fuel assemblies which was shown to have the same characteristics with regard to the effect on existing installations. Dropped rod analyses were performed for cycle 13 and the departure for nucleate boiling (DNB) design basis was satisfied with and without turbine runback. The cycle 13 reload design did not result in the acceptable safety limits for any accident being exceeded. This change did not involve an unreviewed safety question.

29. Dissolved Oxygen Monitoring of the Condensate System at the Air Ejector Condenser Inter-condenser, SE-94-124-TM

This temporary change was implemented to assist in activities associated with the diagnosis of possible sources of dissolved oxygen. It consisted of the removal of a threaded cap on the end of the loop seal piping of the air ejector inter-condenser drains and the installation of a prefabricated cap fitted with stainless steel tubing. The tubing contained a check valve and a stop valve and supplied a sample pump and an oxygen meter. The temporary change was performed on only one of the three air ejector inter-condensers at a time for a short duration after which the system was restored to its original configuration. The failure mode of this change which could have resulted in the decrease in condenser vacuum was abated by the use of stainless steel and the limitation of flow diverted for monitoring purposes. In addition, if a decrease in condenser vacuum were to occur existing alarms were available to alert the CCR personnel. The weight of the equipment associated with this temporary change did not significantly add to floor/support loading. The area fire loading was not significantly increased by the temporary change. This change did not involve an unreviewed safety question.

30. Alternate Hydrogen Supply to the Volume Control Tank (VCT), SE-94-136-TM

This temporary change was implemented to provide an alternate source of hydrogen to the VCT during the repair of the hydrogen truck fill valves after which the system was restored to its original configuration. This source was required to supply the VCT for oxygen and pressure control by maintaining a hydrogen blanket in the VCT. The temporary change consisted of a hydrogen bottle with an attached pressure regulator and stainless steel piping connected to a previously capped valve. The hydrogen bottle, regulator and piping were seismically secured so as to be functional following a seismic event. The possibility of fire due to hydrogen leakage was not increased since the change consisted of fittings and materials similar to that originally in use. The installation was pressure tested to the normal operating pressure of the hydrogen system. The bottle pressure was monitored periodically. This change did not involve an unreviewed safety question.

31. Removal of the Alternate Hydrogen Truck Fill Connection, SE-94-177-MM

This change involved removal of a branch line connected to the No. 21 Hydrogen Supply Manifold fill header used as an alternate fill connection from the hydrogen truck and the installation of a pipe cap. This branch line included one inch piping, three-quarter inch tubing and two valves. The removal of the branch line facilitated the movement of decontamination equipment used to perform the Full System Decontamination (FSD). The removal of this alternate connection does not impact the design and safety related functions of the Hydrogen Supply Manifold to supply hydrogen to the VCT for scavenging oxygen and to the Hydrogen Recombiners for Post LOCA Containment hydrogen control. No new failure modes were introduced and existing protective features such as check valves were not altered or impacted. The branch line being removed was capped as close as practical to the main header and the seismic capability of the remaining pipe and pipe supports was maintained. This change did not involve an unreviewed safety question.

32. Spent Fuel Cooling Flow Test, SE-94-87-PR

This test involved the variation of spent fuel cooling water flow to determine the resulting level of turbulence in the refueling cavity when it is flooded. Spent fuel cooling water flow was varied from a minimum of 500 gpm to a maximum of 3000 gpm. Inspections were performed during the test using a video camera, a view box and a dye to determine the amount of turbulence at the discharge of the cooling water piping, the effect of valve position on vortexing at the cooling water inlet pipe and the noise level near valve 721. The flow values were all within the design operating limits of all components that the cooling water flowed through including the heat exchanger, the pumps and the piping. Precautions were taken to prevent inadvertent introduction of foreign material into the spent fuel cooling line. The use of the dye was approved by chemistry department for this type of application. Spent fuel pool cooling water temperature and flow were monitored and recorded during the test to ensure that an adequate level of cooling for the spent fuel was maintained. This test did not involve an unreviewed safety question.

33. 22 RHR Pump Water Hammer Test, SE-94-348-PR

This change involved the performance of a test on the residual heat removal (RHR) pump no. 22. During the normal quarterly surveillance test on this pump additional test instrumentation was installed on the RHR system and connected to data recording apparatus. The instrumentation included pressure transducers, check valve diagnostic equipment and accelerometers. The purpose of the test was to collect data to be utilized by analysis personnel to verify the cause of a minor water hammer event which commonly occurs when conducting the quarterly test. A containment penetration which contained spare electrical feedthroughs was utilized for the test. This containment penetration also is used for cabling for thirty-one active class 1E core exit thermocouples. The cables from the test equipment which were connected to the penetration were run in such a manner that they did not cross or lie in any cable tray to insure cable separation of Class 1E circuits was not compromised. The penetration was also electrically protected by double Class 1E fuses to preclude single failure. All cable runs were properly restrained to preclude damage to the penetration or nearby safety related equipment in a seismic event. There was no effect or change to any of the system design parameters or functions as a result of this test. The plant systems were returned to their design configuration following the test and all required design conditions and requirements were met The test instruments installed in the RHR system were qualified to the design pressure and temperature of the system. All temporary cable runs of low combustibility cable were within the same fire area and did not significantly increase the fire hazard within each zone traversed. No leakage was caused by the installation and use of the pressure transducers and the maximum leakage acceptance criteria associated with the RHR system was verified. Seismic qualification and environmental qualification of any equipment were not degraded. Electrical separation criteria were not adversely impacted. This test did not impose any increased or more severe testing requirements on any equipment. This change did not involve an unreviewed safety question.

34. Unit 1 River Water Return Header Integrity Test, SE-94-227-PR

This change involved the performance of a test on the Unit 1 river water return header to verify its integrity, including connections for the sphere foundation sump pump (SFSP) discharge. The test procedure directed the isolation of the header from its normal loads and the connection of two hoses to the header from a city water supply and from the SFSP discharge. The header, which was vented to atmosphere, was filled with city water and checked for leaks and proper drain flow. The normal loads to the river water return header are no longer in service since Unit 1 is shutdown, defueled and no longer licensed to operate. Provisions were made to discontinue the test if there were indications that the header was not allowing flow to the river or if excessive leaks occurred. The flows into the river water return header were either non-radioactive or provided with radioactive monitoring devices prior to discharge to the river water return header. The final configuration of the river water return header was connected to the SFSP and isolated from certain loads no longer in use. The SFSP was already equipped with a process radiation monitor. The river water return header does not provide a safety related function. The test did not adversely impact any system, structure or component required by technical specifications. Systems required to fulfill safety related functions on the shutdown, defueled Unit 1 were not impacted by the test. This change did not involve an unreviewed safety question.

35. Full System Decontamination, SE-92-271-MD, 93-389-PR; 93-390-PR; 93-391-PR; 93-392-PR; 93-393-PR; 93-394-PR; 93-395-PR Rev. l; 93-396-PR; 94-178-PR Rev. 2; 94-207-PR Rev. 3

This change involved the chemical decontamination of the reactor coolant system (RCS), residual heat removal (RHR) system and portions of the chemical volume control (CVCS) and primary sample systems during the 1995 refueling outage. The purpose of this full system decontamination (FSD) was to lower the radiation exposure received by nuclear plant workers in operating and maintaining Indian Point Unit No. 2. The FSD was performed without fuel in the reactor vessel, the control rod drives removed from the reactor vessel, the reactor head installed with at least one third of the studs torqued and a nitrogen bubble maintained in the pressurizer. FSD was accomplished by means of installing temporary decontamination equipment in a shielded room, the drum storage room (DSR), performing the decontamination using two different chemical solvents and removing the temporary decontamination equipment following completion of the process. The decontamination equipment was connected to the RHR system. The RHR pumps provided the necessary head to pump the primary system coolant through decontamination process system (DPS) and into the RCS. Other connections to the DPS included service air, demineralized water, liquid waste and electrical power. The decontamination process was described in a Topical Report WCAP-12932-PA, Rev. 2, March 1993 and a Qualification Program Report WCAP-12820, February 1991 which were reviewed and approved by the NRC. The requirements in these reports were incorporated into appropriate procedures. These procedures included the following:

