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10 CFR 50.59

October 12, 2001
ENGC Ltr. 2.01.095

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

Docket No. 50-293
License No. DPR-35

Subject: Report of Changes, Tests, and Experiments Performed at Pilgrim Nuclear Power Station

In accordance with 10 CFR 50.59(d)(2), Entergy Nuclear Generation Company is submitting this report of the changes, tests, and experiments evaluated in accordance with 10 CFR 50.59 at Pilgrim Nuclear Power Station for the period of June 22, 1999 through July 1, 2001.

Sincerely,

W.J. Riggs

JLR/jb

Enclosure: Report of Changes, Tests, and Experiments

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Attachment

Report of Changes, Tests, and Experiments

Acceptance of Reactor Recirculation System Jet Pump Conditions
Safety Evaluation: 3084

This change accepted conditions associated with the reactor recirculation system jet pumps. Visual inspections revealed that various jet pumps have the mixer swing gates of the restrainer bracket assembly installed without their latching pins completely engaged and do not have their restrainer bracket set screws in full contact with the mixer.

This condition did not involve an unreviewed safety question. Vibration and structural analyses were performed to determine the effect of the differences between the as-designed and as-found condition of the restrainer bracket assembly. A conservative set of design loads were developed using input from General Electric (GE) and the Final Safety Analysis Report (FSAR). Results of the vibration analysis showed that the fundamental frequency of the as-found condition was outside the resonance region and thus little amplification would result. Fatigue failure of the jet pump was also determined to be unlikely. The structural analyses showed that stresses remain well within applicable stress limits. Load combinations used in these analyses included thermal, pressure, seismic, line break and plant transient loads. Thus, stresses remain below conservative allowable limits, structural integrity for design basis conditions is assured and jet pump integrity is maintained.

Changes to Instrumentation Setpoints and Surveillance Frequencies
Safety Evaluation: 3112

Changes were made to various instruments to extend existing surveillance frequencies from three to twelve months and/or changes to setpoints were made to increase operating margin. The instrumentation affected provide the following functions:

High Pressure Coolant Injection (HPCI) High Steam Flow Isolation
Reactor Core Isolation Cooling (RCIC) High Steam Flow Isolation
RCIC Minimum Flow Bypass Valve Open/Close Interlock
HPCI Minimum Flow Bypass Valve Open/Close Interlock
Automatic Depressurization System (ADS) Low Pressure Core Standby Cooling System (CSCS) Pump Discharge Pressure Interlock

These changes did not involve an unreviewed safety question. All setpoint changes were evaluated in accordance with Regulatory Guide 1.105 and found acceptable. The analysis was based on a 95% probability that the trips would occur before the design basis analytical limit. The changes had no impact on system operation. Review of instrument failure and reliability data showed that extending the surveillance interval would have a negligible effect on instrument or system reliability and would reduce the possibility of human error during maintenance.

Temperature Switch Setpoint and Surveillance Frequency Changes
Safety Evaluation: 3136

Changes were made to various instruments to extend existing surveillance frequencies from three to twenty four months and/or changes to setpoints were made to increase operating margin. The instrumentation affected provide the following functions:

Reactor Water Cleanup (RWCU) Area Steam Leak Detection
HPCI Valve Station Steam Leak Detection
HPCI Pump Room Steam Leak Detection
HPCI Torus Compartment Steam Leak Detection
RCIC Valve Station Steam Leak Detection
RCIC Pump Room Steam Leak Detection
RCIC Torus Compartment Steam Leak Detection
CSCS Room Cooler Unit Control
Main Steam Tunnel Steam Leak Detection
Turbine Building Main Steam Leak Detection

These changes did not involve an unreviewed safety question. All setpoint changes were evaluated in accordance with Regulatory Guide 1.105 and found acceptable. The analysis was based on a 95% probability that the trips would occur before the design basis analytical limit. The changes had no impact on system operation. Review of instrument failure and reliability data showed that extending the surveillance interval would have a negligible effect on instrument or system reliability and would reduce the possibility of human error during maintenance.

Upgrade Residual Heat Removal (RHR) Total Flow and Containment Spray Flow Instrument Loops
Safety Evaluation: 3210

This change upgraded the Residual Heat Removal (RHR) Total Flow and RHR Containment Spray Flow instrument loops to improve displayed accuracy in the Main Control Room (MCR). The Containment Spray loop was also upgraded to meet Regulatory Guide (R.G.) 1.97 requirements. This was accomplished by replacing existing transmitters with Rosemount transmitters and existing MCR flow indicators were replaced with digital bar-graph meters.

This change did not involve an unreviewed safety question. The performance characteristics of each loop were improved and the loops were environmentally qualified for accident conditions. A digital indicator failure analysis was performed to ensure no new failure effects were possible. The human-machine interface was found to be satisfactory. Electrical and heat loads were reduced. All other applicable design requirements were met by this design change.

Provide Additional Guidance for Managing Emergency Diesel Generator Fuel Supply
Safety Evaluation 3254

Emergency Diesel Generator (EDG) fuel storage capacity is sufficient to support four days of continuous EDG operation at rated loads. In order to extend their capacity to the design requirement of seven days, changes to the EDG operating procedure were necessary. Two attachments were added and are used only when EDG fuel oil capacity needs to be managed during post accident conditions. The first attachment describes the steps that station personnel use to supplement the fuel supply in the EDG fuel storage tanks with fuel from the Station Blackout (SBO) Diesel storage tank. Operators transfer diesel fuel oil from the SBO diesel fuel storage tanks to the EDG fuel storage tanks by establishing a portable fuel oil transfer from one of the SBO Diesel storage tanks to one of the EDG fuel oil storage tanks using hoses and a small auxiliary pump. The second attachment describes the steps to be taken in establishing an EDG fuel management strategy based on the fuel consumption rate of each diesel. The attachment describes the actions to be taken if there is insufficient fuel oil available in the EDG fuel oil tanks. These actions include ordering fuel oil, minimizing EDG loading, shutting down one EDG, and transferring fuel oil from SBO tanks to EDG tanks.

This change required a license amendment in order for the SBO fuel supply to be credited for the seven-day fuel requirement. The fuel oil in the SBO tanks met applicable fuel oil quality specifications. Precautions contained in the procedure prevent water or debris entrainment in the fuel being transferred. The time these actions may need to be taken is four days after the accident at the earliest and thus sufficient time is available. Dose received by personnel taking these actions is well within General Design Criteria (GDC) 19 limits. Also, managing fuel oil capacity was determined to be within the capabilities of the Emergency Response Organization.

Revise FSAR Table to Reflect Emergency Diesel Generator Loading Changes
Safety Evaluation: 3262

Currently, FSAR Table 8.5-1 shows EDG loading during a Loss of Offsite Power (LOOP) transient and during a Loss of Coolant Accident (LOCA) concurrent with a LOOP for two time steps: 0 – 10 minutes and > 10 minutes. The table was revised to show three time steps: 0-10 minutes, 10 minutes – 120 minutes, and > 120 minutes. The intermediate time period is new and reflects steady state loads started during the first ten minutes plus additional loads started by operators after ten minutes. Loads listed in the table were also revised to reflect recent load flow analyses.

This change did not involve an unreviewed safety question. The maximum steady state load on the EDGs due to the worst case accident or operational transient is within the EDG's ratings and therefore components receive adequate electrical power. Also, assuming the worst case single failures, one EDG will be capable of carrying all loads necessary to achieve and maintain safe shutdown conditions. Although the EDGs will operate at slightly higher loads, they are still within manufacturer's ratings and the probability of malfunction of equipment was not increased. The higher loading results in slightly higher fuel consumption but does not affect fuel consumption calculations.

Temporary Modification to Open EDG Control Cabinet Access Doors
Safety Evaluation: 3264

This Temporary Modification (TM) secured the EDG control cabinet access doors in the open position in order to reduce the bulk air temperature inside the cabinets. Higher than normal summer temperatures could adversely affect instrumentation in the cabinets. By opening the access doors, air circulation is improved and reduces the possibility of excessive cabinet temperatures.

This TM did not involve an unreviewed safety question. Opening the access doors improved cooling of instrumentation in the cabinet and did not affect the seismic qualification of the EDG cabinets. Inadvertent exposure of the instrumentation to fire water spray was not a concern. The fire suppression system in the EDG rooms is a pre-action type. Fire detectors would need to actuate to fill the spray header with water and fusible links at the sprinkler heads would need to be actuated for the instrumentation to be exposed. This combination of failures is beyond the assumptions of the licensing basis.

Temporary Modification to Improve Performance of EDG Cooling System
Safety Evaluation: 3265

This TM was implemented to improve EDG cooling system performance by eliminating radiator bypass pathways between the radiator frame and the fan housing by installing metal air dams and by blocking open the engine-side plenum doors of each EDG enclosure.

This TM did not involve an unreviewed safety question. Sealing bypass pathways improved cooling airflow rates through the radiators. This improves engine heat removal rates and helps to maintain jacket water temperatures within design limits. Opening the plenum doors increased outside airflow rates through the engine room and radiator. This reduces room air temperature, combustion air temperature and jacket water temperature. All elements of the TM were fully compatible with the affected components, including seismic, structural material and design process requirements.

Apply Freeze Seal to Perform Repair of Reactor Core Isolation Cooling System Pump Discharge Check Valve
Safety Evaluation: 3266

The RCIC system pump discharge ties into the RWCU system. RWCU ties into the feedwater system. A freeze seal was applied to the six-inch, carbon steel pipe downstream of the RCIC tie-in point in order to perform a repair of the RCIC pump discharge check valve. The freeze seal was applied with the RCIC and RWCU systems out of service and the feedwater system operating at less than 300°F.

This activity did not involve an unreviewed safety question. The system pressure boundary was not breached by this activity; consequently, there was no possibility for loss of system inventory. The freeze seal had no adverse affect on the ability of the feedwater system to provide primary containment isolation for any design basis event. The seal was applied in accordance with plant procedures. Any exemption from procedural requirements was evaluated and found to be acceptable. A surface examination of the piping section to which the seal was applied was performed after the seal was removed. The plant configuration during this work was within Technical Specification (TS) requirements.

Repair RCIC Rump Injection Check Valve
Safety Evaluation: 3268

This modification welded a cap assembly to the hinge pin cover flange neck of the injection check valve. The cap assembly consisted of a carbon steel ring welded onto the neck. The purpose of the ring was to collect minute amounts of leakage from the hinge pin cover flange.

This change did not involve an unreviewed safety question. The welded assembly was analyzed for full system design pressure since this condition may be reached as the leakage fills the cap and pressurizes it. Welding will not cause any distortion of valve components. The welding is done to approved codes and thus probability of pressure boundary failure is not increased. The valve hinge and disk remain free to rotate so its safety functions are not adversely affected. All structural and operational limits are met. Post-work non-destructive examination was performed to ensure weld quality ensures weld quality.

Add Operator Caution to Core Spray and Low Pressure Coolant Injection Keep Fill Checks
Safety Evaluation: 3270

Core Spray (CS) and Low Pressure Coolant Injection (LPCI) procedures were revised to add a caution statement while performing system keep-fill surveillance checks. The caution directs the operator to return any open vent valves to their normal operating positions if an automatic LPCI or CS initiation signal is received while performing the keep-fill system check.

This change did not involve an unreviewed safety question. There was no change to the manipulation or sequencing of valves. Closure of the vent valves ensures restoration of the CSCS pressure boundary integrity. There are two vent valves so a single failure of one valve does not prevent closure of the other valve. An operator is dedicated to the task and is in constant attendance. All equipment is controlled and available for use by the operators. Valve manipulations are simple and the caution provides sufficient administrative controls to ensure the action is completed.

Red Line Building and Potential Impact on the 10CFR50 Appendix R Safe Shutdown Analysis
Safety Evaluation: 3271

This evaluation supported Fire Protection Engineering Evaluation (FPEE) 123 "New Red Line Building-Exposure to Process Buildings". The new Red Line Building adjoins the turbine building and the radwaste and control building. Also, it was constructed above the Appendix R duct banks. Pilgrim's license allows "changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire". FPEE 123 demonstrates that the new Red Line Building does not adversely affect the safe shutdown capability of the plant.

This FPEE was prepared in accordance with an approved engineering procedure and NRC Generic Letter 86-10 to demonstrate that the new Red Line Building does not impact the safe shutdown capability of the plant. This FPEE considered the NRC approved Exemption Request #15, the Appendix R duct banks that run under the building and the potential fire exposure that the new building presents to the adjoining process buildings. The new building is protected throughout with an automatic wet-pipe sprinkler system. The new building is constructed of non-combustible materials and has few combustible materials in the areas that expose the process buildings containing equipment used for safe shutdown. With the exception of the Switchgear Room and part of a corridor, automatic sprinklers also protect the process buildings. In addition, the fire barriers between the buildings are considered adequate. The FPEE concluded that there are no credible fire events, that could damage redundant trains of safe shutdown equipment and that the basis for the approval of the exemption request is not affected.

Compensatory Actions for Feedwater Regulating Valve Instability
Safety Evaluation: 3274

The procedural change allowed operation of the feedwater control system partially in manual control, i.e., one feedwater regulation valve placed in manual control when maneuvering reactor power.