The DPS Installation and Testing Procedure
The Post FSD DPS Disassembly and Removal Procedure
The FSD RCS Operating Procedure
The FSD RCS Temperature Control Procedure
The FSD Reactor Coolant Pump (RCP) Startup and Shutdown Procedure
The Spent Fuel Pit Cooling Procedure
The Component Cooling System Operation Procedure
The Charging, Seal Water and Letdown Control Procedure
The Pressurizer Level Control Procedure
The Pressurizer Pressure Control Procedure

In addition to these procedures, check off lists (COL's) and abnormal operating instructions were revised as appropriate. The facility alterations that were completed to enable installation and operation of the DPS included the following:

Placement of the DSR alleyway lifting rig out-of-service
Removal of the DSR ventilation duct
Relocation of DSR door security alarm
Installation of DSR floor drain plugs
Reconfiguration of the spent resin transfer valve lineup
DPS test pump power connection
Enlargement of the DSR door opening and installation of a door

The requirements of the Topical Report supplemented with plant specific measures were implemented to ensure that the functions of affected systems, structures and components were protected. Calculations were performed to quantify the potential offsite radiation exposure in the event of a postulated leak in the DPS or in the containment building with the equipment hatch open. The results of these calculations concluded that the maximum potential integrated dose commitment at the site boundary would be less than 1 mrem, which is well within the dose limits of 10CFR20. The effect of the FSD chemical solvents on instruments, pumps, valves and the RCP seals was evaluated to ensure that the changes did not affect the operation of the plant following FSD. The boron concentration was adjusted following completion of the FSD prior to the RCP's being stopped to ensure adequate mixing and prevent any dilution concerns when the fuel was loaded into the reactor vessel. To prevent the possibility of chemicals leaking from the primary side of the steam generators, the secondary side of the steam generators were pressurized with water and nitrogen to a value higher than the primary side. There was no adverse impact to any Class 1E equipment. No permanent loads were added to the Unit 2 vital or non-vital power supplies and there was no impact to the emergency diesel generators or the station batteries. There were no adverse effects on seismic qualification and the fire protection program plan requirements were met. Operators were appropriately trained on the FSD operating procedures. All temporary FSD structures and systems were designed to meet the same criteria for high winds that other similar structures on site do. The total integrated radiation dose effect to environmentally qualified equipment as a result of FSD was decreased. This change did not involve an unreviewed safety question.

36. Cable Separation, SE-89-22-GM; SE-90-365-MD Rev. 1; SE-92-131-GM

This change involved the installation of dividers/barriers, blankets, arc proof tape or transition trays to enhance cable separation and maintain required redundancy and isolation. In some instances movement of cables in localized areas was necessary to enhance separation. In other instances fuses were added to the cables. These changes were made to correct cable separation anomalies which were identified from Con Edison's Electrical Separation Program for power cables and small power, control and instrumentation cables. Barriers were used in the raceway system when the minimum separation distances could not be maintained. The primary cable separation function of these barriers is to protect safety related cables associated with one channel from damage by overheating induced by electrical faults and failures internal to the electrical circuits of a redundant system cable. No changes to equipment functions were made. Engineering evaluations determined that this change would not introduce any adverse interactions that could prevent safety related equipment from performing its required function. These evaluations included appropriate consideration of material effectiveness and compatibility, cable derating, seismic installation and support, and harsh environment conditions. This change did not involve an unreviewed safety question.

37. Replace SV Voltage Relays, SE-92-175-MD

This change involved the replacing the Westinghouse SV relays used in the 480V safety related electrical buses' undervoltage circuitry and the SV relays used in the 6.9kV electrical buses' undervoltage circuitry with Asea Brown Boveri (ABB) Type 27N high accuracy relays. According to the vendor, the ABB relays are at least as reliable as the SV relays in continuously energized applications. The relays associated with the 480V buses provide the degraded voltage signals which transfer the 480V buses to the emergency diesel generators in the event of a sustained undervoltage condition. The relays associated with the 6.9 kV buses provide the undervoltage signals which trip the reactor in the event of sustained undervoltage on the buses which provide power to the RCP's. The ABB relays use a 125VDC source for control power to the circuits containing Agastat time delay relays for the degraded voltage actuation. No new failure modes were introduced. The relays associated with the 480V buses were seismically qualified. The additional loads to the 125VDC system were small and were well within the limits of the system capacity. The ABB relays provide the same undervoltage detection function as the SV relays. The overall time delays for the degraded voltage actuation and the undervoltage reactor trip voltage setpoint remained unchanged. This change did not involve an unreviewed safety question.

38. Relay Protection Modification of 345kV Feeder, SE-92-299-MD

This change involved the replacement of the protective relay used for direct tripping of the Buchanan substation upon a unit trip at Indian Point Unit 2. A lockout relay with electrical reset was installed at the replacement protective relay. The replacement was done to increase reliability because of the sensitivity of the existing relay which can result in false trips at Buchanan substation and consequential unit trips. The source of power for the relay circuit was changed from a 125VDC source from Battery No. 21 at Indian Point Unit 2 to a DC source at Buchanan substation to minimize the voltage drop for the new reset coil of the lockout relay. This modification does not affect any safety related functions at Indian Point Unit 2. A supervisory relay which monitors the new lockout relay at Buchanan substation was installed in the Indian Point Unit 2 central control room (CCR). The addition of the supervisory relay had no impact on the seismic installation of the CCR panel. The change does not have any adverse impact on the fire protection program plan or environment qualification. This change did not involve an unreviewed safety question.

39. Upgrade of Penetration Protection for Lighting Panel Feeders, SE-92-300-MD

This change involved the modification of the circuit protection for lighting panels 215, 216 and a change to the trip setpoint of the feed to MCC-28A. Each of the lighting panel circuits was modified to add fuses to the power feeds and to replace the existing molded case circuit breaker with a currently available 1E equivalent. The long delay time setting associated with the breaker for the MCC 28A feeder circuit was reduced. These changes enhanced the protection of the containment electrical penetrations in the affected circuits. None of the loads associated with these lighting panels are safety related or required for design basis accident mitigation. The safety function of the circuit protection devices is to prevent damage to the containment electrical penetrations by interruption of a fault current. The fuses, associated additional cables and molded case circuit breakers were seismically installed. The fuses and circuit breakers were electrically coordinated appropriately. There were no functional changes made to any equipment and no adverse interactions introduced. The change does not adversely affect the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

40. EDG Control & Power Modification, SE-94-075-MD Rev. 2

This change involved enhancements of the emergency diesel generator (EDG) auxiliary power supplies, EDG output breaker controls and associate alarms and indication. Fuses were installed in EDG No. 21 and 22 auxiliary loads including: fuel oil pump, air compressor, pre-lubrication pump, jacket water heater and lubricating heater to reduce the likelihood of losing all of these loads due to a fault on any one of these loads. A similar arrangement of fuses had already been installed on EDG No. 23. The non-latching voltage monitoring relays were replaced with latching relays on all three EDG's to ensure that the EDG output breaker is not tripped due to the momentary interruption of 125VDC control power during a transfer of the 125 VDC source. These changes were implemented to increase the reliability of the EDG's. No additional electrical loads were added to the EDG's, the station batteries or to any Unit 2 bus. The replacement relays and the fuses were seismically installed. No new failure modes were created. Separation criteria were maintained for all new cable added. There was no impact on the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

41. Fuse Breaker Coordination, SE-94-078-MM

This change involved replacement and installation of fuses and overload coils to improve coordination of AC and DC small power and control circuits for non-safety motor control center (MCC) loads. Also involved were changes to the amptector settings of the control rod power supply feeder breakers to enhance coordination. The improved coordination provided for an enhanced margin between the load, load supply and upstream panel or bus supply protective device characteristics. There were no additional loads added to the 480V buses, MCC's, the instrument buses or the 125 VDC system. Existing seismic capabilities were maintained with the new protective devices. There was no impact on the fire protection program plan. No new failure modes were introduced. Electrical and mechanical redundancy, isolation and separation were not impacted by this change. This change did not involve an unreviewed safety question.

42. EDG Jacket Water Pressure Switches, SE-94-139-MD

This change involved the replacement of the EDG jacket water pressure switches and lubricating oil temperature switches, the installation of valves, test connections and tubing for the jacket water pressure switches and the installation of surge suppression diodes across contactor coils for each of the three EDGs. The purpose of this change was to improve the reliability of the EDGs since the jacket water pressure switches are used to govern the EDG generators' field timing. A low temperature alarm was also provided to alert the operators before the temperature of the lubricating oil drops below an unacceptable level. The setpoint of the jacket water pressure switches was changed to a higher value to assure that the EDG generator field current is secured when the EDGs stop. This reduces the potential for extended periods of field current resulting in the actuation of the EDG control fuses rendering the EDGs inoperable. No new failure modes were introduced by this change and no new electrical loads were added. All equipment replaced or installed was seismically supported There was no impact on the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

43. VC Electrical Penetration Change, SE-94-212-MD Rev. 2

This change involved the replacement of feedthroughs of containment electrical penetration H40 with feedthroughs which will be used to provide a continuous electrical power supply in containment. Each replacement feedthrough was installed as Class 1E to maintain the integrity of the containment although their application for the supply of maintenance power is not safety related. Fusible disconnects and a second set of fuses in series were installed to ensure isolation of any postulated electrical fault and thus allow the use of the penetration above cold shutdown condition on an as needed basis. There was no change in the types of materials used. The cables were qualified to fire retardancy requirements and all other materials were non-PVC and non-aluminum. The installation of associated equipment was seismically qualified. No additional loads are added to any electrical bus, the EDG's or the station batteries. There was no adverse impact on the fire protection program plan or environmental qualification (EQ). The one fire barrier which was penetrated was resealed afterward with a fire watch posted in the interim. Containment electrical penetration H40 has been removed from the EQ master list. This change did not involve an unreviewed safety question.

44. Replacement of Emergency Lighting Battery Charger, SE-94-255-DE

This change involved the replacement of the battery chargers for the plant emergency lights which are required to operate during a loss of AC power at Indian Point Unit 2 to illuminate areas needed for the operation of safe shutdown equipment and in access and egress routes. The function of the emergency lighting batteries was not altered. The input and output power requirements remained the same. The replacement charger control circuit was designed with solid state components which minimize charger weight and prolong its life. The replacement charger was equipped with an Light Emitting Diode (LED) display for charging/battery indication. This change did not adversely affect any FSAR transient or accident. The seismic restraint of equipment, where required, was maintained. This change did not impact the ability of the equipment to meet 10CFR50, Appendix R requirements. No new failure modes were introduced. All electrical characteristics were unchanged. The electrical loading requirements for the replacement charger was not changed and there was no additional load added to any bus. There were no impacts on the fire protection program plan and no EQ equipment was affected. This change did not involve an unreviewed safety question.

45. Temporary Procedure Change (TPC) to Alarm Response Procedure (ARP) for Static Inverter No. 24, SE-94-298-PR

This temporary change involved a TPC to the Alarm Response Procedure (ARP) in an effort to reduce the effects of a loss of instrument bus no. 24 which could happen with Static Inverter No. 24 powered from its alternate source for a prolonged period. Under these conditions the procedure change provided for the defeat of turbine supervisory instrumentation (TSI); the removal of the loop no. 4 average RCS temperature input to the controlling function of pressurizer level, rod control and steam dumps; the removal of the loop no. 4 differential hot leg/cold leg temperature input to the controlling function of the rod control system; the selection of loop nos. 1 and 2 as the input to the controlling function of pressurizer pressure and the block of dropped rod protection relays for nuclear instrumentation channel (NIS) N-44. The dropped rod protection relays from the NIS channels are used to initiate a main turbine runback when the NIS system detects an indication of a dropped rod. The turbine runback on a dropped rod was determined to be not required for cycle 12. The use of the defeat switches and blocking devices described in this TPC were already included in individual abnormal operating instructions (AOI's) for the affected instrumentation. The actions in this TPC only affected controlling functions not protection functions with the exception of the dropped rod turbine runback which was determined not to be required. This change did not add any electrical load and did not adversely affect seismic qualification, the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

46. Main Boiler Feedwater Pump (MBFP) Runback Defeat Switch, SE-94-306-TM

This temporary change involved placing the MBFP turbine runback defeat switch in the defeat position during trouble shooting of the MBFP turbine supervisory instrumentation. Normally upon a loss or trip of a MBFP, sensed by the TSI as a drop in turbine speed, the MBFP turbine runback circuit would actuate to reduce steam demand and runback the main turbine to about 72% power. This temporary change was implemented to avoid an inadvertent unit runback and an unnecessary plant transient during the trouble shooting activities. Upon completion of the trouble shooting activities, the defeat switch was placed in its normal position. In the event that a MBFP trip had occurred and the main turbine did not automatically or manually runback , the unit would trip on safety related reactor trip signals. No new failure modes were introduced. The CCR operator training included instructions to take manual action to runback the main turbine in the event of a loss or trip of a MBFP even in anticipation of the automatic runback. There was no impact to any separation, isolation or redundancy criteria. No electrical loads were added. No protection circuits described in the FSAR were affected. This change did not impact seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

47. Containment Fan Cooler Unit Damper Indication Power Supply, SE-94-330-MD

This change provided power that will not be stripped following a safety injection (SI) for the fan cooler unit (FCU) damper position indication. The power supply for half of the damper solenoids and all of the indicating lights was changed to 120 VAC Distribution Panel No. 1 which is supplied by 480 V MCC-26AA and is not stripped on an SI. This change included the installation of cable, conduit, junction boxes, fuses, blocks and a terminal block. The redundancy of the SI contacts and solenoids was not affected. Although the change resulted in one train of the solenoids continuing to have control power, instead of being stripped as it was previously following an SI, the SI signal continued to be redundant to both trains of solenoids, and either train continued to be available to cause the dampers to go to their incident mode positions. The additional load on EDG No. 21 and associated bus loading inventory was determined to be acceptable. Separation criteria were unaffected. Double isolation protection was provided for each containment penetration. Seismic qualification was maintained. Since the control and SI signals would still de-energize the FCU solenoids, possibility of a hot short is precluded. This change did not involve an unreviewed safety question.

48. Channelization of Containment Spray Pumps, SE-95-144-MD

This change involved the channelization of containment spray pump actuation and prevention of cross train inhibition of containment spray actuation. The change consisted of the lifting and taping of four wires from two relays. This change eliminated the possibility of both spray pumps failing to start due to the failure of one logic train. Such a failure will now only impact the spray pump in its own train. The sequencing of the CSP's during a safety injection was not affected. No new failure modes were introduced; one failure mode was eliminated. The disconnected wires were taped; no splices were made with this tape. The electrical tape used was compatible with the existing installation. The wires were secured so they would not interfere with any other class 1E equipment. This change did not impact the electrical system capacity, output or voltage and did not add any electrical loads. There were no impacts on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

49. Temporary Power Supply Radiation Monitors, SE-95-185-TM

This change involved the temporary routing of electrical power from motor control center (MCC) no. 27 to four radiation monitors (containment and vent stack, gas and particulate monitors.) The provision of an alternate power supply allowed the radiation monitors to continue to be operable while their normal power supply, MCC 26, was taken out of service for maintenance. This temporary change was only applied during the cold shutdown conditions when the loading of MCC 27 was at a minimum and was restored to its original configuration following the maintenance on MCC 26 prior to moving the operating condition of the plant above cold shutdown. Bus loading was not challenged. Cable separation was maintained. Seismic qualification of associated supports was not challenged. The change did not impact the fire protection program plan or environmental qualification. All functions provided by the radiation monitors continued to be available. The cable used to connect the electrical power was compatible with the existing installation. This change did not involve an unreviewed safety question.