This change did not involve an unreviewed safety question. Analyses of anticipated operational transients and accidents were reviewed for impact. Accident analyses assume feedwater is terminated immediately and this change does not impact that assumption. The probability or consequences of feedwater line breaks and control rod drop accidents were unaffected. The consequences for the limiting Minimum Critical Power Ratio (MCPR) event were unaffected. Pressurization events, loss of feedwater heating events, loss of feedwater events and control rod withdrawal error events were all reviewed and determined to be unaffected. Consequences of inadvertent opening or closing of the feedwater regulating valves are bounded by existing analyzed malfunctions. The probability of occurrence of a transient was not increased since it increases control system stability. It was concluded that the consequences or the probability of any event analyzed in the FSAR were not adversely affected.

Evaluation of the Normal and Emergency Operation of the Turbine Building Closed Cooling Water System
Safety Evaluation: 3273

The FSAR describes the Turbine Building Closed Cooling Water (TBCCW) system as using both TBCCW pumps and heat exchangers during normal station operation. It is also stated that following the loss of AC power, both TBCCW pumps will be sequentially started automatically (due to low header pressure) and loaded on their respective diesel generator. Following a LOCA with a loss of AC power, it is stated that nonessential loads will be remote manually isolated by closure of two motor operated gate valves such that only one pump is required to provide adequate system pressure and flow to the essential components only. These descriptions are being changed based on past and present plant operating practices in which only one TBCCW Pump is normally operating while the second pump is in a standby mode. Following a loss of AC power with or without a LOCA, only one TBCCW Pump will be automatically started since the valve lineup does not change and isolation of the nonessential loads is not necessary.

This change did not involve an unreviewed safety question. The original design was based on operating both TBCCW Pumps under normal conditions. The actual heat loads and flow demands are such that only one TBCCW Pump is required to provide adequate system pressure, flow, and heat removal. The margin of safety is not fundamentally changed in any way by the actual operation of the TBCCW system. It is preferable to not isolate the nonessential loads in TBCCW and this change eliminates the need for an operator action following a LOCA with loss of AC power. The purpose of this evaluation was to explicitly describe the past and present operation of the TBCCW system in the FSAR. The function, ratings, and design capabilities of the TBCCW system are not being changed.

Installation of Blocking Rectifiers on HPCI Drain Pot Valves
Safety Evaluation: 3275

The change installed a blocking rectifier across the 125-Volt coil of each of the two HPCI Drain Pot Outlet Valves. This was done to reduce the possibility of introducing back-voltage spikes into the valves' 125 V DC supply busses that could possibly lead to damage. The "B" bus supplies power to the HPCI Inverter in addition to its valve. The "A" bus, in addition to its valve, supplies power to relays that provide HPCI auto-isolation logic.

This change did not involve an unreviewed safety question. Its purpose is to reduce the likelihood of HPCI Inverter failure. The operation, failure or malfunction of the rectifiers cannot adversely impact the fail-safe design of the HPCI drain pot isolation valves. The blocking rectifiers are highly reliable solid state devices that do not increase the probability of valve failure. The operation of the HPCI System, as described in the FSAR, remains unchanged and no new failure effects are introduced.

Calibrate the Augmented Offgas Steam Flow and Pressure Instrumentation with the Plant at Power
Safety Evaluation: 3276

Non-condensable gas from the main condenser is processed through the Augmented Offgas (AOG) system. Automatic closure of AOG isolation valves occurs if steam dilution flow decreases below the low flow setpoint in order to prevent the possibility of hydrogen conflagration. In order to calibrate Low Flow Isolation and Low Flow Alarm instrumentation, they must be bypassed with the plant at power. A TP was approved to perform these calibrations. The isolation function was bypassed by moving the control switch to the Bypass position.

This change did not involve an unreviewed safety question. The AOG system is not essential for the prevention of accidents or for the mitigation of any accident. During the calibration, steam dilution flow was manually controlled. With reactor power constant, steam dilution flow would be constant. Local steam pressure indication was provided so that any flow oscillation would be indicated. If any unexpected oscillations occurred, the procedure would be terminated. These administrative controls were sufficient to ensure the AOG system would not isolate or be subject to hydrogen conflagrations.

Evaluation of Higher Core Spray Pump Runout Flow
Safety Evaluation: 3277

A new CS system design basis hydraulic analysis was created and it includes a change to the maximum runout flow for the CS pumps. The new CS pump maximum flow rate is 4950 GPM versus 4400 GPM previously used. CS pump runout flow affects the Net Positive Suction Head (NPSH) analysis and motor AC power loading. This evaluation assessed the new design basis hydraulic analysis, changes to the NPSH total available margin, and the EDG AC power loading.

The higher CS pump runout flow of 4950 GPM has been included in the Net Positive Suction Head (NPSH) Margin and emergency AC power calculations. There is no change to the design basis methodologies used to evaluate the pump runout flow at this higher value and the licensing basis for total NPSH margin calculations is not changed. The increase in brake horsepower at runout is within the EDG loading conditions. The evaluation also showed that long term operation in a cavitating mode would not occur. Increasing the CS pump runout flow to 4950 GPM increases the total strainer flow by 550 GPM and increases the head loss due to LOCA debris that is evaluated in calculations that are also being updated and were included in an NRC license amendment submittal. Updating of the debris head loss calculations required NRC approval of the debris analysis and the containment pressurization (overpressure) that may be assumed in the NPSH evaluation. The methods used to determine total available NPSH Margin are fully described in the FSAR and the changes included in this evaluation are only to accommodate the updated runout flow rate using the established methodology. Since the methodology used to calculate NPSH margin is already described in the FSAR and this evaluation is considering only a change to the Core Spray Pump runout flow, there is no unreviewed safety question involved with the updated analyses referenced by this evaluation.

Reduce the Low Temperature Design Limit for the EDG Building
Safety Evaluation: 3278

EDG room temperatures have (on occasion) fallen below the FSAR listed "Design Temperature (Winter)" value of 60°F. FSAR Table 10.9-1 provides indoor design winter temperature values for selected site buildings, listing the Diesel Building indoor temperature as 60°F. The FSAR value is based on the design value used for the heating systems to ensure that no potential for system freezing existed. Based on this value, 60°F was then used as a design input for the EDGs and other Diesel Building components.

In response to a request to evaluate the impact of low ambient temperature on the EDGs, the EDG designer indicated that while room air temperatures remain at or above 40°F, engine-starting capability would not be impacted. Therefore, the purpose of the evaluation was to evaluate reducing the design low temperature limit from 60°F to 40°F in the standby configuration and 32°F for operating conditions.

This change did not involve an unreviewed safety question. Allowing for the design low ambient temperature of the Diesel Generator building to be reduced from 60°F to 40°F(standby), and 32°F (operating), does not impact the operating characteristics or function of any components or systems installed in this building. No physical modification results from this change. A calculation verifies that while ambient room air temperature remains at or above 40°F (standby), and 32°F (operating), no impact on installed components is anticipated. Furthermore, operation of all components evaluated will not be degraded if ambient temperature were to reach 40°F (standby), and 32°F (operating).

Use of Temporary Fuel Storage Tanks during EDG Fuel Tank Cleaning
Safety Evaluation: 3281

Two temporary fuel storage tanks were used to store diesel fuel while the "A" EDG diesel fuel underground storage tank (UST) was cleaned and repaired. One of these tanks helped satisfy the licensing basis fuel requirement for one EDG that requires enough onsite fuel to run the EDG for 7 days following postulated accidents. The other tank was used to store drained fuel from the tank being cleaned. This work was performed on-line during an "A" EDG Limiting Condition for Operation (LCO).

This TM did not involve an unreviewed safety question. There are two USTs with fuel transfer capability between each tank that supply fuel oil to their respective EDG. To ensure "B" EDG operability, a seven-day, on-site supply of fuel for the "B" EDG, contained in Class I structures, must be maintained. With one of the storage tanks emptied for cleaning, an additional fuel storage tank was necessary. The only licensing basis event that required more fuel than contained in the existing "B" UST alone was the design basis LOCA. Therefore, one of the aboveground, temporary fuel storage tanks was qualified as a Class I structure consistent with the licensing requirements for a LOCA. Seismic forces bound hurricane forces. Tornado forces were not considered since concurrent LOCA and tornado is not part of the licensing basis. The tank locations were reviewed for weight overburden on underground piping/conduit, fire protection, and interactions with other safety related equipment, the locations were found acceptable.

Standing Plant Design Change for Minor Mechanical Modifications - 2000
Safety Evaluation: 3282

This Plant Design Change (PDC) is a standing document that addresses specific, limited classes of well-defined modifications that do not constitute changes in the facility as described in the FSAR. A safety evaluation is performed solely for administrative convenience. Changes are limited to physical changes that do not affect the system design basis, function or operation and in all cases comply with FSAR requirements.

Standing Plant Design Change for Minor Instrumentation and Control Modifications - 2000
Safety Evaluation: 3283

This PDC is a standing document that addresses specific, limited classes of well-defined modifications that do not constitute changes in the facility as described in the FSAR. A safety evaluation is performed solely for administrative convenience. Changes are limited to changes that do not affect the system design basis, function or operation and in all cases comply with FSAR requirements.

Standing Plant Design Change for Minor Electrical Modifications - 2000
Safety Evaluation: 3284

This PDC is a standing document that addresses specific, limited classes of well-defined modifications that do not constitute changes in the facility as described in the FSAR. A safety evaluation is performed solely for administrative convenience. Changes are limited to changes that do not affect the system design basis, function or operation and in all cases comply with FSAR requirements.

Evaluation of Revised RCIC Pump Test Acceptance Criteria Required by Revision to the Design Basis Hydraulic Analysis
Safety Evaluation: 3285

This change revised the acceptance criteria for RCIC pump testing at normal reactor pressure. The pump head requirement was increased to account for system losses indicated by the benchmarking of a recently developed hydraulic model. No changes to the FSAR were made by this safety evaluation.

The change did not involve an unreviewed safety question. The FSAR description of the test is unchanged. The procedure changes are designed to provide assurance by test of system capability to meet the design requirements. RCIC provides core cooling following transients and accidents. There are not changes to any fundamental assumptions in the design basis or to the Technical Specifications involving this change.

Revise Description of Low Pressure Service Air System in FSAR
Safety Evaluation: 3286

The FSAR description of the Low Pressure Service Air System (LPSA) was revised to indicate that the system has the potential to contain radioactivity due to the nature of its design and operation and due to its interfaces with normally radioactive systems. The change also indicates that the system should be maintained free of radioactivity to the greatest extent practicable.

This revision did not involve an unreviewed safety question. This system was intended to be a normally non-radioactive system but that due to the nature of its design and interfaces with several normally radioactive systems, a potential exists for cross-contamination. The LPSA system has no safety function and this change has no affect on system function. There is no potential for unmonitored or uncontrolled release of radioactivity to the environment due to its design and ventilation systems that collect and route potential leakage to monitored discharge points. The level of radioactivity expected to remain in the system is negligible and creates no adverse impact on area dose rates or on total effective dose to station personnel. There will be no impact on connected safety related systems.

Freeze Seal Section of the Common Equipment Drain Sump Discharge Line to Repair Leaking Valves
Safety Evaluation: 3287

Two leaking valves in the common discharge line from equipment drain sumps were repaired. In order to ensure a positive isolation, application of a freeze seal was necessary.

This repair did not involve an unreviewed safety question. The piping material was suitable for freeze seal application. Proper controls were established to ensure piping integrity and seal integrity. Provisions for pumping the equipment drain sumps through alternate pipelines were made. Contingency plans were established to mitigate the consequences of a seal failure. An assessment of the maximum water volume that could be released in the unlikely event of a pipe failure was performed. All leakage would be contained in the immediate area and routed to normal drains in the area.

Replacement of EDG Air Start Motors and Post-Work Test Procedure
Safety Evaluation: 3288

The air start motors on the EDGs were replaced with a more efficient design that allows the EDGs to meet the ten second start time requirement with more margin. The Temporary Procedure (TP)covers the post-work test of the motors.

This change did not involve an unreviewed safety question. This change was a maintenance activity and did not involve a change to EDG function or design. All functional requirements of the original air start motor were met or exceeded by the replacement motor.

Procedure Revised to Add Actions to be taken in the Event of a Resin Intrusion
Safety Evaluation: 3290

The abnormal reactor water chemistry procedure was revised to include actions to be taken by operators if there is a discernible reduction in core thermal power and/or a jet pump drive/driven flow mismatch following a resin intrusion. These actions include: reducing reactor power, verifying the jet pump drive/driven flow relationship, and if the mismatch is beyond a specified quantitative limit, then maneuvering the reactor to lower power. A resin intrusion may change the properties of primary coolant in ways that may affect the ability to properly determine MCPR. A resin intrusion may reduce the surface tension of water causing an increase in core average void content, a change in the drive/driven flow rates of the reactor vessel jet pumps and an observable reduction in reactor power.

Actions to be taken if there is a discernible reduction in reactor power are to assume the operating limit MCPR is nonconservative (the MCPR Technical Specification LCO is entered) and reactor power is reduced to < 25 %. The drive/driven flow relationship is also verified, whether or not a discernible power reduction exists, and if the mismatch is greater than 5% then the Average Power Range Monitor (APRM) flow-biased scram and rod block setpoints are considered nonconservative and the plant is maneuvered to < 25% reactor power. Once power has been reduced, then flow biased APRM scram and rod block setpoints are set down to 38.5% and 29.1% reactor power respectively, regardless of actual flow rates. Once the abnormal chemistry condition clears, the setpoints are returned to their normal configuration.