50. Installation of High Pressure Steam Dump Quick Close Relays, SE-95-198-TM

This change involved the installation of quick closure relays in the high pressure (HP) steam dump control system. These relays were designed to act to limit the potential for an excessive plant cooldown event following automatic steam dump actuation by rapidly closing the dump valves if the average reactor coolant system temperature (Tavg) decreases below a predetermined setpoint. The change consisted of installing a relay actuated from a spare temperature controller bistable, installing an additional auxiliary relay to enable operation of the HP steam dump system in the pressure mode at temperatures below the Tavg setpoint and installing associated wiring. No accidents described in the Final Safety Analysis Report assume operation of the HP steam dumps. No new failure modes were introduced. The probability of receiving a safety injection signal following a reactor trip/turbine trip is reduced. All materials used were similar to and compatible with existing wiring and relays. There was no impact on class 1E equipment. Electrical isolation and separation was unaffected. There was no increase in electrical loading. The change did not impact seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

51. Turbine Control System Test Jumper, SE-95-207-TM

This temporary change involved an adjustment to a dump valve jack screw on a control valve in the turbine control oil system. This adjustment allowed the performance of troubleshooting of a malfunction in the oil system. This malfunction resulted in the inability of the unit to maintain proper control oil pressure, limiting the unit load to approximately 60% power. Although the closure time of the turbine control valve was increased as a result of this change, there was no impact on the closure time of the redundant stop valve. In addition, the independent electrical overspeed protection for the turbine was available. This change did not adversely affect the seismic response capability of the equipment impacted. The fire protection program plan was not affected. The temporary change was removed following correction of the control oil system malfunction. This change did not involve an unreviewed safety question.

52. Emergency Diesel Generator (EDG) Engine Auxiliary Control Panel Alternate Feed, SE-95-211-TM

This change involved the temporary routing of electrical power from Bus 5A via motor control center (MCC) no. 26A to the EDG engine auxiliary control panel. This panel controls equipment which provides fuel to the EDG's including: the fuel oil shutoff valves, day tank level switches and the fuel oil storage tank (FOST) level switches. The provision of an alternate power supply allowed the EDG's to continue to be operable while their normal power supply, Bus 6A, was taken out of service for maintenance. This temporary change was only applied during the cold shutdown conditions when the loading of Bus 5A was at a minimum and was restored to its originally configuration following the maintenance on Bus 6A prior to moving the operating condition of the plant above cold shutdown. Bus loading was not challenged. Cable separation was maintained. Seismic qualification of associated supports was not challenged. The change did not impact the fire protection program plan or environmental qualification. The cable used to connect the electrical power was compatible with the existing installation. This change did not involve an unreviewed safety question.

53. Replacement of Temperature Switches, SE-93-169-MD Rev. 1

This change involved the replacement of temperature switches in the auxiliary boiler feedwater pump (ABFP) room. These temperature switches are required for automatic closure of two steam isolation valves in series on high ABFP room temperature in the event of a high energy steam line break to protect the safety related equipment in the ABFP room by mitigating the postulated line break. The switches were replaced to improve their capability of being maintained and calibrated. The power supplies for the two solenoid operated valves (SOV's) were also changed to separate the environmentally qualified SOV's from the non-environmentally qualifed SOV's. No functional changes were introduced to the temperature switches. The response time of the replacement switches continued to be sufficient to protect the safety related equipment in the room. The replacement switches were environmentally and seismically qualified and are at least as reliable as the previous switches. The switch setting was not changed. The installation maintained mechanical and electrical redundancy, isolation and separation criteria. No additional loads were added. EDG and bus loading was not challenged. Fire protection, security, and emergency planning were not adversely affected. This change did not involve an unreviewed safety question.

54. Local Control of Letdown Isolation Valve, SE-94-011-MD

This change involved the installation of a local control switch in the primary auxiliary building (PAB) to provided local control of the letdown inlet isolation valve to the regenerative heat exchanger. This switch would defeat control of the valve from the central control room (CCR) in most circumstances and was intended to be used primarily when the CCR is inaccessible. To prevent the mispositioning of the switch, a protective cover over the switch was provided. A category alarm in the CCR was also provided to alert the operators when the local switch is placed in a position other than "remote". The circuit routing provided by this modification has been determined not to impact the existing associated circuits analysis for letdown in that the previously evaluated failures remain the same. This control circuit is not required for safe shutdown of the reactor nor is it required to prevent or mitigate the consequences of an accident. Isolation devices were installed between the non-safety and the safety related circuits. Mechanical and electrical separation criteria were maintained. There were no changes to electrical loads. Seismic qualification was maintained. There were no adverse effects on the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

55. Replacement of Bearing Temperature Monitor, SE-94-043-MD

This change involved the replacement of the bearing temperature monitor alarm processor, printer and input frame located in the CCR and remounting other existing equipment on a replacement face mounting plate. This monitor continuously monitors the following components: turbine generator, main boiler feedwater pump nos. 21 and 22, circulating water pump nos. 21 through 26, heater drain pump nos. 21 and 22, condensate pump nos. 21, 22 and 23 and containment recirculation fan cooler unit (FCU) nos. 21 through 25. The bearing temperature monitor is not relied upon in accident analysis and its function would not cause a malfunction or failure of safety related equipment. No new failure modes were introduced. All equipment was seismically installed. During the installation, alternate means were provided to monitor the bearings of the FCU's. No electrical loads were added. There was no adverse effects on the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

56. Replacement of Filter Regulators, SE-94-172-GM

This change involved the replacement of filter regulators or pressure regulating valves (PRV's) used to control the pressure of the instrument air (or other fluid) being supplied to a valve or other device. In addition, supports were altered as required to meet the seismic requirements. Tubing and fittings for the filter regulator or PRV replacement were altered as necessary. Flexible hoses were installed as necessary as part of the tubing alteration. No new failure modes were introduced. The system function in which the replacement or new filter regulator or PRV was installed was not affected. Component material differences were evaluated and justified to ensure no adverse effects on the affected system. Redundancy, isolation and separation were not affected. There was no adverse effect on the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

57. Instrument Air System Isolation Valves, SE-90-401-GM Rev. 2

This change involved the replacement of existing isolation valves and the installation of new isolation valves to segregate sections of the instrument air system. Drain valves were installed to provide for condensate removal. Fittings and quick-connects were installed to facilitate testing. This change also included the replacement of copper piping sized up to and including two inch diameter with stainless steel pipe. Seismic supports for piping and valves were installed as necessary. These changes did not affect the pressure and flow characteristics of the instrument air system. No new failure modes were introduced. All existing material and piping specifications were met. Materials used in the modification were compatible with the existing system. Double barriers for drain and test connections were maintained. Performance characteristics were maintained or enhanced. The addition of the test fittings provide better assurance of proper system performance while the addition of low point drains improve air quality. The addition of isolation valves improve system flexibility and plant operation in isolating zones for maintenance purposes reducing the number of valves that become inoperable. Environmental qualification was not affected. Evaluations of impact on the fire protection program plan were performed, as required, to ensure the impact was acceptable. This change did not involve an unreviewed safety question.

58. Removal of Flow Element on FCU Discharge Line, SE-93-361-MD

This change involved the removal of flow elements which had been previously placed out of service and the installation of caps on the flow element pipe connections. These flow elements were located in the piping penetration on the service water header lines discharging from the fan cooler units (FCU). Associated cables, conduits and valves were also removed. Removal of these components eliminated a potential leak path from the service water system into containment. No new failure modes were introduced. The material for the caps was stainless steel and was compatible with the service water system. No electrical loads were added. This change did not affect environmental qualification, seismic qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

59. Replacement of Angle Gauge Valves with Vogt Globe Valves, SE-94-115-DE

This change involved the replacement of Angle Gauge valves with Vogt valves. These new valves have a design which incorporates a bellows seal gland for more effective service under vacuum conditions. These valves were located on sight glasses on the low pressure feedwater heaters, the main boiler feedwater pump (MBFP) drip tank and the MBFP seal water drain tank. The purpose of these valves is to enable the isolation of the sight glasses to facilitate maintenance. No new failure modes were introduced. This change did not involve safety related equipment and was not located near safety related equipment. The valves do not play a role in accident mitigation. This change did not impact any system's design pressures, temperatures, flow rates or other parameters. The material of the new valves was compatible with the existing installation. No additional electrical loads were added. No cables were re-routed. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

60. Steam Line Drain Valve & Pipe Replacement, SE-95-159-DE

This change involved the replacement of drain valves on the steam lines supplying high pressure turbine cylinder heating and sealing steam. All parameters associated with the valves were identical with the exception of weight, pressure rating, size body material and valve type. None of these differences resulted in an adverse impact on the system or the valve function. The replacement drain valves were high pressure drop angle valves. The functions of these valves include acting as a pressure reducing orifices and providing fast, precise and repeatable flow operation for moisture removal. This change also included replacement of piping and fittings with that of a more erosion/corrosion resistant material. The cylinder heating and sealing steam functions are not safety related, relied upon for mitigating design basis accidents or required to be available during or following a seismic event. No new failure modes were introduced. The materials of the replacement parts were compatible with the existing installation. No new electrical loads were added. There were no adverse impacts on the fire protection program plan or environmental qualification. This change did not involve an unreviewed safety question.

61. Generic Piping Alteration, SE-92-10-GM Rev. 1

This change involved the generic replacement of piping or the modification of piping physical arrangement without changing any existing flow paths or sequences. This replacement or modification of piping included vents, drains and test points including manual vent, drain and/or test valves. Also involved in this change was the removal of piping which was no longer in service (retired-in-place). This change authorizes replacing piping for improved operations and/or maintenance but does not impact existing system performance or design criteria. The systems in which this generic alteration was implemented included: service water, waste gas, steam generator blowdown, primary drains, hydrogen gas, feedwater, primary make-up, and primary sampling. Seismic qualification was maintained as necessary. No new failure modes were introduced. Manual vents or drains were provided with double barrier isolation. Alterations affecting the fire protection program plan were evaluated to ensure no adverse impact. There was no impact on bus loading, cable routing, or environmental qualification. This change did not involve an unreviewed safety question.

62. Removal of Unit 1 Circulating Water Pumps (CWP's) and Motors Gantry Crane, SE-93-408-MD

This change involved the removal of the Unit 1 gantry crane and circulating water pumps from the Unit 1 dock area. This equipment was not safety-related and did not functionally interface with any safety related equipment. However, it was located in an area which contained safety related equipment. Therefore, to eliminate the potential for accidental interactions with this safety related equipment, all heavy lifts were made in a direction opposite that of the safety related equipment. In addition, to preclude an interaction with the Unit 1 river water pumps creating a Unit 2 trip potential, the loads normally supplied by the river water pumps were transferred to the service water pumps. The capacity and number of redundant slings used for each lift were sufficient to make the possibility of an inadvertent heavy load drop small. The crane and barge had been inspected and accepted by the U.S. Coast Guard. The barge assembly and crane were physically tied to the Unit 1 dock to eliminate movement and possible damage to the Unit 1 or Unit 2 intake structure or blockage of flow to the Unit 2 circulating water pumps or service water pumps. The barge was manned continuously and direct communication was maintained between the hoisting operator and the ground rigger during lifts. Lifts were not made during severe weather involving high winds. There was no impact on seismic qualification, electrical loading, environmental qualification, emergency planning or the fire protection program plan. This change did not involve an unreviewed safety question.

63. Resin Hopper Additional Valve, SE-94-173-MM

This change involved the installation of an additional valve on the Unit 2 resin fill hopper and an associated piping modification. The new valve was a diaphragm valve. This valve was installed to reduce the need for plant personnel to go to the ion exchange gallery each time new resin is added, thereby reducing radiation exposure. Failure or malfunction of this valve would not adversely affect any safety related equipment. The materials used were compatible with the existing system. This change did not affect any accident analysis. No additional electrical loads were added. The resin fill hopper is not required to be functional following a seismic event. The installation of a new valve did not require any additional restraints. There was no impact on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

64. Unit 1 Curtain Drain Collection, SE-94-204-PR Rev. 1

This change involved the plugging of the Unit 1 curtain drain discharge line and the arrangement of piping, sump pump, drum and hose to facilitate the collection and processing of the curtain drain system liquid. The normal function of the curtain drain is to accumulate and discharge ground water. The radioactive concentration of this liquid is low enough to permit the continued direct release. However, for added assurance the liquid was re-routed by this change to the liquid waste system for processing. The sump pumps and the flow path were verified prior to implementation of the change. Curtain drain discharge line plug leakage would be detected by periodic monitoring of the drain header downstream of the plug. There are no safety related functions affected by this change. Unit 1 is a non-operating, defueled facility and, as such, the majority of the previously evaluated accidents for this facility are no longer applicable. Furthermore, this change does not affect any accident initiators for any previously evaluated accidents. There were no additional electrical loads added to any vital buses. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

65. Onsite Storage of Dry Active Waste, SE-94-313-PR

This change involved the development and implementation of a procedure to provide for the safe storage of low level radioactive waste at Indian Point Station. This procedure provides for the placement, inventory and inspection of the waste during the storage period. There were no physical changes required to any system. The storage location, inside the Unit 1 containment, was originally designed to provide for maintenance of and biological shielding and enclosure of the Unit 1 reactor coolant system. The minimal increase in the fire loading was compensated through the fire protection program plan by the installed fire protection system as well as the waste form and storage containers themselves. The increase in site boundary dose as a result of this storage was calculated to be insignificant. While the facility ventilation exhaust is monitored for radioactivity, the potential for radioactivity would be low since the waste would be packaged in a form suitable for shipment prior to storage and all handlers were trained. This storage did not utilize any Unit 2 support systems and did not affect Unit 2 operations. The storage of dry active waste (DAW) is passive and has no functional interface with equipment needed for safe shutdown or accident mitigation. Floor loading requirements were adequately evaluated and addressed. This change did not involve an unreviewed safety question.

66. Unit 1 Spent Fuel Pool (SFP) Heat Removal, SE-94-322-PR

This change involved the temporary installation of a submersible pump, a portable, plate-type, heat exchanger with valves and hoses to remove heat from the Unit 1 spent fuel pool. The cooling media for the heat exchanger was water from the low pressure fire protection system. Because the amount of decay heat from the Unit 1 SFP is minimal, the normal means for removing heat is by conduction and evaporation. However, in order to calculate a more precise value of SFP inventory loss due to potential SFP leakage, it was desirable to eliminate inventory loss due to evaporation. Following the inventory calculation, the temporary installation was removed and the SFP restored to its original cooling mode. The potential failure modes were evaluated and addressed. The pump suction was physically restrained to ensure the pool was not pumped to less than 47 feet. The pump was placed in the East Fuel Storage Pool not the West Fuel Storage Pool where the fuel is stored to ensure that no fuel was uncovered. The SFP design ensures that fuel would remain covered even if the East Fuel Storage Pool were pumped dry, which would be known through level alarms and radiation monitors. Even if the Unit 1 spent fuel were uncovered, air cooling would adequately cool the fuel. Provisions were taken to prevent or mitigate any possible radioactive spillage. The provisions included a berm in place at a doorway, hoses and components hydrostatically tested to ensure integrity prior to use, leak checks during system startup, and continuous manning during pumping operations. The hose used to supply water from the fire protection system to the heat exchanger was sized such that the resulting flow from a guillotine break of the hose would not be enough to degrade the fire protection system. The pressure of the SFP water in the heat exchanger was maintained less than the pressure of the fire protection system water, so that any heat exchanger leak would be into instead of out of the SFP. The heat exchanger cooling water discharge water was continuously monitored for radioactivity and sampled once per day to ensure that there was no release of radioactivity. Dose rates in the SFP area were monitored throughout the pumping operation to ensure that any changing radiological conditions were known. Any postulated accidental radioactive release would have been bounded by the accident analysis in the final safety analysis report. There were no additional loads added to any vital buses. There was no impact on seismic qualification or environmental qualification. This change did not involve an unreviewed safety question.