This change did not involve an unreviewed safety question. Maneuvering the plant to below 25% power decreases the void fraction sufficiently to preclude the onset of transition boiling. It also satisfies the Technical Specification Limiting Condition for Operation. A flow mismatch of 5% being considered the threshold for taking action is consistent with assumptions used in the stability analysis for total core flow uncertainty. Setting the scram and rod block setpoints at a fixed value independent from core flow is conservative and ensures boiling transition would not occur during any plant transient. The setpoint also provide equivalent stability margin.

Revise FSAR Description of Primary Containment Penetration X-46E
Safety Evaluation: 3291

The penetration had originally been connected by piping to a reference vessel inside containment and was connected to a sample point outside containment used for Integrated Leak Rate Testing (ILRT). It had been abandoned in place. Isolation of the piping outside containment had been through a manual isolation valve. The penetration was used for supplying makeup nitrogen to the safety relief valve (SRV) accumulators. Check valves and associated piping were added inside containment and the external piping was extended outside containment to two new manual isolation valves.

This change did not involve an unreviewed safety question. The system is used following a seismic event to recharge the accumulators and thereby extend their mission time to 72 hours. Operator actions can be performed with the expected plant conditions and time constraints. The piping is routed to avoid potential missile hazards, high-energy pipe whip, external hazards and heavy equipment traffic. The components are designed to accommodate seismic conditions. The valves are normally closed and are opened only for a short time to recharge the accumulators. The new equipment will not affect operation of the SRVs. The capability for recharging the accumulators increases the time available for placing shutdown cooling in service. Installation of this system does not adversely affect use of the normal nitrogen make-up to the SRV accumulators.

Install Isolation Dampers in Main Control Room Ventilation Ductwork
Safety Evaluation: 3292

The change provides Class I isolation capability of the main control room environmental control system (MCRECS) to ensure the control room envelope can be properly isolated during periods when the Control Room High Efficiency Air Filtration Subsystem (CRHEAFS) is required for radiological events. Redundant manually operated dampers were installed in the normal air conditioning ducts penetrating the control room wall.

This change did not involve an unreviewed safety question. Its purpose was to resolve a non-conformance identified in the original design of the MCRECS. During a seismic event post-LOCA, non-safety related ductwork could fail such that the ability of the MCR to be pressurized by CRHEAFS could be adversely affected. A permanent modification was made that fully restored the intended design functions of MCRECS and CRHEAFS. The associated dampers and ductwork meet Class I design requirements, permitting the control room envelope to be isolated and pressurized by CRHEAFS during and after a seismic event. The decrease in CRHEAFS airflow rate is small and has no adverse impact on the ability of CRHEAFS to maintain positive pressure in the MCR. It has no affect on MCR heat load analyses. Operator action to isolate the dampers is required and poses no undue burden on the operators. All actions can be performed within the dose limits of GDC 19.

Modify Valve Body Drains to Prevent Pressure Locking
Safety Evaluation: 3293

This design change modified the Residual Heat Removal (RHR) pump suction from the torus isolation valves by connecting each valve body drain line to an upstream RHR system test connection to address pressure locking (PL) concerns. The modification eliminated valve susceptibility to pressure locking by equalizing pressure between the bonnet cavity of these valves and its upstream piping. It also eliminated the need for routine operator action to drain the valve bonnet(s), when initially entering shutdown cooling. Also, an orifice is installed in each continuous vent line to limit flow to 10 gpm/valve in the event that excessive leakage of a valve, while closed, were to occur.

This change did not involve an unreviewed safety question. In the normal standby lineup for RHR-LPCI mode, these valves are open. These valves are expected to remain open during all safety-related RHR system modes. There valves are closed to prevent the loss of torus inventory through a leak in the RHR system and to isolate the pump suction from the torus when the RHR pump is aligned in the shutdown cooling mode in order to prevent a reactor drain-down. Under both circumstances, the proposed modification ensures that differential pressure can develop across only one disc half. The motor operator is sized to open the valve under those circumstances.

The proposed vent line(s) are designed, constructed, and tested to the same quality and levels of quality assurance as the primary containment itself. This requirement provides assurance that primary containment integrity and water storage requirements will be maintained during design basis transients, accidents and external events.

Revise Shutdown from Outside Control Room Procedure
Safety Evaluation: 3295

It was discovered that when the EDGs are operated from the Alternate Shutdown Panels (ASP) after a postulated fire in the Control Room /Cable Spreading Room (CR/CSR), an electrical fault could cause the EDGs to falsely trip on over-current. The procedure was revised to include the manual action of isolating the EDG over-current relay trip function by removing the connecting plugs while operating the EDGs from the ASPs. This action disables over-current protection and prevents the possibility of a fire induced spurious trip of the EDGs.

This change did not involve an unreviewed safety question. Pulling the connecting plugs meets the requirements of 10CFR50 Appendix R by eliminating the possibility of any one spurious signal from preventing safe shutdown. Operator actions required to accomplish this step were evaluated and the slight increase in operator response time did not affect safe shutdown capability. Bypassing over-current protection during accident conditions is a generally accepted industry practice and is consistent with Regulatory Guide 1.9. Manual loading of the EDGs provides sufficient over-current protection and other than loss of offsite power, Appendix R does not require an additional single failure to be assumed.

Remove and Replace Reactor Building Roofing Material
Safety Evaluation: 3296

Aging, wear and weather exposure has brought the original roof covering system to the end of its normal design life. This change removed the existing built-up roofing material from the reactor building and replaced it with a new membrane roofing system to prevent further deterioration and leaks.

This change did not involve an unreviewed safety question. Much of the reactor building constitutes the secondary containment boundary. Safety functions provided by the reactor building roof structure are containment of radionuclides that leak from the primary containment and release of overpressure in the reactor building due to design basis tornado and pipe break outside containment events.

The solid steel roof decking is overlapped and welded together and then welded to the roof support beams. It provides structural integrity and although it is not leak tight, it provides the primary boundary for the leak containment function. The roof covering material acts as a passive gasket in that it provides a leak tightness function, not a secondary containment structural function. In the unlikely event a defect were to develop across the entire thickness of the roofing system, the resistance to leakage flow would be great and is within the capability of the standby gas treatment system to maintain the reactor building at a negative pressure. There are no explicit loads on a roof covering system so its failure mechanisms would be gradual and age related. The new roof covering, in conjunction with the decking provides insulation and environmental protection for building components and this function is unaffected. The new roofing material is attached to the decking in a manner that does not affect its pressure relief function if reactor building pressure reaches its design pressure during design basis events.

Evaluation of Compensatory Measures to Ensure Operability of "A" EDG
Safety Evaluation: 3297

Compensatory measures were necessary to ensure operability of the "A" EDG due to inoperable supply check valves between the "A" EDG air compressor and its associated air start receivers. These check valves close against back-pressure from the air receivers and provide the safety related pressure boundary function for the air receivers. Components that are outboard of these valves back to the air compressors do not perform a safety-related function. The compensatory measure involved manually closing inboard isolation valves to provide isolation of the safety-related air receiver boundary and maintain EDG operability while the check valves were being repaired.

This change did not involve an unreviewed safety question. Closure of the isolation valves provided a seismic isolation boundary equivalent to the normally closed check valves. The decay rate of the air receiver tank was monitored to ensure the receivers were maintained above minimum requirements. Recharging of the air receiver tanks was prevented with these isolation valves closed. Therefore, monitoring of the pressure decay rate for each receiver was performed to ensure pressure was maintained above minimum system requirements. The check valves were restored and returned to service before recharging of either receiver was required.

Temporary Modification to Drywell Equipment and Floor Sump Containment Isolation Valves Control Switch
Safety Evaluation: 3299

This temporary modification (TM) was a contingency to provide an alternate method of operating the drywell equipment and floor sump primary containment isolation valves (PCIV) during maintenance to one of the valve's control switches. Each sump discharge line has two independent and redundant PCIVs. Two valves in each discharge line are normally closed and are manually opened to allow pumping the contents of the drywell equipment sumps and drywell floor sumps to the radwaste systems for treatment. One valve in each line is supplied from an "A" division power source and the other valve from a separate "B" division power source. Isolating electrical power prevents pumping from either drywell sump. This TM provided the means to jumper control switch contacts in order to allow the valves to be operated by opening and closing their respective electrical breakers without declaring the valves inoperable.

This activity did not involve an unreviewed safety question. This TM provided an alternate method of operating these valves and its use was not expected. Implementation of the TM only affects the ability to pump down the sumps. The ability to isolate primary containment was not affected. With no power supplied to the valves, they would stay closed. If open, automatic isolation of these valves could still occur because the isolation relays did not have jumpers installed. A system design requirement is to be single failure proof and it was not affected. The redundancy and independence of each PCIV in each line was maintained during each stage of the maintenance with or without the jumpers installed.

Temporary Procedure to Enhance Augmented Offgas Charcoal Bed Drying
Safety Evaluation: 3300

This TP provided the administrative controls to raise the ambient temperature of the charcoal vault room of the Augmented Offgas (AOG) system. The moisture content of the AOG charcoal beds was too high and it was affecting system performance. A drying process was implemented to reduce the charcoal bed moisture content. The drying process could be accelerated if the vault temperature was increased in combination with decreasing the relative humidity of the incoming air. The vault temperature was raised in a controlled manner from its normal operating range of 72°F to 82°F to a higher temporary band of 110°F to 120°F. The temperature increase was accomplished by slowing raising the setpoint of the room temperature controller. Initial system parameters were recorded and prior to making a setpoint adjustment, the system was allowed to reach steady state conditions prior to increasing the temperature further. System parameters were recorded and adjusted until the desired drying rate was achieved. Once the charcoal bed moisture content was achieved then the vault temperature would be returned to normal.

This procedure did not involve an unreviewed safety question. As the bed temperature increases the K-factor (measure of effectiveness of charcoal for delaying the release of noble gases) decreases but is offset by an increase in the K-factor as the moisture content of the bed decreases. It was anticipated that the noble gas release rate would peak at fifty hours and return to nominal conditions after one hundred hours. The peak release rate would be less than 1.6 times the current release rate, which is less than 1% of the current Technical Specification limit. After ten days release rates would be less than existing rates. Abort criteria were provided if release rates approached site boundary allowable limits. Existing site procedures that require an isotopic analysis of the effluent if the activity increases by more than 50 % over the previous day were kept in force. The higher moisture content of the exiting air stream has no effect on downstream piping or equipment. Industry operating experience was incorporated. The bed temperature was increased by raising the room temperature instead of using gas-fired heaters blowing directly on the vessel walls. This prevents the possibility of igniting the charcoal beds. Isolation of the AOG system is unaffected, as the process isolation instrumentation was unaffected and outside the room. All equipment in the vault room was evaluated for higher temperatures and was unaffected. An alarm would be received in the MCR if room temperature exceeded 125°F. The pressure boundary of the vessels was not adversely affected by the heating. The FSAR assumes that all radionuclides in the holdup piping and charcoal beds are instantaneously released to the environment. This assumption bounds any potential consequence that could result from this procedure.

Installation of Temporary Pressure Gauges on Reactor Building Closed Cooling Water Heat Exchanger

Safety Evaluation: 3301

This TM installed temporary pressure gauges downstream of the vent and drain valves on the bell end of the Reactor Building Closed Cooling Water (RBCCW) to monitor differential pressure across each tube pass. The safety related pressure boundary is at the valves and the gauges were installed on the downstream or non-safety related portion. This TM also installed test gauges in place of plant gauges on the Salt Service Water (SSW) inlet and outlet lines. The pressure boundary is normally the pressure gauges but was moved to the respective root valves for the duration of this TM. These gauges provide local indication only.

This change did not involve an unreviewed safety question. The gauges were installed for monitoring purposes only and the respective root/isolation valves remained closed except to obtain pressure readings. The root/drain valve configuration was controlled under Caution tags. The temporary gauges had no effect on the capability of the SSW system to provide a heat sink for the RBCCW system under transient or accident conditions. An operator was in constant attendance while the readings were taken and the root valves were closed afterwards. The additional weight of the temporary gauges on the vent and drain connections was negligible and had no effect on the existing piping/component analysis. The test gauges installed in place of the plant gauges were of approximately the same weight and did not impact any existing analysis. This TM had no effect on the safety function of either the SSW system or the RBCCW system.

Change in Radwaste Truck Lock Dewatering Process

Safety Evaluation: 3302

Spent resin and reactor water cleanup sludge is sluiced into containers in the radwaste trucklock. The containers are dewatered to allow shipments to offsite disposal sites. The water in the sluice mixture is returned to the radwaste system. The dewatering system now in use was replaced with a system employing improved hardware with remote viewing and control capabilities.

This change did not involve an unreviewed safety question. The new system requires fewer operator actions. The new system uses diverse and redundant instrumentation. Ultrasonic level instrumentation and an internal camera monitor level in the container. A vent tank with integral HEPA filter has an additional, independent level switch that alarms and isolates the system on high level. Each of these features helps to reduce the possibility of a spill. A truck lock floor plug has been drilled to ensure all spills are routed to radwaste and eliminates the need to lift a floor plug for each dewatering operation. Any airborne radionuclides are contained within the radwaste building and are monitored prior to release from an elevated release point so that consequences of a spill are unaffected. The resultant dewatered product and container meet all applicable regulations.