67. Unit 2 New Refueling Cavity Lights, SE-94-294-MD Rev. 1

This change involved the replacement of the refueling cavity lights. Unlike the existing lights, the replacement lights are easier to repair and can be used either submerged in water or air. This replacement included the removal of the existing lights, poles, junction boxes, conduit outlet boxes and cables and the installation of nine new poles with two lights each, junction boxes, ballasts, cable and new ground fault interrupter (GFI) breakers. The cavity lights do not perform any safety related function but provide lighting for various refueling activities. No new failure modes were introduced. The new lights contain a small amount of mercury that has been determined to have little possibility of entering the reactor coolant system or emergency core cooling system. The fixtures were permanently installed. The lights and ballasts were seismically installed to prevent them from becoming a missile. Leakage was prevented by the use of approved sealing agents and methods. No additional loads were added. The new GFI breakers ensure proper isolation should a fault occur in any of the circuits. The power supply is non-class and none of the cables were run near any safety related cables. There was no impact on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

68. Temporary Operation of Control Rod Drive Fan (CRDF) No. 21 without Exhaust Damper, SE-94-334-EV

This change involved the temporary operation of CRDF No. 21 without its exhaust damper installed. The function of the damper is to keep the fan from rotating backwards when it is not operating, thereby limiting the starting current required to place the fan in operation. The damper assembly was removed in a seismically restrained manner. The fan was started utilizing a manually controlled damper which was removed after fan started. This evolution was controlled with plant procedures. The operation or failure of a CRDF, including the removal of a damper, has no effect on any accident previously evaluated in the final safety analysis report. Removing the damper from CRDF No. 21 had no effect on the CRDF while it is in service. Should this fan have been shut down or have tripped, the other fans would cool the coil stacks and continued operation would have been permitted. No electrical loads were added. There was no impact on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

69. Reactor Head Laydown Bio-Shield Wall Cutout, SE-95-065-TM

This change involved the cutting of a two foot by twenty-one inch section out of the reactor head laydown biological shield wall to facilitate access to the underside of the reactor head for reactor head penetration inspection. Whenever the reactor was located on the head stand, equivalent shielding was installed during the period when access through the wall was not required. Administrative controls were in place to maintain radiation exposure as low as reasonably achievable (ALARA). The anchorage of the wall was not adversely affected by this change. The structural seismic capability of the wall was maintained. This change did not affect previously evaluated accidents. No new failure modes were introduced. No electrical loads were added. There was no impact on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

70. Operation of Containment Polar Crane without Seismic Restraint, SE-95-095-TM

This change involved the temporary operation of the Unit 2 containment polar crane without seismic restraints. These restraints were removed to facilitate proper machining due to crane rail replacements. Following the crane rail replacement the seismic restraints were installed and the operation of the crane returned to its normal functions. The postulated failure of the crane during the refueling outage while all of the fuel is in the fuel storage building with all the isolation gates/valves closed and the reactor coolant system decontamination completed would not result in any offsite releases or any ALARA concerns. A restriction was placed in effect to keep from lifting the reactor head with the polar crane in this condition. No electrical loads were added. There was no effect on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

71. Repair of Containment 46 Foot Elevation Floor, SE-95-129-EV

This change involved the removal of the existing coating on the concrete floor and concrete piers of the containment 46 foot elevation and the reapplication of a new coating on this surface. The activity included cleaning, recoating and repairing the concrete floors. The function of the coating is to protect the concrete during a design basis accident and preserve the function of the recirculation sump pumps and containment sump pumps by preventing the potential for debris accumulation and entrapment during the design basis accident. The concrete surface preparation and protective coating application was carefully controlled to ensure that the above functions would be maintained. Personnel safety features were in place. Adequate means for removing all radioactive particulate matter created by the removal of the existing coating and concrete dust during the surface preparation was furnished. High efficiency particulate absorbing (HEPA) filtration was used to limit airborne radioactivity. Surface cleaning debris which was contaminated was placed in disposable drums for removal to a waste disposal area. The removal of a portion of the concrete floor surface did not impact the structural integrity of the containment structure. The fire loading for the new coating did not increase the fire loading inventory. No new failure modes were introduced. No electrical loads were added. There was no impact on seismic qualification or environmental qualification. Activities involved with the floor coating did not adversely impact the floor design capability and did not have any potential interaction with safety related equipment. This change did not involve an unreviewed safety question.

72. Temporary Repair of Fuel Transfer Cart Wheels, SE-95-074-TR

This change involved the temporary repair of the fuel transfer cart wheels. The repair included the re-installation of the wheel and shaft assembly including the wheel, locking pin, washers, bushing and shaft. There are normally 16 wheels on the cart. Although not all wheels are required for load bearing purposes, all wheels do help with cart alignment to the upender in the fuel storage building. The repaired assembly met the same functional capability requirements as the original assembly including allowable tolerances for other assembly parts. The replacement wheel and shaft was made out of stainless steel so that no new corrosion mechanisms were introduced. A postulated failure of the replacement assembly would not have resulted in fuel damage since the remaining seven sets of shaft/wheel assemblies provide more than adequate support for the combined car/fuel assembly even under a safe shutdown earthquake (SSE). Welding or mechanical assembly of the shaft/wheels assembly with tack welds or other positive physical means was used to ensure tightness. No new failure modes were introduced. There was no affect on electrical equipment, electrical loads, electrical isolation or separation. The restoration of the wheels did not adversely affect the seismic capability of the cart. There was no impact on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

73. Increase Maximum Concrete Temperature, SE-94-238-EV

This change involved an evaluation of an increase in the maximum allowable concrete temperature for the containment in the area of the steam and feedwater piping penetrations. The maximum allowable concrete temperature was increased from 200 degrees Fahrenheit to 250 degrees Fahrenheit. There were no physical plant changes associated with this evaluation. The containment concrete wall provides a barrier between the normal and post-accident environment and the atmosphere and provides seismic support for plant equipment and protection from external events. The increase in the maximum allowable concrete temperature would not result in an adverse impact on the design strength of the concrete structure or its radiological shielding integrity. This conclusion was justified through an engineering evaluation. No new failure modes were introduced. There was no impact on electrical loading, separation, isolation or redundancy. The fire protection program plan was unaffected and there was no impact on environmental qualification or seismic qualification. This change did not involve an unreviewed safety question.

74. Steam Generator (SG) Foreign Object Search And Retrieval (FOSAR), SE-93-109-EV Rev. 1

This change involved an assessment of the effects of operating with two objects left in the steam generators, one in each of SG nos. 23 and 24. These objects were discovered by indications on the digital metal impact monitoring system (DMIMS) following a reactor trip and plant cooldown to approximately 529 degrees Fahrenheit. The mass of each of these objects has been estimated to be between 0.25 and 0.5 pounds. The presence of the foreign objects would not affect the response of the steam generator or instrumentation lines in the steam generator or steam lines to accident conditions. The steam generators are expected to meet all previously applicable design criteria. Primary to secondary leakage through a steam generator tube is not expected as a result of the foreign objects during Cycle 13. Therefore, the presence of these objects in the steam generators are not expected to cause an operational problem for the remainder of the current operating cycle (cycle 13). Since tube bundle integrity and leak-tightness are expected to be maintained, there is no mechanism for the foreign objects to affect any other portions of the steam generators or any other components. No other system connecting with the steam generator or other safety related component could be adversely affected by the operation of the steam generator with the foreign objects present. This change did not involve an unreviewed safety question.

75. Steam Generator Wide Range Level Upgrade, SE-94-272-MD

This change involved the upgrade of the steam generator wide range level (SGWRL) instrumentation for post accident monitoring capability in accordance with Regulatory Guide (RG) No. 1.97. The SGWRL transmitters were replaced with transmitters which were environmentally qualified. Cable splices were upgraded, as necessary, to be environmentally qualified. The system was upgraded to two trains, two channels per train. The power supply for two channels of indication were powered from one instrument bus, while the other two channels were separately powered from a second instrument bus, thereby, increasing equipment redundancy and independence. Two channels of indication were located in an analog instrument control rack and were protected from non-safety related equipment by class 1E fuses. No single fuse failure would disable all four channels of indication. Each transmitter power supply was designed to include internal fuses to prevent failure of one component from propagating to other safety equipment on the instrument bus. Existing process impulse tubing was used and no other equipment within containment was affected. The additional weight of the replacement components was determined to have no adverse impact on the seismic capability of the transmitter racks, cable trays and instrument racks. The change was compatible with the design, material and construction standards applicable to the affected system and equipment. The change did not affect the overall performance of any system in that the response characteristics of the instrumentation remained the same, the systems were not operated outside their design limits and there were no changes to system interfaces. Cable separation involving associated instrumentation and cabling was improved inside containment. Affected circuit capacity, as well as emergency diesel generator and station battery loading, were determined to be adequate for any net increases in load as a result of this change. Justification was provided to show that as a result of the upgrade there are no credible single failures that would cause the loss of meaningful SGWRL indication. Other RG 1.97 variables such as SG narrow range level, SG pressure and auxiliary feedwater flow would continue to be available to aid in determining the availability of secondary heat removal. There were no adverse impacts on the fire protection program plan. This change would not adversely impact the radiological consequences of a postulated accident. This change did not involve an unreviewed safety question.