Temporary Modification to Drywell Equipment and Floor Sump Containment Isolation Valves

Safety Evaluation: 3303

Safety Evaluation 3299 (reference above discussion) evaluated a TM to the control switches of these valves that would be implemented only if the sumps needed to be pumped prior to completion of maintenance. This TM was an additional contingency to provide an alternate method of operating the valves during maintenance to the control switch. This change consists of two equivalent but separate pneumatic jumpers for the inboard isolation valves on each line. These jumpers bypass the respective solenoid valves and allow air to be supplied to each valve manually in the event the valves are declared inoperable due to a failure of the electrical circuit that powers both valves.

This activity did not involve an unreviewed safety question. Installation of the jumpers bypasses the automatic Group 2 Primary Containment Isolation Signal (PCIS). Therefore, an automatic safety function would be defeated and the valves are declared inoperable. Before the TM could be installed they are no longer credited as a primary containment isolation barrier. To compensate for the inoperable valves, the Technical Specifications requires one of the two valves in each line containing an inoperable PCIV be deactivated in its closed isolation position. The outboard pair of valves are closed and deactivated to meet this requirement. If the PCIVs must be opened to pump the drywell sumps, a LCO will be entered for the period of time the outboard valves are reactivated to allow them to operate. When the valves are deactivated again following pump down of the sumps, the LCO will be exited.

This TM had no effect on the Primary Containment Isolation function of the outboard valves. These valves are powered from a separate and independent source with a separate PCIS relay and remained fully operable if the TM needed to be implemented.

Temporary Bypass of the Reactor Recirculation Pump Speed Limiter
Safety Evaluation: 3304

A planned plant evolution to single recirculation loop operation necessitated feedwater flow adjustments. Feedwater flow measurement during the transition to single loop operation could become erratic and erroneously trigger the low feedwater flow trip setting, resulting in a runback of the operating recirculation pump to 26 % of rated speed. A jumper would be installed that bypasses the pump speed limiter during the time the feedwater flow measurement could be erratic.

This change did not involve an unreviewed safety question. The function of the speed limiter is equipment protection. It runs back recirculation pump speed to 26 % to prevent jet pump and recirculation pump cavitation at low core power and high recirculation flow conditions. The runback will be performed by operator action using more accurate neutron monitoring indications. The reasonableness of operator actions were reviewed against the criteria in NRC Information Notice 97-78 and found acceptable.

Temporary Modification to Allow Installation of a Temporary Piping support in Place of a RBCCW Pump
Safety Evaluation: 3305

During the period required to perform pump overhaul, a temporary support (dummy pump) was installed in place of a RBCCW pump. An active LCO was entered for the period it took to remove the RBCCW pump and install the dummy pump and then again for the period it took to reinstall the RBCCW pump. The dummy pump provided the necessary support to the mating piping to allow the loop to remain operable.

This change did not involve an unreviewed safety question. While the pump was being removed and the temporary support (dummy pump) was being installed, an LCO was required because the respective loop of the RBCCW system was be operable. While the temporary support was installed, the RBCCW system was able to function as designed and there was no adverse impact on the safety design basis for RBCCW.

The flanges on the dummy pump are only for support; they did not provide a pressure boundary. The normal isolation valves provide the pressure boundary required for maintenance. The dummy pump was designed for the same dead weight, thermal and seismic loading as an RBCCW pump and is equivalent to an RBCCW pump relative to support. The existing piping analysis for the RBCCW system is valid when the temporary support is installed. The third pump would not be available as a spare but this is in accordance with FSAR analyses.

Procedure Delineating Emergency and Transient Response Expectations for Operating Crews
Safety Evaluation: 3306

This new procedure provides licensed operators with clarifying guidance concerning management expectations for execution of Emergency Operating Procedures (EOP) and Transient Response Operating Procedures (TROP). The EOPs specify operator actions to respond in a symptomatic way to plant events by controlling plant parameters associated with reactor power, reactor pressure, reactor level, and primary and secondary containment. The TROGs specify operator actions that are event specific and provide operators with symptoms, immediate actions and follow-up mitigation strategies. The procedure contains definitions, expectations for procedure usage, administrative controls, checklists to ensure accuracy and completeness of operator actions, control bands for transients, and lessons learned regarding selective operating practices.

This procedure does not involve an unreviewed safety question. Lessons learned and operating experience are incorporated to reduce the consequences of accidents. Equipment operation is not specified that is not already included in approved operating procedures. The expectations and guidelines in this procedure do not conflict, countermand or substitute for any other plant procedure. The procedure fulfills the procedural guidance for required operator actions delineated in ANSI 58.8.

Establish Alternate Flow Path for Sampling Reactor Water if Normal Sample Path is Out of Service
Safety Evaluation: 3307

This procedure change established an alternate flow path for sampling reactor water in the event that the normal sample path is out service. The normal sample path is from the "B" Recirculation Loop Discharge line through containment isolation valves to reactor water sampling panels. The alternate sample path is from the Reactor Water Cleanup suction line through the Crack Arrest Verification (CAV) system to reactor water sampling panels.

This procedure change did not involve an unreviewed safety question. The alternate path meets the requirements for the reactor coolant boundary. Existing control logic, instrumentation and automatic isolation valves provide automatic isolation of the alternate sample path for the same reactor conditions as the normal path. The alternate path meets requirements for reactor water pressure and temperature. The alternate path provides samples that meet Technical Specification requirements for reactor water quality.

Temporary Modification to Monitor HPCI Inverter Input and Output Voltages and Currents
Safety Evaluation: 3308

This TM was installed to monitor the HPCI inverter input and output voltages and currents. Since this inverter has input over and under voltage protection that turns its output off if the DC input goes out of the acceptable range, the TM installed a recorder to monitor the DC input and AC output of the inverter. The input voltages were checked from the positive and negative leads to ground. The input DC current was measured with a clamp-on ammeter. The inverter output AC voltage from line to neutral was monitored, and the neutral to ground voltage was monitored as well. The inverter's AC output current was measured with a clamp-on ammeter.

The change did not involve an unreviewed safety question. The installation of this TM had no adverse effect on the HPCI system to perform its safety function. The electrical connections to the input and output of the inverter were made to a high impedance input recording device through isolation resistors to limit the available short circuit current in the unlikely event of a fault either together or to ground of the test leads. The resistors were procured as safety related items to ensure that the test connections were made with components of the same quality as the component being monitored. The resistors provided over-current isolation. The equipment was secured to prevent seismic interaction. The currents were measured with clamp-on ammeter probes that did not electrically connect to the circuit but were magnetically coupled to it. These clamp-on ammeters had a negligible effect on the circuit by minutely changing the inductance of the circuit.

Perform a RCIC Pump Test at Reactor Pressure < 150 PSIG at Rated Flow with Test Line Orifice Installed
Safety Evaluation: 3309

This change was a test of the RCIC system to determine if it is capable of passing the once per cycle pump test at reactor pressure <150 psig with the restricting orifice installed in the test line. Previous testing at rated flow was performed with the test line orifice removed. New information regarding the test line orifice size and RCIC turbine performance indicates that the turbine has the capacity to provide rated flow (i.e., $\geq$ 400 gpm) with the orifice installed and reactor pressure less than 150 psig. This TP was designed to test for that capability. Based on the results of this test, routine surveillance procedures were modified to perform future 150 psig testing in a similar manner.

The change did not involve an unreviewed safety question. The FSAR description of the test is unchanged. The procedure changes are designed to provide assurance by test of system capability to meet the design requirements with a new system lineup, e.g. orifice installed. During the conduct of the test, the system remained available and capable of automatic transfer from the test mode to the injection mode. There are no changes to any fundamental assumptions in the design basis or to the Technical Specifications involving this change. This test was conducted in a reactor pressure range where RCIC operability is not required.

Emergency Diesel Generator Ventilation and Radiator Fan Modifications
Safety Evaluation: 3310, 3298

Modifications to the EDGs resulted in the ability to re-establish summer design temperatures. These modifications were intended to improve the performance of the EDG cooling system. The modifications consisted of installing a higher capacity engine driven cooling fan, eliminating existing radiator bypass pathways, improving ventilation dampers and flows in the EDG rooms, modifying alarm circuits to detect damper misposition, and optimizing glycol concentrations in the engine jacket cooling water.

This change did not involve an unreviewed safety question. The new fan provides increased cooling flow while not adversely impacting the capability of the EDG to operate at the existing design capacity ratings or EDG fuel oil consumption under design basis conditions. The existing fan drive system components are fully capable of operating the new fan. The new fan is designed to operate under design basis seismic conditions and will not impose any unacceptable loads to the fan/EDG support structure.

The ventilation system modifications are safety related and seismically designed and will have no adverse impact on the structural integrity of the affected system. Operation of the removable and/or manually operated panels in the ventilation system will be incorporated into appropriate operating procedures and can be easily accomplished by the plant operators. The EDG ventilation system will be seasonally aligned and there are no additional short-term manual operator actions required during accident mitigation. Cooling system performance is further improved by reducing the glycol concentration in the jacket water coolant. A glycol concentration range is selected to provide adequate freeze protection during cold weather operation and higher heat transfer capabilities during warm weather operation.

Replacement of Turbine Building Closed Cooling Water Temperature Instrumentation
Safety Evaluation: 3311

The TBCCW system temperature transmitter and controller were replaced due to their obsolescence with newer digital equipment. This instrumentation controls TBCCW temperature by varying the bypass flow rate around the TBCCW heat exchanger.

This change did not involve an unreviewed safety question. The new instrumentation maintains the existing control modes. There are no functional changes to the system. This change increased the reliability of the equipment. The industry guideline for digital upgrades was followed. The software was reviewed and found to be acceptable. Failure modes were evaluated and were acceptable. Electromagnetic interference testing was performed with acceptable results. Power requirements are less and environmental conditions were found acceptable.

Replacement of Reactor Building Closed Cooling Water Temperature Instrumentation
Safety Evaluation: 3312

This change replaced the temperature transmitter, the temperature controller, and the temperature recorder for the RBCCW temperature control loops with newer digital equipment. The old equipment models were no longer available. This instrumentation controls RBCCW temperature by varying the bypass flow around the RBCCW heat exchanger. The change resulted in the same control modes as the existing controller (proportional plus integral).

This change did not involve an unreviewed safety question. The new instrumentation maintains the existing control modes. There are no functional changes to the system. This change increased the reliability of the equipment. The industry guideline for digital upgrades was followed. The software was reviewed and found to be acceptable. Failure modes were evaluated and were acceptable. Electromagnetic interference testing was performed with acceptable results. Power requirements are less and environmental conditions were found acceptable.

Replacement of Augmented Offgas Level Controllers
Safety Evaluation: 3313

This change replaced obsolete control equipment that provided level control for the AOG condensers with newer and more reliable equipment. The existing equipment had become an operational and maintenance problem. The AOG condensers cool the recombiner effluent and this instrumentation controls the water level inside the condensers. This change will result in the same control actions as the current equipment.

This change did not involve an unreviewed safety question. Replacement of controllers, power supplies, alarms, and level transmitters did not functionally impact the AOG System. The new equipment provides the same functions as the old equipment and controls the level in the AOG condensers in the same manner as the old controls. The new controls will increase the reliability of the AOG System. The new equipment is considered digital equipment and the industry guideline for digital upgrade was followed. The software was reviewed and found acceptable. Electromagnetic interference testing was performed with acceptable results. Power requirements are less and environmental conditions were found acceptable.

Spent Fuel Pool Capacity Expansion
Safety Evaluation: 3314

The primary objective of this design change was to increase the spent fuel storage capacity by adding two new spent fuel storage racks with a combined capacity of 513 fuel cells. This change increased the spent fuel storage capacity to 3404 fuel cells and provides the capability to accomplish full core off load, if needed, until the year 2009.

This change did not involve an unreviewed safety question. Technical Specifications were changed earlier to increase the licensed spent fuel storage capacity from 2320 to 3859 fuel cells. With the addition of these two racks, the present capacity of the spent fuel pool is still within Technical Specification limits. Technical Specifications were also changed earlier to allow Boral as the neutron absorbing material and to increase the loads allowed over the fuel assemblies from 1000 pounds to 2000 pounds respectively. The analyses of record demonstrate the acceptability of expanding the spent fuel pool storage capacity. These analyses demonstrate that K_{eff} will remain within acceptable limits even if an abnormal event such as a fuel assembly mis-loading or assembly drop occur. It has also been demonstrated that the spent fuel cooling system will continue to provide acceptable cooling of stored assemblies and that adequate time exists to take appropriate corrective action should all cooling be inadvertently lost. The racks are designed to seismic class I requirements. An assembly drop on the racks would not affect the ability of the racks to perform their function. The radiological consequences of a fuel handling accident remain within previously established licensing limits.

Replace 125 V DC Magnetic Breakers with Thermal-Magnetic Type Breakers
Safety Evaluation: 3315

This modification replaced magnetic-only feeder breakers in the 125-volt DC system with thermal-magnetic breakers. The new breakers have a time-overcurrent trip not present on older breakers.

This change did not involve an unreviewed safety question. Breaker replacement improved overall system reliability. The new breakers were sized for the component or system being powered resulting in no adverse effects on equipment reliability. They also have improved characteristics for containing electrical overload faults through enhanced breaker coordination. Failure modes of the new breakers are bounded by the previous analysis.

Temporary Procedure to Control Installation of Dampers in Control Room Ventilation Ductwork with Plant On-Line
Safety Evaluation: 3316, 3347, and 3349

A TP was created to allow modification and installation of dual, Class 1 isolation dampers in the six ventilation ducts that penetrate the control room boundary while the plant was online and the Control Room High Efficiency Air Filtration System (CRHEAFS) was required to be operable. The TP was developed to ensure that the Main Control Room Environmental Control System (MCRECS) normal cooling function was maintained to the greatest extent practicable and CRHEAFS operability would not be affected during the construction period.

This activity did not involve an unreviewed safety question. Implementation of the TP had no adverse effect on the function of CHREAFS. During initial stages of construction when larger access panels were being cut into the normal air supply ductwork to allow installation of the duct blank-off devices a CRHEAFS LCO was entered. The work was judged to be an easy task to perform of short duration that could be successfully performed well within the time constraints of the LCO. When the ductwork integrity was restored the LCO was exited. After initial construction, blank-off devices were installed in ductwork prior to any ductwork being breached that could compromise the control room boundary. CRHEAFS was still capable of providing filtered outside air to the main control room and maintaining it at a positive pressure if required during construction activities. No CRHEAFS ductwork or components were modified or in any way disturbed. Fire dampers in each section of ductwork to be worked were closed to provide isolation. Leakage and seismic capabilities were assessed and found acceptable. Impact on the MCR environment was assessed and found to be acceptable for operator comfort and equipment functionality. Prior to crediting the isolation devices the CRHEAFS was operated to ensure positive pressure could be maintained.

Additional operator actions that may be required to initiate CRHEAFS prior to construction activities being complete were evaluated against the criteria in Information Notice 97-78 and are acceptable. The actions are contained in written operating procedures and did not increase the total CRHEAFS manual initiation time beyond the thirty minutes assumed in accident analyses. TP 00-008 maintains the operability of the CRHEAF system and ensures that the control room envelope can be reliably isolated within the time assumed in the accident analyses should it be required anytime during the modification sequence.

Revise FSAR HPCI System Analysis and Requirements and Related Surveillance Test Procedure
Safety Evaluation: 3317

This change incorporates information into the FSAR that defines performance requirements for the HPCI system derived from analysis of small pipe breaks. The Technical Specifications require the HPCI pump to deliver 4250 gpm for a system head corresponding to a reactor pressure of 1000 psig. Previously, the upper value of 1000 psig in the operating range was determined to be nonconservative because a range of 150 psig to 1125 psig is an input to current core cooling analysis. However further reviews of HPCI system design basis and accident analyses have determined that pipe break analysis requirements for a flow rate of 4250 gpm are met at a system head corresponding to a reactor pressure of 1000 psig. The change also revises the related pump test surveillance procedure to be consistent with the revised FSAR requirements and the Technical Specifications.

The proposed changes to the FSAR describe the requirements for system operation more fully than before and the procedure changes are designed to provide assurance by test of system capability to meet the design requirements. HPCI provides core cooling following transients and accidents. The proposed pump testing provides assurance that HPCI can provide core cooling over the applicable range of reactor pressures and flow rates. The pump test acceptance criteria has changed but the test method was not altered. The proposed changes do not involve any mechanism whereby a greater amount of radioactive material can be released to the surroundings. The changes did not reduce or change the design capacity of the system, minimum performance requirements, design margins, ability to depressurize the reactor or provide core cooling.

Temporary Modification to Bypass a Single Source Range Monitor Not Full In Rod Block Signal
Safety Evaluation: 3318

The drive motor for a single Source Range Monitor (SRM) was degraded such that its limit switch would not provide a full-in signal to the Reactor Manual Control System (RMCS) logic. This was causing an unnecessary rod block. The SRM was fully inserted manually and a temporary jumper was installed to provide the detector full in condition to the RMCS and thereby allow control rod motion.

This TM did not involve an unreviewed safety question. The SRM was fully inserted and its position verified by direct observation of resistance to movement and from SRM drive cable position. The jumper provided the correct information to the RMCS by a means other than the drive motor limit switch. The SRM itself was functionally operable and unaffected by this TM. The jumper could not introduce a failure mode that would render other SRMs inoperable. Also, the SRM select pushbutton was red tagged to prevent inadvertent movement of the SRM.

Evaluation of a Delay in Low Pressure Coolant Injection Flow to the Reactor During a Design Basis Loss of Coolant Accident
Safety Evaluation: 3319

The Design Basis LOCA analysis assumes the time period that elapses between the initiation of a postulated recirculation line break in one loop and the closure of the recirculation pump discharge valve on the intact loop is 49.7 seconds. Credit for full LPCI flow is assumed at this time. However, calculations showed that swing bus transfer time and other instrumentation time delays could result in a valve closure time of 55.9 seconds. A conservative value of 59.7 seconds was assumed in this evaluation and the effect on the LOCA analysis was analyzed.

This change did not involve an unreviewed safety question. The limiting event for the LOCA analysis is the case of LPCI injection valve failure. The calculated delay times had no effect on this case. The Peak Clad Temperature (PCT) increased to 1794°F for the less limiting battery failure case combined with the additional delay in valve closing. This value is less than the limiting Appendix K PCT of 1821°F and less than the Licensing Basis PCT of 1825°F and much less than the acceptance criteria in 10 CFR 50.46. The criteria were also met for local metal water reaction and core wide metal water reaction.

Temporary Modification to Install Chart Recorder on Recirculation Pump Scoop Tube Positioner Circuit
Safety Evaluation: 3320, 3325

This TM installed a chart recorder on the recirculation pump scoop tube positioner circuit in order to monitor the circuit while the plant was operating. The additional monitoring capability was necessary to diagnose controller problems that affected recirculation pump speed. In addition, a temporary mechanical stop was installed on the scoop tube positioner crankarm to prevent recirculation pump speed increases greater than 2% that could result from potential spurious controller signals.

This TM did not involve an unreviewed safety question. This change did not alter response characteristics of the scoop tube positioner circuit or the recirculation flow controller's reaction to changing conditions in the reactor. Thus, consequences of transients or accidents analyzed in the FSAR are not affected. Installation of the TM considered interactions with the monitored circuits and appropriate measures were taken to ensure the probability of transients evaluated in the FSAR was not increased. This TM did not affect the existing speed, flow or power indications available to operators in the MCR. The location of the chart recorder was chosen to prevent any potential seismic interactions with safety related equipment. The circuits being monitored are low voltage and the possibility of electrical faults causing fire was precluded.

Replace Reactor Water Cleanup Flow Control Instrument Loops
Safety Evaluation: 3321

The RWCU filter/demineralizer flow control instrument loops were upgraded to remove obsolete flow controllers and replaced with pneumatic gradual control switches. This change provides the capability for precise manual adjustment of RWCU flow and removes the capability to automatically control RWCU flow. The flow recorders were also replaced with pneumatic indicators.

This change did not involve an unreviewed safety question. Automatic control of RWCU is not required. Manual adjustment will be more reliable and allow for finer flow control. Automatic closure of the flow control valve is not affected. The RWCU system is not required to operate during or following any event analyzed in the FSAR. Automatic isolation of the system from the primary containment is not affected. The modification does not impact RWCU pressure integrity.

New Standby Gas Treatment Air Accumulator and Air Compressor
Safety Evaluation: 3322

The modification involved the installation of an additional safety-related, ASME Section VIII, Standby Gas Treatment System (SBGTS) air accumulator tank to tie into the existing air supply system. The air supply system provides motive force for system dampers. This plant design change also installed a non safety-related air compressor to tie into the existing SBGT air supply system to be used as an alternate means of charging the SBGTS air accumulator tanks. The new compressor will also tie into the existing Torus Vacuum Breaker (TVB) air supply system and will be used as an alternate method of charging the TVB air accumulator tanks. The Low Pressure alarm setpoint for the accumulator was increased. This change increases the allowable leakage margin by assuring that a higher minimum pressure is maintained in the SBGT air supply system.

This change did not involve an unreviewed safety question. The addition of an additional SBGTS air accumulator tank and an increase in minimum system operating pressure does not adversely affect the safety function of the SBGTS or the SBGTS air supply system. The new accumulator tanks and Low Pressure set point change only increases the air supply capacity of the system, which increases the design margin for allowable daily air leakage. The existing high-pressure charging air bottle system will remain in place and system operation will not change. The addition of the new compressor will not alter the existing air supply system. The compressor only provides an alternate method of manually charging the air accumulator tanks and eliminates the need for personnel to move high-pressure gas bottles through the reactor building. The TVB air supply system is similarly improved. Proper overpressure protection is provided for both systems, and independence is maintained between safety and non-safety related portions.

New Maintenance Procedure to Standardize Maintenance on RBCCW Pumps
Safety Evaluation: 3323

The new procedure provides the work steps necessary to perform routine maintenance on the RBCCW pumps. There are several attachments, each applicable to a particular work activity. The attachments include a pump check-off sheet, coupling preventive maintenance checklist, coupling alignment data, seal assembly removal/installation, rebuild of an RBCCW pump, and installation/removal of a temporary support. During the period required to perform a pump overhaul, a temporary support will be installed in place of the applicable RBCCW pump. Including this work in this procedure obviates the need to perform a TM.

This new procedure does not involve an unreviewed safety question. While an RBCCW pump is being removed and the temporary support is being installed, an LCO is declared because the respective loop of the RBCCW system will not be operable. While the temporary support is installed, the RBCCW system will function as designed and there is no adverse impact on the safety design basis for RBCCW or on overall safety of the plant. The only difference is that the third pump in that loop will not be available as a spare, which is within the design basis for the system. The temporary support will provide the necessary support to the mating piping to allow the loop to remain operable. It does not provide a pressure boundary; the flanges on the temporary support are for support only. The normal isolation valves provide the pressure boundary required for maintenance. The temporary support has been designed for the same dead weight, thermal and seismic loading as an RBCCW pump, and therefore is equivalent to an RBCCW pump relative to support. The existing piping analysis for the RBCCW system is valid when the temporary support is installed. Likewise, while the temporary support is being removed and the applicable RBCCW pump is reinstalled, an LCO will be required because the applicable RBCCW loop will again be inoperable. Following maintenance on an RBCCW pump, an operability test in accordance with PNPS procedures is performed as a minimum as part of the post work testing.

Addition of an Isolation Valve in the Nitrogen System Supply to each Inboard MSIV Accumulator
Safety Evaluation: 3324

Prior to implementation of this design change, the nitrogen supply to the four inboard MSIV accumulators was controlled by one two-inch isolation valve on a common header. The two-inch common header splits into four one-inch nitrogen supply lines that individually feed the four inboard MSIV accumulators. This change adds four one-inch isolation valves in each of the one-inch nitrogen supply lines to allow isolation and testing of individual MSIVs.

This change did not involve an unreviewed safety question. The nitrogen supply system to the four individual MSIV accumulators provides a reliable source to maintain the MSIVs in the open position, and to exercise the individual MSIVs when required. The accumulator volume is adequate to provide full stroking of the MSIV through one-half cycle (open/close) if the nitrogen supply to the accumulators has failed. The nitrogen supply line to the accumulators is large enough to make up pressure to the accumulators at a faster rate than pressure bleeding from valve operation during valve opening and closing. Also, the nitrogen supply is capable of maintaining an individual MSIV in the open position. The nitrogen supply from the accumulator, and the valve spring force are each capable of independently closing a MSIV against full reactor pressure and this function is not affected. The position of the newly installed one-inch gate valves is controlled by procedure and they are left in the normally open position. In this position the new isolation valves will not alter the nitrogen system pressure or flow delivery. Therefore there will be no adverse effect on the function of the nitrogen supply system to the inboard MSIV accumulators.

Procedure Revision to Permit Manual Control of RBCCW Loop Temperature
Safety Evaluation: 3326

RBCCW temperature is automatically controlled by a control valve that varies the quantity of water flowing through the bypass line around each RBCCW heat exchanger. The RBCCW system operating procedure was revised to permit isolation of the RBCCW heat exchanger bypass lines. Temperature control can be performed by manual operation of the heat exchanger Salt Service Water (SSW) outlet valves. This change was made to enhance operational flexibility.

This change did not involve an unreviewed safety question. The procedure provides instructions for maintaining RBCCW temperature within the design operating range while the automatic temperature control function is isolated. The temperature control function is non-safety related and is designed to fail in the closed position on loss of instrument air or control power to maximize water flow through the RBCCW heat exchangers. This configuration maximizes cooling flow. The SSW outlet valves receive an automatic signal to fully open under certain accident conditions thereby maximizing cooling. The proposed procedure does not override or defeat these interlocks.

Temporary Modification to Place Torus to Reactor Building Vacuum Breaker in Closed Position to Allow Maintenance On its Actuator
Safety Evaluation: 3327

This TM placed the torus to reactor building vacuum breaker in the closed position by installing a valve stem gag in order to perform maintenance on its valve actuator with the plant operating. This valve functions to provide containment isolation in its closed position and provides vacuum relief for the torus in its open position. The isolation function was operable with the gag in place but the relief function was inoperable. Since the isolation function and the redundant torus to reactor building vacuum breaker train were operable, Technical Specifications allowed this configuration to exist for up to 7 days.

This change did not involve an unreviewed safety question. The valve gag components provide an isolation capability that is greater than the actuator. The gag configuration was evaluated for seismic and LOCA forces. The gag provides a passive restraint to movement and would not damage the valve or stem while it was installed. The gag had no interaction with any other component.