76. Installation and Use of Nozzle Dams & Rings, SE-94-323-MD Rev. 2

This change involved the temporary installation and use of nozzle dams over the hot and cold leg nozzles inside the steam generator channel heads. The change included the permanent installation of stainless steel nozzle dam retention rings in the steam generator channel heads. The rings, welded to the channel head cladding, provided a means for the temporary attachment of the nozzle dams. The nozzle dams were a "slam latch" design with spring loaded latches that close to engage the underside of the nozzle ring ledge. The rings were designed to accommodate either the slam latch design or the threaded fastener design. Procedures were in place to control the installation and removal of the nozzle dams, the establishment of the necessary ventilation path, the operation of the air pressure to the dam seals and the response to off-normal and alarm conditions. The use of the nozzle dams permitted the performance of maintenance activities in the steam generator channel heads with the water level in the reactor coolant system above the nozzles, such as during refueling operation. In addition, the nozzle dams minimized the potential for the loss of foreign objects into the reactor coolant system piping from the steam generator channel heads. The presence of the nozzle dams did not affect any assumptions previously made in accident analyses described in the final safety analysis report. The nozzle dams were shown through evaluation to maintain their integrity under normal and faulted plant conditions, plant shutdown operations, and postulated loss of RHR during shutdown conditions. The integrity of the steam generator was not affected by the installation of the ring and the effect on the flow characteristics into and out of the steam generator channel head was insignificant. The nozzle dams were only used during shutdown conditions when the function of the steam generators is not required and were removed prior to placing the unit in a condition above cold shutdown. This change did not involve an unreviewed safety question.

77. Use of Slip Ring for Steam Generator Eddy Current Inspection, SE-95-069-EV

This change involved the evaluation of a slip ring used during the performance of steam generator tube eddy current data acquisition. This slip ring, a rotating electrical contact (ROTEC), contained less than two grams of mercury, which was used as a contact seal. The ROTEC device had four environmentally sealed boundaries to prevent mercury escape and incorporated a mercury neutralization substance. This slip ring reduced the amount of noise in the data and consequently reduced the performance time and the radiation exposure. The slip ring did not perform a safety-related function and was only used during cold shutdown conditions. The quadruple seal arrangement was designed to prevent mercury escape in to the environment or containment. This change did not affect the design, operation or reliability of any safety-related equipment. Electrical and mechanical redundancy, isolation and separation were not impacted by this change. No additional loads were added to vital buses. There was no effect on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

78. Steam Generator Leak Test, SE-95-094-PR

This change involved the performance of a leak test procedure on steam generator nos. 22 and 24. This procedure provided for the pressurization of the steam secondary side of these steam generators. The main steam line from each main steam line isolation valve to the top of the respective steam generator was filled with water and approximately 340 grams of soluble fluorescein and disodium salt. The secondary sides were then pressurized to approximately 800 psig. An ultra violet light was then placed in each steam generator channel head to identify the location of any leakage. The evolution was performed at defueled, cold shutdown conditions during which the secondary side of the steam generators and the steam lines did not perform any safety related functions. The equipment was pressurized and maintained within design conditions. The filling of the steam lines with water did not have a detrimental effect on any piping or equipment, since the main steam lines were appropriately fitted with hydro pins or chain falls. Following this evolution, all equipment was restored to normal conditions prior to declaring the steam generators and steam lines operable. Since the reactor was in a defueled condition during this evolution, boron dilution was not a concern. The concentrations of fluorescein used in the secondary side of the steam generators did not cause any chemistry limits or industry guidelines to be exceeded including any residue which may have remained after draining and refilling. Furthermore, any leakage into the primary side would have been sufficiently diluted so that primary side flushing would not have been required. There was no impact on the electrical system, seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

79. Modification of Independent Electrical Overspeed Protection System (IEOPS), SE-95-002-MM

This change involved the modification of the IEOPS circuitry to eliminate a trip of the main turbine upon a IEOPS failure signal. The modification consisted of adding a short length of electrical wire across existing relay contacts in the central control room IEOPS cabinet. Prior to this change, the failure detection circuit provided for a turbine trip upon loss of input signal in a redundant two out of three logic. This feature has been eliminated. With this change, the IEOPS continues to provide for a turbine trip signal on overspeed in a redundant two out of three logic. Analysis demonstrates that IEOPS is not required to prevent a destructive turbine overspeed event. The change does not affect the turbine trip interface with the reactor protection system or other plant systems. There was no impact on any safety related equipment. No new failure modes were introduced. No additional electrical loads were added. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

80. Installation of Signal Generator on IEOPS Channel, SE-95-026-TM

This change involved the temporary installation of a signal generator (sine wave oscillator) on one IEOPs channel at a time. This signal generator was used to provide a signal to an IEOPS channel to maintain the channel in an "untripped" condition. A failure of the generator would have resulted in a trip signal from that channel. Two out of three channel tripping causes a turbine trip. Therefore the trip coincidence was changed from two out of three to two out of two for the duration of the temporary change. This change allowed continued operation of the unit with a failed IEOPS probe or channel and reduced the probability of an inadvertent turbine trip should another probe have failed. IEOPS is not required to preserve the plant design basis for turbine missile ejection probabilities from a turbine overspeed condition. No additional electrical loads were added to vital buses, emergency diesel generators or station batteries. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

81. Removal of Pressurizer Missile Shield, SE-95-034-MD, SE-95-037-EV

This change involved the removal of the missile shield plate mounted on the top of the concrete enclosure of the pressurizer. New bearing plates were installed to facilitate the safe transfer of the post tension forces of the tension rods to the enclosure concrete sections. There are no credible missiles within the containment or within the pressurizer compartment that would necessitate the protective function of a pressurizer missile shield. This change did not adversely affect the safety related functions or the structural integrity of the pressurizer and its supporting structure. Structural analysis of the pressurizer concrete enclosure, without the cover plate, indicated that its structural integrity would be maintained during and after postulated seismic events including an operating basis earthquake (OBE) or a design basis earthquake (DBE). Removal of the missile shield resulted in increased ambient air circulation and heat loss and reduced ambient temperature around the pressurizer. The pressurizer heaters were determined to be adequately sized to overcome the expected heat losses from operation with the pressurizer missile shield removed. There was no impact on environmental qualification or the fire protection program plan. There was no impact on the expected level of occupational radiation exposure for workers performing maintenance during operations. This change did not involve an unreviewed safety question.

82. Use of Freeze Seals for Repair of Components, SE-94-210-PR, 95-038-PR, 95-039-PR, 95-040-PR, 95-066-PR, 95-206-PR

This change involved the use of repair procedures which call for system isolation by means of the application of freeze sealing methods. These procedures include the performance of pre-freeze liquid penetrant examinations and pipe diameter measurements, the provision of ventilation paths, the installation of freeze collars, the addition of liquid nitrogen to the freeze collars, the performance of the repair, the post-freeze thaw and the post-freeze liquid penetrant examination and pipe diameter measurements. This technique was applied to stainless steel and carbon steel piping to enable repair/modification of the following components: charging pump no. 23 ventilation isolation valve, residual heat removal system heat exchanger no. 22 component cooling water outlet isolation valve, reactor coolant pump component cooling water supply isolation valve, reactor coolant pump thermal barrier component cooling water return isolation valve, component cooling water loop radiation monitor sample pipe and service water strainer no. 22 blowdown pipe. The potential failure modes of pipe fracture or freeze plug failure were analyzed and precluded. Compensatory measures were also implemented in the unlikely event of these failures. Personnel habitability considerations were covered through continuous oxygen monitoring during the freeze seal and maintenance activity. There was no impact on environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

83. Air Supply for Weld Channel System Pressure Regulators, SE-93-322-MM Rev. 1

This change involved the substitution of upstream weld channel air/nitrogen for instrument air as the air source for the pressure regulators. The regulators were designed to regulate as a function of their air source pressure and would previously have failed closed upon loss of instrument air. The weld channel system was intended to continue to be operable following a loss of instrument air. This change provided for the air from the weld channel air receiver or the nitrogen backup system to continue to maintain a pressurized source of air to the weld channel system pressure regulators thus allowing the system to function as designed. The seismic qualification of the system was maintained. Even though the weld channel system provides for the mitigation of offsite releases following an accident, its operation was not assumed in the calculation of 10 CFR Part 100 dose calculations. No new failure modes were introduced. The materials used for the tubing was stainless steel which is compatible with the existing installation. There was no adverse effect on the fire protection program plan. This change did not involve an unreviewed safety question.