Modify RHR Torus Spray Isolation Valve Bonnet Vent to Preclude Potential for Pressure Locking
Safety Evaluation: 3328

NRC Generic Letter 95-07 required that utilities evaluate safety-related power operated valves for susceptibility to the phenomenon of pressure locking (PL) and thermal binding (TB). To address this issue, this design change modified the existing RHR torus spray isolation valve bonnet vent line to address all possible PL scenarios. The modification required removing the existing bonnet vent relief valve and connecting the bonnet vent to an existing downstream RHR test connection between the valves and the torus.

This change did not involve an unreviewed safety question. The evaluation considered the impact of the modification on the RHR and Primary Containment systems. The torus spray function is to reduce torus pressure and temperature by spraying the free space in the torus with water. The modification has no impact on this function. The modification provides a continuous vent path from the valve bonnet to the downstream (torus side) of the valve while in the closed position. While in the closed position the valve is credited for providing an isolation boundary between the containment (torus) and the RHR system. The vent path installed by this design change allows for the valve bonnet to communicate directly with a downstream system test connection. These two sections of primary containment were previously separated when the valve was closed. Both the valve bonnet and the test connection are within primary containment boundary and are designed, constructed and tested to the same requirements and quality standards as the primary containment and its integrity is not adversely affected. The modification established the upstream valve disc as the sealing surface and Local Leak Rate Testing ensures any leakage is within acceptable limits.

Change EDG Fuel Consumption Rate and Day Tank Capacity Description in FSAR
Safety Evaluation: 3329

This change corrects the FSAR description of the capacity of the EDG fuel storage day tanks and the EDG fuel consumption rate. The FSAR currently states the fuel consumption rate is .0764 gal/kW/hr. This was changed to 222 gallons per hour at 2860 kW. The FSAR also states that the EDG fuel storage day tanks provide enough fuel for a minimum of 2.5 hours of full load EDG operation. Based on the current EDG fuel consumption rates the actual value is 2.3 hours. The change also eliminates the EDG fuel consumption rate from the Technical Specifications Bases that is currently inconsistent with the rate currently assumed.

This change did not involve an unreviewed safety question. The changes did not affect the reliability or availability of the EDGs to perform their design basis functions. The EDGs are able to start, load and run as before. It does not affect plant mitigation strategies if an EDG fails. These fuel consumption rates formed the basis for License Amendment 184 and these changes are consistent with it. The fuel in the EDG day tank is not credited toward the seven-day onsite fuel storage requirement, thus there is no reduction in safety margin. This change merely changed the FSAR to be consistent with licensing basis information.

Install New Drain Line Instruments and Valves for Main Condenser Gland Seal Condenser Flow
Safety Evaluation: 3330

This modification re-routed Gland Seal Condenser (GSC) condensate to a new atmospheric drain tank from which the condensate can be recycled by vacuum drag to the main condenser hotwell via a normal drain line. When vacuum is unavailable, the condensate can drain from the new tank to the turbine building equipment sump via an overflow line.

The capability to route condensate from the gland seal condenser back to the main condenser is a feature described in the FSAR. This modification implements the feature by use of a separate tank instead of the turbine building equipment sump. Fundamentally, the characteristics of the proposed modification are the same as the original scheme and do not create the possibility of a different type of malfunction of equipment important to safety or potential for release of radioactive material to the environment than previously evaluated in the Final Safety Analysis Report. There are no Technical Specifications or TS Bases that directly relate to the handling of gland seal condensate. However, Technical Specification administrative requirements are in place for a Radioactive Effluent Controls Program that includes liquid and gaseous radioactive effluents. This modification reduces the quantity of liquid radwaste potentially requiring discharge to the environment.

Modifications to RWCU Return Isolation Valve
Safety Evaluation: 3331

To improve performance and test accuracy, the RWCU outboard return isolation valve actuator was increased from Limitorque size SMB-0 to SMB-1, the motor was increased from 25 ft-lb. to 40 ft-lb. and the overall gear ratio was increased. The valve yoke was changed, the valve disc replaced and the stem replaced with a Smartstem. The changes were required in order for the MOV to achieve the required output thrust and torque under design basis accident conditions. These changes are consistent with other changes to other MOVs to comply with the requirements of NRC GL 89-10.

This change did not involve an unreviewed safety question. The increase in motor size resulted in an increase of 1.4KW load on each EDG causing the peak short term load following a LOCA with Loss of Offsite Power to increase slightly. This additional loading is still well below the EDGs peak short-term load. The changes also resulted in a change in stroke time but remain below the FSAR maximum stroke time of 30 seconds. Due to the larger motor, which causes a larger inrush starting current, all other MOVs that operate during load block 1 and 2 were evaluated to have adequate voltage margin. The valve yoke is larger and was designed to accommodate the higher thrust loads. The higher thrust loads are less than the weak link limit. Seismic qualification of the valve/piping system is maintained.

Increase Stroke Time of RHR Pump Torus Suction Valves
Safety Evaluation: 3332

The required stroke time for the RHR pump torus suction valves was increased from 120 seconds to 150 seconds. In order for the motor operator to develop the required thrust and torque to operate these valves under all design basis conditions with adequate margin, the gearing ratio must be changed. The change in gearing ratio required an increase in valve stroke time.

This change did not involve an unreviewed safety question. The valves are normally open for operation in the LPCI, suppression pool cooling and containment spray modes and this change does not affect their position. A review of design and licensing basis documentation revealed no operational, analytical or any other limitations on extending the valve stroke time. The valve must be capable of closing after an accident to maintain torus water inventory. The increase in valve stroke time has no effect on this function. The valve is also required to open if the RHR system is in the Shutdown Cooling mode and needs to be realigned to the LPCI mode. This is a manual operation that requires many other manipulations by operators. A stroke time increase of thirty seconds has no adverse affect. The increased gear ratio improves the thrust margin for the valves. Valve structural limits are maintained. Motor overheating due to a longer operating time was assessed and is of no concern.

Revise Procedures to Incorporate a Revision to the Definition of Adequate Core Cooling
Safety Evaluation: 3333

Emergency Operating Procedures (EOP) defined adequate core cooling as being fulfilled by water level at Top of Active Fuel (TAF) or, under very specific plant conditions, by establishing steam cooling. However, for a DB LOCA, reactor water level would be limited to 2/3-core height, and EOPs would require operators to prematurely flood and vent primary containment as a result. This change adds a third definition for adequate core cooling; 2/3-core coverage with a minimum core spray system flow rate of 3,600 gpm.

This change did not involve an unreviewed safety question. The current direction to operators in the EOPs was a result of the best guidance available at the time and covered all design basis and beyond design basis events that could impact adequate core cooling. The NRC concurred with this guidance. After recognition that the guidance could be nonconservative for the DB LOCA, additional analysis was performed. The current licensing basis analysis for the DB LOCA shows that adequate core cooling would be achieved during core reflood with one core spray pump and two RHR pumps operating in the LPCI mode. After initial reflood adequate core cooling would be achieved with only one core spray pump allowing the other two RHR pumps to be diverted to containment cooling. Analysis of the adequacy of long term core cooling with this configuration was performed and found to be adequate. The analysis considered fuel cladding perforation, metal/water reaction, core spray distribution pattern, flow rate per fuel bundle, flow diversion and other factors. The NRC approved this analysis and procedures have been revised accordingly.

Temporary Modification to Raise Alarm Set Points for "B" Safety Relief Valve Tailpipe Temperature
Safety Evaluation: 3334

This TM reconfigured the alarm circuitry in the SRV Temperature Recorder and the alarm setting for the computer data point. These changes served to eliminate continuous annunciation in the MCR. This change was made due to leakage through the SRV pilot valve. The plant changes made are temporary and normal settings will be restored after SRV repair.

This change did not involve an unreviewed safety question. The proposed TM of the alarm set points does not involve any initiators associated with accidents described in the FSAR. The recorder and computer point affected by the proposed TM are not relied on to directly mitigate or provide operators information essential for the mitigation of any of the accidents evaluated in the FSAR. The recorder and computer point affected by the proposed TM provide information to operators on the current state of the SRV. The recorder also provides temperature trend information and alarms for all relief and safety valves. The proposed TM, by raising the alarm setpoint, returns functionality of the alarms for each of the other safety and relief valves. The consequences of malfunction of equipment important to safety are not affected by this change. The changes to the tail pipe temperature measurement instrument loops do not affect the read-out capability of the loops. Therefore, the loops are fully capable of providing their function of providing the operators with current tail pipe temperature information. The limiting conditions, surveillance requirements, bases, margins and required actions remain unchanged by the proposed alarm setpoint changes.

Temporary Procedure to Replace RBCCW Temperature Transmitters
Safety Evaluation: 3335

This TP provides instructions for RBCCW System operation with the RBCCW heat exchanger bypass line isolated to allow replacement of two temperature transmitters. The current operating procedure permits isolation of the heat exchanger bypass lines and manual control of loop temperature. The purpose of this TP is to allow the use of alternative measurement devices to monitor RBCCW loop temperature while the temperature transmitters are being replaced.

This change did not involve an unreviewed safety question. The alternate temperature indicators used by this TP to monitor RBCCW heat exchanger outlet temperature are of the same quality class, reliability and accuracy as the ones for which they are being temporarily substituted. They are calibrated using the same procedure, to the same level of accuracy, and are used as inputs for RBCCW heat exchanger performance calculations. This ensures that the accuracy of measurements made by the alternate indicators will be equivalent to the ones for which they are substituted and ensures accurate representation of the system temperature. The TP allows operation of the RBCCW system while the bypass line is isolated and provides administrative controls while operating in this configuration. Operation with the bypass line closed provides maximum cooling to the RBCCW heat exchanger. Therefore there is no adverse impact on the safety function of the RBCCW system.

Standing Plant Design Change for Minor Electrical Modifications - 2001
Safety Evaluation: 3337

This PDC is a standing document that addresses specific, limited classes of well-defined modifications that do not constitute changes in the facility as described in the FSAR. A safety evaluation is performed solely for administrative convenience. Changes are limited to changes that do not affect the system design basis, function or operation and in all cases comply with FSAR requirements.

Standing Plant Design Change for Minor Mechanical and Civil/Structural Modifications - 2001
Safety Evaluation: 3338

This PDC is a standing document that addresses specific, limited classes of well-defined modifications that do not constitute changes in the facility as described in the FSAR. A safety evaluation is performed solely for administrative convenience. Changes are limited to physical changes that do not affect the system design basis, function or operation and in all cases comply with FSAR requirements.

Standing Plant Design Change for Minor Instrumentation and Control Modifications - 2001
Safety Evaluation: 3339

This PDC is a standing document that addresses specific, limited classes of well-defined modifications that do not constitute changes in the facility as described in the FSAR. A safety evaluation is performed solely for administrative convenience. Changes are limited to changes that do not affect the system design basis, function or operation and in all cases comply with FSAR requirements.

Temporary Modification to Install Chart Recorder on Feedwater Flow Instrumentation
Safety Evaluation: 3341

This TM installed a chart recorder on the reactor feedwater flow instruments of the feedwater control system for trouble shooting and monitoring purposes. Existing test point connections were used to monitor input and output signals. This TM was installed while the plant was operating.

The change did not involve an unreviewed safety question. The proposed TM did not alter the response characteristics of the feedwater control system. The feedwater flow instrument circuit has been designed with pre-engineered test points to allow the system to be monitored. The chart recorder was isolated by design such that the feedwater flow instruments were not affected by any chart recorder failures. If the chart recorder test probes were to become disconnected and short, which is most likely to occur during installation or removal, protection provided by high impedance resistors in the test point locations would limit the impact on the circuit. Therefore, the probability and consequences of transients and accidents evaluated in the FSAR are not affected by this modification.

Temporary Procedure for Troubleshooting the Main Turbine Mechanical Pressure Regulator
Safety Evaluation: 3342

This TP provided technical guidance for on-line troubleshooting of the mechanical pressure regulator (MPR). The MPR servomotor position was experiencing oscillations. Based on industry experience, it was believed that foreign material in the tuning valve for the MPR was preventing it from functioning properly. The intent of the TP was to exercise the tuning valve in an attempt to clear any possible obstructions.

This procedure did not involve an unreviewed safety question. The function of the MPR is to backup the electric pressure regulator (EPR) and limit the increase in reactor pressure if the EPR failed. The tuning valve would be cycled several times. Closure of the tuning valve does not render the MPR inoperable nor does it affect primary EPR control. Operator actions were addressed and controls included in the TP. Criteria were provided for recognizing pressure control failure and include actions for restoring MPR function. This TP was not expected to result in a MPR system or valve malfunction and it would still be capable to operate in accordance with the assumptions for analyzed transients in the FSAR.

Temporary Procedure for Post Work Testing of Swing Bus Transfer Logic Relays
Safety Evaluation: 3344

All essential 480V auxiliaries required during abnormal operational transients and accidents are supplied from three emergency service busses: B1, B2 and B6. Bus B1 (B2) receives emergency power from 4160V bus A5 (A6) and in turn this bus receives power from EDG A (EDG B). Bus B6 preferentially receives power from B1 and if not available, from B2. If offsite power is lost or electrical offsite power grid degraded voltage conditions exists, automatic starting and loading of the EDGs is required. Under these conditions, as the EDGs come up to rated speed and voltage and begin to close onto their respective busses, the potential exists for both B1 and B2 to close onto B6 simultaneously. Relays had been installed to delay B2 closure for two seconds, but defects were discovered that could prevent the delay. New relays were installed to restore this design function. The purpose of this TP was to provide the necessary controls and instructions to perform post-work testing in a safe, deliberate and controlled manner.