84. Weld Channel System Upgrade, SE-94-143-MD Rev. 1

This change involved the modification, repair and inspection of the Weld Channel and Penetration Pressurization System (WC&PPS). The change included the replacement of expansion joints on the feedwater and main steam pipe sleeves, the replacement of the pressure control valves for the four main WC&PPS air pressure regulating valves and the four WC&PPS nitrogen backup pressure regulating valves, the installation of strainers upstream of the air pressure control valves, the repair of WC&PPS piping and the inspection of the WC&PPS air receivers. The addition of strainers upstream of the air pressure control valves was intended to protect the pressure control valves from premature seat wear due to line debris. The weld channel pressurizing support rack was modified to support the additional weight of the strainers. The seismic qualification of the system was not adversely affected. The materials used were compatible with the existing installation. The changes do not adversely affect the safety functions of the WC&PPS. The system is not taken credit for in the accident analyses, although it is discussed in the final safety analysis report because of its offsite dose mitigating effects and is required by the technical specifications. No additional electrical loads were added. There was no impact on seismic qualification, environmental qualification, or the fire protection program plan. This change did not involve an unreviewed safety question.

85. Pressurize WC&PPS Containment Floor Zones, SE-95-059-TM

This change involved the pressurization of WC&PPS containment floor zones to prevent intrusion of moisture and debris in to the floor zones. The change was implemented during the 1995 refueling outage when the WC&PPS was not required. The system was restored to its original configuration prior to returning the unit to above cold shutdown conditions following the refueling outage. Instrument air was temporarily connected through a check valve to WC&PPS rack no. 14 which supplies the floor zones. In addition, a nitrogen bottle with a pressure regulator was temporarily connected to rack no. 14 in parallel with the instrument air connection. The floor zones were pressurized at about 25 psig which was less than normal WC&PPS operating pressure since the system was not required for containment integrity. A relief valve was temporarily installed between the two check valves and rack no. 14 to prevent overpressurization of the floor zones in the event of a pressure regulator failure. All materials used were compatible with the existing installation. No electrical loads were added. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

86. Reactor Coolant System (RCS) Vacuum Refill, SE-95-067-PR Rev. 1

This change involved the filling of the reactor coolant system under vacuum conditions. The vacuum assisted in the removal of air and non-condensible gases from the RCS. The technique was performed first under defueled conditions and then again with fuel in the reactor core. A vacuum pump was used to transfer air from the RCS to the plant vent system and to return water condensed out of the gas mixture back to the containment sump for processing. After the gases were removed, the vacuum was maintained until the reactor coolant system was filled to approximately 50% of the pressurizer level. When the RCS fill was completed the system was pressurized and the reactor coolant pumps (RCP's) started once. This procedure reduced the time required for filling the RCS, improved RCP seal performance by limiting the amount of starts and simplified plant chemistry operations by reducing the amount of hydrazine required to scavenge oxygen in the RCS. The pressurizer, steam generators, pumps, tanks, heat exchangers and reactor vessel are adequately designed to withstand the effects of vacuum refill. The was no safety related effect on the reactor internals nor the control rod drive mechanisms (CRDM). The safety function of the reactor coolant pump seals was preserved. There were insignificant effects on RCS piping. The integrity of the reactor coolant system pressure boundary was unaffected. Transmitter calibration following the refill operations detected no sensing element damage. The integrity of the upper reactor head instrumentation components was not adversely affected. Since the vacuum refill was performed under cold shutdown conditions, there was no effect on the accident analysis. Actions were taken to ensure that RCS mid-loop operations were not impaired. No unanticipated radiological consequences were expected as a result of the vacuum refill and none were experienced. This change did not involve an unreviewed safety question.

87. Alternate RCS Level Indication, SE-95-214-TM

This change involved the temporary installation of an alternate means for monitoring RCS level while the ultrasonic level indicator was inoperable. The installation consisted of stainless steel tubing and swage lock fittings connecting a manifold containing a Heise pressure transducer and a pressure gauge to RCS loop no. 23 hot leg instrument tap through an isolation valve. A pressure transducer to monitor ambient pressure was also temporarily mounted on the same vicinity as the manifold. This temporary change was implemented while the plant was in the cold shutdown condition. The configuration met the independent requirement of NRC Generic Letter 88-17. There were no intrusive electrical or mechanical connections. The temporary equipment was removed and the system restored to its previous condition prior to RCS pressurization. The materials used were compatible with the existing installation. No electrical loads were added. There was no impact on seismic qualification, electrical qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

88. Waste Gas Decay Tanks Discharge To Plant Vent, SE-95-083-TM

This change involved the temporary means to allow release of the gas decay tank with the plant vent stack radiation monitor out of service. The change consisted of the removal of a wire from a terminal point in the local control unit for the radiation monitor. This wire removal resulted in the maintenance of a continuous air supply to the valve positioner for the waste gas decay tanks discharge valve allowing it to be opened. In accordance with technical specifications, two independent samples of the gas decay tanks were performed prior to the release. Two technically qualified members of the facility staff independently verified the release rate calculation and the discharge path line-up. Manual isolation of the release path continued to be available to the operators. This change was not impacted by nor did it impact the plant's mode of operation. There was no impact to any system's design pressures, temperatures or flow rates. No electrical loads were added. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

89. Procedure for Resetting Safety Injection, SE-94-368-PR

This change involved the means for resetting the automatic safety injection signal. The purpose of resetting the safety injection circuitry is to allow the operator to take manual control over the safeguards equipment. Previous to this change, the means for reset was to trip the containment high pressure safety injection bistables in the reactor protection racks. As a result of an event that occurred at the Salem nuclear plant, it was discovered that under certain circumstances where an automatic safety injection only partially occurs, the tripping of the bistable may subsequently cause an inadvertent, unnecessary safety injection signal along with its bus stripping feature. The new means for reset consists of defeating both trains of automatic safety injection using key switches. This removes the possibility of subsequent inadvertent, unnecessary full or partial automatic safety injection signals. The capability to manually initiate safety injection at any time was unaffected by this change. The key switches are easily accessible to the operators in the operating area of the central control room. This change did not alter the response of any equipment assumed to operate during an accident or transient described in the final safety analysis report.

No electrical loads were added. There was no effect on seismic qualification, environmental qualification or the fire protection program plan. There were no physical changes to instrument or controls as a result of this change. This change did not involve an unreviewed safety question.

90. EOP Revision, SE-95-104-PR

This change involved the revision of the emergency operating procedures (EOP's). The EOP's are a network of predefined and prioritized symptom-based response strategies which guide operator response to analyzed postulated transients and accidents for restoration of the plant. This revision included changes to the means for resetting a safety injection signal, setpoint changes and other improvements. The revision involved only procedure changes and did not impact the physical hardware of the plant. The time response of the operators during emergencies was demonstrated to be within the time limits previously assumed. The changes did not adversely affect equipment operation. The setpoint changes remained conservative with respect to accident analysis. The revision did not involve a change in the strategies and tactics used for accident mitigation. No new failure modes were introduced. No electrical loads were added. There was no impact on seismic qualification, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question.

91. Temporary Fill Connections for Feedwater Heaters, SE-95-216-TM

This change involved the temporary installation of hose from connections from the condensate storage tank (CST) to a feedwater heater drain isolation valve. This temporary installation was seperately implemented on feedwater heater no. 21B and 21C on two different occasions. This purpose of these hose installations were to fill the shell side of the feedwater heater to facilitate the identification of sustpected tube leaks. Less than one percent of the capacity of the CST was used for these evolutions. The temporary installations were only implemented when the average temperature of the reactor coolant system was below 350 degrees Fahrenheit and were removed prior to exceeding that condition. The hose was not located in any areas where safety related equipment was located. Operating equipment was not adversely affected. No electrical loads were added. There was no impact on seismic qualifications, environmental qualification or the fire protection program plan. This change did not involve an unreviewed safety question. This change did not involve an unreviewed safety question.