This change did not involve an unreviewed safety question. This TP was implemented with the plant in operation but in a 7 day LCO because of B6 isolation. Administrative and physical controls implemented by the TP ensured that any adverse effects of the post work testing were confined to B6 and its associated supply breakers and controls. Loss of B6 had already been analyzed and accounted for in station design and procedures. The SSW division control valves were placed in their post-accident position so that any transient on the system would be eliminated. The controls in the TP ensured that the objectives were achieved and that configuration control was maintained such that the plant remained within its licensing basis at all times.

Increase SGTS Inlet Air Heater Capacity
Safety Evaluation: 3345

The SGTS inlet air heater units were designed to decrease the relative humidity (RH) of the inlet air stream to be no greater than 70% RH under inlet design basis conditions. Degraded voltage conditions can decrease the effective capacity of the heaters. Based on revised calculations, the capacity of the heaters for degraded voltage conditions needed to be increased. The heater units consist of four banks with three heating elements in each bank. Prior to this modification, one bank of heating elements was wired in a "delta" configuration and the other three were configured as "Y". One additional heater bank was configured as "delta", increasing the rated capacity of the heaters for normal and degraded voltage conditions.

This modification did not involve an unreviewed safety question. Loading on the EDGs due to the increased heater capacity was assessed and found to be acceptable. The additional heat load and its effect on room temperature and environmental qualification of equipment was evaluated. The air stream temperature and effect on halogen decay heat loading was determined to be within the limits for the SGTS charcoal banks. Relays that trip to isolate heater failures were assessed. Cable ampacity and degraded voltage calculations were also revised. Technical Specifications will be revised via a license amendment request to reflect the new heater capacity requirements.

Temporary Procedure for On-Line Isolation of TBCCW "B" Heat Exchanger for Cleaning and Tube Plugging
Safety Evaluation: 3346

This TP enabled the "B" TBCCW heat exchanger to be temporarily secured and isolated from both the TBCCW and SSW systems to verify that on-line cleaning and tube plugging can be performed successfully.

The change did not involve an unreviewed safety question. The TBCCW system is not required to operate for any event analyzed in the FSAR. Isolation of the heat exchanger has no effect on the ability of the RBCCW or SSW systems to perform their safety functions. Isolation of the heat exchanger could result in increased flow through the RBCCW heat exchanger. The increase in differential pressure was evaluated to be acceptable. The procedure would be aborted if loss of AC power occurred and the TBCCW and SSW systems would be restored to their normal lineup. RBCCW system temperatures are monitored and the procedure would be aborted if they increased to unacceptable levels. TBCCW system temperatures were evaluated with one heat exchanger out of service and it was determined that the remaining heat exchanger could maintain system temperatures within acceptable limits. As an added precaution, the TP is planned for implementation in the winter months when ultimate heat sink temperatures are lower.

Procedure Revisions to Shutoff Access Control Area Fans During CRHEAFS Startup
Safety Evaluation: 3348

The procedure revisions ensure that the MCR access control area fans are shutdown so that they cannot create a condition in the space adjacent to the control room north wall that could challenge CRHEAFS ability to maintain a positive pressure in the MCR.

This change did not involve an unreviewed safety question. The effect of shutting down Control Room Access Area Fans would be to remove a potential source of positive pressure in the area adjacent to the control room north wall. These actions remove the potential for unfiltered air to be forced into the control room around the main control room door against CRHEAFS pressure. There is no impact on the conclusions of the control room habitability radiological analysis. This analysis assumes that the normal supply and recirculation fans remain in operation for thirty minutes following receipt of either the Control Room Air Inlet Radiation Monitor alarm or the Control Room Area Radiation Monitor alarm. The new actions to turn off the access control area fans in addition to the normal control room supply and exhaust fans were evaluated against Information Notice 97-78. These actions were proceduralized and will not increase the total CRHEAFS manual initiation time beyond the thirty minutes assumed in accident analyses.

Removal and Abandonment of the Condenser Deaerating Sparger Line
Safety Evaluation: 3350

This change allowed for the complete removal, or the cutting, capping and abandonment of the condenser deaerating sparger line (CDSL). The purpose of the CDSL was to provide main steam to the main condenser during periods of low load or low circulating water temperatures to supplement the condensate deaeration process. The CDSL has not been used and is not needed. Valve leaks in the line contributed to a reduction in condenser vacuum and abandoning and/or removing portions of this line will increase plant efficiency.

This change did not involve an unreviewed safety question. Removal of the CDSL will reduce the potential for a loss of vacuum transient due to inleakage. Oxygen levels in the condensate are acceptable under low power conditions without this line. Removal of the CDSL will have no effect on operation or pressure boundary of the main steam system. The modification will also reduce Main Steam Isolation Valve leakage from the main condenser under post-accident conditions serving to reduce radiological consequences to the public and control room personnel.

Change Molded Case Circuit Breaker Instantaneous Trip Acceptance Criterion
Safety Evaluation: 3351

Circuit breaker instantaneous (magnetic) trip tests are performed to verify the breaker functions within vendor published trip curves. The current acceptance criterion is too short due to limitations in field testing equipment and test configurations. The acceptance criterion was increased to be consistent with the capabilities of the test equipment.

The change did not involve an unreviewed safety question. The breakers are designed to interrupt maximum fault current, support in-rush current and operating load, provide margin between primary and secondary protection (i.e. coordination) and provide overload protection for cables. The only function affected by the change was coordination. The trip acceptance time was increased from ≤ 0.04 seconds to < 0.09 seconds for all 480V molded case circuit breakers except those associated with electrical penetrations. The change in acceptance criterion is still more conservative than the NEMA-AB4 testing criterion of 5 to 10 cycles (0.083 to 0.166 seconds). This acceptance criterion provides assurance that the breaker magnetic trip unit is functional. The evaluation did not identify any other adverse impacts.

Reduce Reactor Water Cleanup Inlet Isolation Valve Stroke Time
Safety Evaluation: 3352

The RWCU inlet isolation valve maximum allowable stroke time used in Pipe Break Outside Containment (PBOC) analyses of RWCU pipe breaks was reduced from twenty-five seconds to twenty seconds. PBOC analyses assume a pipe break occurs concurrently with a loss of offsite power (LOOP). Time delays in electrical power supply logic were discovered that could adversely affect the valve closure time assumed in the analysis. As a result, the stroke time for the valve was reduced so that newly discovered time delays combined with the allowable stroke time would still be bounded by the total closure time assumed in the analysis. There was no physical modification made to the valve.

This change did not involve an unreviewed safety question. The total allowable valve closing time from the pipe break to the valve being completely closed affects the mass and energy release to the secondary containment. There was no affect on these analyses since the total closing time did not change. The valve is fully capable of closing within the reduced stroke time without any physical modifications.

Temporary Modification to Install Alternate Above-Ground SSW Discharge Piping
Safety Evaluation: 3353

A temporary pipe was installed on the A (B) loop of the SSW return piping from the pipe vault, to ground level and out to the discharge canal. This modification allowed repair of the buried loop of A (B) SSW piping to proceed while the other SSW loop remained in service. This TM was only installed during plant cold conditions (<212°F). This change also revised TS Bases Section B3/4.5.B.3 and B3/4.5.B.4 to reflect when RBCCW and SSW are required to be operable.

This change did not involve an unreviewed safety question. Its purpose was to maintain a SSW loop available during repairs to the underground discharge piping. As the loop being worked was not required to be operable in accordance with Technical Specifications, the use of a nonseismic temporary pipe was satisfactory. If the pipe broke in the pipe vault, the water would be contained in the vault. The vault does not contain electrical equipment whose function could be affected. If the pipe broke at ground level the resulting flow would be directed to the ocean and have no impact on other plant equipment. The change also revised TS Bases to reflect that RBCCW and SSW are not required during cold shutdown or refueling. During cold shutdown or refueling, the operability requirements of the RBCCW and SSW systems are determined by the systems they support.

Restore Control Rod Drive System Flow Stabilizer Loop
Safety Evaluation: 3354

This modification restored an original design feature of the Control Rod Drive (CRD) system in order to enhance system performance by stabilizing system flow rates and reducing system pressure transients when individual control rods are moved. Carbon steel piping associated with the system was replaced with stainless steel pipe to reduce the potential for corrosion products to enter individual CRD units.

This change did not involve an unreviewed safety question. The modification had no adverse effects on the ability to shutdown the reactor since the neutron monitoring system, reactor protection system and reactor manual control system were unaffected. Requirements for reactivity control, control rod drive operability, scram insertion times and CRD accumulator operability were unaffected. System functionality was improved so the probability of equipment malfunctions was reduced.

Storage of Solid and Dewatered Radioactive Waste in the Low Level Radwaste Storage Facility
Safety Evaluation: 3355

Solid radioactive waste/material is temporarily stored in the Low Level Radwaste Storage Facility (LLRWSF) prior to shipment for burial. The change allows for the temporary storage of dewatered wet radioactive waste/material, in addition to solidified waste/material, in the LLRWSF prior to shipment to a processing facility for preparation for burial. The change revised the FSAR to include a complete description of the type of radioactive waste/material that can be stored temporarily in the LLRWSF.

This change did not involve an unreviewed safety question. The change did not involve any modifications to the design of the LLRWSF. The LLRWSF and the waste storage containers that will be used for the temporary storage of radioactive waste were evaluated against the criteria provided in Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Waste at Power Reactor Sites." The design features of the LLRWSF and the storage containers were evaluated to ensure that the radiological consequences of design basis events (fire, tornado, seismic event, flood) would not exceed a small fraction (10%) of 10 CFR 100 (i. e., no more than a few Rem whole body dose).

Provide Temporary Power to Instrumentation and Control Power Supplies During Plant Outages
Safety Evaluation: 3356

A new plant procedure was created to provide instructions for installing temporary power to 120 VAC power supply panels. The panels provide power to various safety-related and other important instrumentation and controls. The temporary power would be supplied during outages when the normal power supply is not available due to testing or maintenance. The temporary power feed enabled the equipment powered from the panels to be available for use.

This change did not involve an unreviewed safety question. With the reactor in a cold condition, temporary power to these panels enhanced equipment availability and increased capability to monitor equipment status. The procedure provides prerequisites that prevent temporary power from being provided when primary and secondary containment was required or fuel was being moved. The temporary feed is not credited to maintain system operability to meet the requirements of Technical Specifications. The power cables equal the capacity of the permanent supply cables. Panel loading when the procedure is used is within existing loading assumptions. Breaker coordination and fault protection is maintained. Plant configuration is maintained so there will be no adverse effects on equipment required to be operable.

Install Jumpers Around High Reactor Water Level Primary Containment Isolation Logic During Refueling Conditions
Safety Evaluation: 3357

A station procedure was revised to allow station personnel to disable the Primary Containment Isolation System (PCIS) Group 1 isolation signal (automatic closure function) during reactor shutdown when Technical Specifications do not require the PCIS signal to be operable. This procedure installs jumpers across contacts of Group 1 isolation logic relays in the circuits of main steam line isolation, main steam drain and recirculation loop sample valves in order to bypass the automatic closure function. This procedure change was intended to allow operation and stroke testing of the valves during cold shutdown.

This change did not involve an unreviewed safety question. The procedure provides administrative controls to ensure that it is only implemented when PCIS is not required and establishes system isolation boundaries prior to installing the jumpers. A single jumper disconnection or short to ground would result in an open circuit across the relay contact or opening of the circuit protective device. Either failure would result in valves remaining as-is or closing per design. Setting isolation boundaries ensures an inadvertent drain-down of the reactor vessel would not occur. This modification shall remain in effect only during the refueling outage while containment integrity is not required in accordance with Technical Specifications.

Reload 13 Core Design
Safety Evaluation: 3358

This safety evaluation addresses the changes to the Pilgrim reactor core during RFO 13. The change involved the replacement of 144 irradiated fuel assemblies with 144 fresh fuel assemblies. The fresh fuel assemblies are based on the 10x10 GE14 bundle type with a bundle average enrichment of 4.12 weight percent. For operation during Cycle 14, the Pilgrim core was configured according to the loading pattern specified in the Fuel Loading Plan. Operation will be consistent with the thermal limits presented in the Supplemental Reload License Report. The MCPR operating limits are based on a Cycle 14 MCPR safety limit of 1.06 that is reduced from the current Technical Specification value of 1.08 used in Cycle 13. A separate license amendment was filed with the NRC for this Technical Specification change.

This change did not involve an unreviewed safety question. The Reload 13/Cycle 14 core design does not impair the ability of either the fuel assemblies or the pressure relief system to perform their respective design safety functions. All nuclear design criteria outlined in the FSAR and referenced in GESTAR (the generic fuel licensing document) have been met by the Cycle 14 core design. MCPR is calculated to always exceed Operating Limit MCPR and therefore prevent MCPR from falling below the Safety Limit MCPR. Additionally, the other thermal limits, i.e. MAPLHGR and MFLPD are met by Cycle 14 design with adequate margin. The previously analyzed transients do not violate thermal limits and design criteria, and have acceptable consequences using NRC approved methods described in GESTAR. The types of accidents and transients evaluated in the FSAR are described in Chapter 14. The dose consequences for the design basis accidents affected by Cycle 14 fuel are higher than previously reported; however, they remain significantly below the acceptance limits. No changes are proposed that adversely affect equipment reliability or equipment performance requirements or capabilities.

Freeze Seal of CRD Pipe Sections
Safety Evaluation: 3359

Three freeze seals will be installed in accordance with station procedure. The first two freeze seals involve the freeze sealing of 1 ½" schedule 80 stainless steel CRD flow control piping. The freeze seals are needed to repair the valves. The third freeze seal involves installation of a freeze seal on 1" schedule 80 stainless steel CRD pipe while the reactor is shut down and depressurized. The freeze seal is needed so that a valve can be replaced.

This activity did not involve an unreviewed safety question. The freeze seal procedure defines the necessary controls and processes for proper application of freeze seals. The procedure and NUMARC 91-06 guidelines provide controls and contingencies to ensure that there is an extremely low probability of a loss of CRD and reactor water inventory. The freeze seals of the CRD piping at the Flow Control Station and at the HCU area meet the requirements of these documents. Also, the integrity of the piping is not affected during and following the application of the freeze seal. This will be verified by a surface examination following removal of the freeze seal.

Use of Temporary Machine Shop in Reactor Auxiliary Building during RFO 13
Safety Evaluation: 3360

Machining equipment, such as a lathe and drill press, was installed in the water treatment area of the Reactor Auxiliary Building to support RFO 13 maintenance activities. A containment building with High Efficiency Particulate Air (HEPA) filters enclosed the equipment because it was used to machine contaminated items.

This change did not involve an unreviewed safety question. The evaluation concluded that the fire protection function for this area was not adversely impacted by the temporary installation of the machining equipment and its containment building. The evaluation assessed the physical and administrative controls and monitoring techniques established through the Radiation Work Process for work in the containment building. Containment and HEPA units are highly effective and reliable. Radiological burden of the items to be worked in the containment was assessed. This installation was temporary rather than a permanently installed, continuously operating, unmanned plant system. It was concluded that this TM would not result in releases to the environment in excess of 10 CFR 20 or PNPS Technical Specification limits. The effect of the equipment on floor loading was evaluated. The allowable floor live loading at elevation 23' of the reactor auxiliary building (water treatment area) adequately envelopes the added loading due to the weight of the equipment.

Modify Maintenance Requirements for Non-Safety Related 480V Breakers
Safety Evaluation: 3361

This change modifies the maintenance requirements on non-safety related 480 VAC Motor Control Center (MCC) breakers whose power cables run solely in non-safety related raceways, do not pass through primary containment penetrations and do not affect any fire hazards analyses. The change would provide the option to perform a comprehensive visual exam, including cycling in lieu of electrical testing on non-critical breakers. Additionally, the firm cap on the maximum grace period on scheduled maintenance activities on all electrical buses and breakers is being removed from the FSAR.

This change did not involve an unreviewed safety question. The changes to breaker maintenance practices are consistent with vendor recommendations that specify complete visual inspections and periodic cycling rather than electrical testing. Electrical testing involves removal of the MCC bucket and introduces the possibility for maintenance error and errors during return to service. Other industry groups have issued guidelines that are consistent with the change. A review of plant maintenance records was performed and no failures were found that are related to failure to trip due to an overload or fault. Problems are most likely to be discovered during cycling, not during electrical testing.

The maximum grace period was changed because there may be instances when nominal maintenance intervals including grace periods are exceeded. Time intervals were derived from industry practices and plant experience but are not absolute for maintaining an effective maintenance program. Extension of the grace period is expected to be rare and the recommended maintenance will not increase the possibility of a malfunction of equipment.

New Procedure for Installing Jumpers with Neutron Monitoring Motor Drive Modules Removed for Maintenance
Safety Evaluation: 3362

This procedure provides instructions to install jumpers in the Source Range Monitoring (SRM) and Intermediate Range Monitoring (IRM) systems to bypass the detector full-in permissive associated with the Reactor Manual Control System (RMCS) rod block withdrawal logic when the SRM and IRM motor drive modules are removed for maintenance. The jumpers simulate the detectors full-in position so that a rod block signal is not received.

This change did not involve an unreviewed safety question. The jumpers would not be installed unless the reactor mode switch was in the Refuel position and the rod block or RPS input is not required to be operable. Prior to installing the jumpers the detectors are verified to be full-in and control power to the drive motors is de-energized to ensure the detectors can not be moved from their position. Neutron monitoring is fully functional with the jumpers installed. No other rod block functions are affected. The procedure ensures the resulting plant configuration meets all Technical Specification requirements. The fire loading is not increased in the control room. If the jumper wire is dislodged or fails, a rod block signal is generated. This is a conservative action.

Temporary Power to Motor Control Centers B10 & B20
Safety Evaluation: 3363

This TM provided an alternate source of power to selected loads on MCCs B10 and B20 while planned Load Center B6 bus maintenance activities were performed during RFO 13. The normal source of power was isolated electrically and safety related jumper cables were connected from Bus B2 to the load side of the breakers that normally supply power to B10 and B20. The trip setting of the breaker feeding B10 and B20 was reset to match the values of the normal power supply breakers. The trip setting of Breaker 52-202 was reset to match the values on Breakers 52-603 and 52-604 to protect the jumper cables and maintain existing breaker coordination conditions. The TM contained specific instructions that ensured the original design configuration was restored when the temporary power feeds were removed.

This TM did not involve an unreviewed safety question. The TM was installed and then removed to provide power to B10 and B20 during the short duration when Load Center B6 was not available due to maintenance activities. Bus B2 was energized from the operable B Train power system credited at the time during which Bus B6 was out of service. The B Train Systems and equipment were operable to meet Technical Specification requirements. The procedure ensured other systems necessary to ensure key safety functions were available prior to de-energizing Bus B6. The TM was performed during outage conditions with the reactor in cold shutdown. The TM was not performed when primary containment was required to be operable. The plant accidents and transients applicable to outage conditions when this TM is in use are loss of offsite power, loss of shutdown cooling, refueling accidents and inadvertent vessel drain down. They were previously evaluated and installing the temporary power feeds does not increase their consequences or probability of occurrence. Technical Specifications, operating procedures, and outage risk assessments ensured that key safety functions were satisfied and that required equipment was available to maintain cold conditions.

125V DC Battery Discharge Test During Fuel Movement in RFO13
Safety Evaluation: 3364

During 125V DC battery discharge testing during plant cold conditions, the associated 125V DC bus was energized via a temporary battery configuration with a battery charger maintaining DC voltage on the bus. The temporary battery configuration consisted of a non-class 1E battery fed through a class1E, seismically mounted temporary panel. The DC bus is considered to be available but not operable.

The change did not involve an unreviewed safety question. The temporary panel provided separation between the non-class1E battery and the class1E 125V DC bus, effectively providing a method to ensure that cross-tieing safety related 125V DC buses cannot occur. The change had no effect on the safety analysis applicable during fuel movement. The refueling accident analysis assumes secondary containment isolation and operation of SGTS. All Technical Specification requirements were met. Radiation monitors are powered independent of DC power. Therefore, the Secondary Containment system will remain operable, capable of sensing a high refuel floor exhaust radiation signal, initiating SGTS, and isolating the reactor building in the event of a fuel handling accident.

Add HPCI Test Loop Adjustable Orifice Valve
Safety Evaluation: 3365

This modification replaced the HPCI test loop restricting orifice with a manually adjustable orifice valve. Prior to the modification the restricting orifice needed to be removed to perform HPCI pump testing with reactor pressure less than 150 psig. The system needed to be drained and the orifice reinstalled prior to increasing reactor pressure. This resulted in unnecessary startup delays. The new valve performs the identical function but is adjustable to simulate various system head conditions and thus avoids these delays.

This change did not involve an unreviewed safety question. There was no active safety function of the existing orifice. A safety related spacer was installed to provide Class 1 pressure boundary. The new valve resulted in a change in the location of pressure drops in the system but was evaluated to be acceptable. The new valve meets all relevant quality and design requirements for pressure boundary integrity for its location. The existing pipe supports have adequate margin to accommodate the increased loads. Surveillance test procedures were revised to reflect its use. The system operating modes, injection into the reactor vessel and reactor pressure control using the test line, were not adversely affected. The valve was pre-positioned at its desired pressure drop and locked in place.

Procedure Revision to Allow Removal of the Last Layer of Reactor Cavity Shield Plugs
Safety Evaluation: 3366

This procedure provides personnel with the information necessary for disassembly of the reactor pressure vessel. This revision permits the last layer of cavity shield plugs to be removed one hour following reactor shutdown with the reactor coolant temperature > 212°F. The purpose of this revision is to optimize outage performance.

This change did not involve an unreviewed safety question. This safety evaluation evaluates the impact of the change on the timing of the removal of the final layer of plugs. The specific activity of removing the shield plugs has already been analyzed and brought into compliance with the requirements of NUREG 0612. The probability of dropping a third layer shield plug onto the drywell head, however remote, has always existed. The evaluation addressed the impact on primary containment from such a heavy load drop while primary containment is required. A conservative and bounding finite element analysis concluded that the drywell head can withstand a drop of any of the third layer plugs from the maximum allowed lift height and primary containment integrity will be maintained. An analysis of the postulated radiological consequences that would result if the DB LOCA occurred with all three layers of shield plugs removed from above the drywell head concluded that the consequences (radiation dose) of the DB LOCA are not increased with the three layers of shield plugs removed.

Installation of Reactor Vessel Jet Pump Ultrasonic Testing Device
Safety Evaluation: 3367

This TP provides the administrative controls necessary for installation, removal and movement of the ultrasonic jet pump assessment apparatus. The tooling is lowered onto the reactor vessel flange and nondestructive examination of the jet pump internals is performed.

This TP did not involve an unreviewed safety question. This was essentially a maintenance activity. The weight of the apparatus (750 pounds) was well within the lifting capacities of the cranes employed and existing station procedures are used to position it on the flange. The flange was fully capable of carrying the weight. Inspection of the vessel studs and seating surfaces was required to ensure the vessel flange was not damaged. All materials in contact with the vessel water conform to the BWR materials handbook. Foreign material exclusion was considered and appropriate design controls implemented.

Simulation of All Control Rods Full In with Multiple Control Rods Removed During RFO 13
Safety Evaluation: 3368

The TM installed one jumper on each probe buffer card associated with control rod drive mechanisms or control rod blades that were replaced or worked during RFO 13. The jumper simulated that the affected control rod was at position 00 (full in) regardless of actual position. This allowed multiple control rods to be removed while maintaining the refueling interlocks operational for all non-affected control rods.

The TM did not involve an unreviewed safety question. Technical Specification Sections 3.10.A and 3.10.D specifically allow the change. T.S. 3.10.A specifies that the one-rod out interlock may be bypassed as required for those control rods or control rod drives to be removed after the fuel assemblies have been removed. The TM did not affect any control rods that were not being replaced and movement of these rods still provided input to the refueling interlock circuitry. Refueling interlocks are not required for control rods associated with control rod drive mechanisms that are being replaced because control rod movement within a core cell is not considered a core alteration provided that there are no fuel assemblies in the associated core cell. Removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. Station procedures ensure that all requirements of Tech. Spec. 3.10.A and 3.10.D are met prior to implementing the modification to each affected control rod. With no fuel in the affected core cell and no ability to load fuel into it, the potential for inadvertent criticality is precluded.

Changes to Maintenance Requirements for 4160V Switchgear and 480V Breakers
Safety Evaluation: 3369

The changes increased the interval between overhauls of 4160V switchgear and 480V load center breakers from once per eight years to once per twenty years. Breakers will undergo preventive maintenance every four years. Requirements for annual cycling of these breakers were eliminated. The number of breaker maintenance categories was consolidated to safety and non-safety related.

This change did not involve an unreviewed safety question. The change in overhaul schedules reflects the work of the Boiling Water Reactor Owner's Group (BWROG) Breaker Maintenance Committee. The service life of lubricant used in 4160V breakers was determined to be greater than twenty years. If overhaul frequency were increased to twenty years in conjunction with an inspection frequency of four and one-half years, then breaker reliability would not be affected. The same conclusions were made for 480V breakers. Previous recommendations from industry organizations for mechanical related failures had already been implemented. Industry data indicates that the majority of breaker failures were caused by maintenance errors and not by mechanical or lubrication failures. Increasing the interval will reduce the probability of human induced errors and increase breaker reliability. The new lubricant does not require cycling to extend its service life, so this requirement was eliminated. This also enabled the consolidation of breaker categories since the categories were based on which breakers could be cycled annually.

Install Lining in Salt Service Water Discharge Piping
Safety Evaluation: 3370

This modification installed a cured-in-place lining to the inside diameter of both loops of the SSW discharge piping. The liner is mainly comprised of an epoxy resin mixture that will prevent future deterioration of the SSW rubber liner and steel pipe. The lining does not provide a pressure boundary function, this function is still provided by the steel pipe.

This change did not involve an unreviewed safety question. The protective layer applied to the piping serves no safety function. The passive function of the liner is to protect the rubber lining and steel piping and provide unobstructed SSW discharge flow to the ocean. Hydraulic calculations show that even with the reduced pipe inner diameter caused by application of the lining, there in no adverse effect on the ability of the system to perform its safety functions. There is also no effect on flow testing criteria. Calculations were performed to determine the minimum required thickness based on external and internal pressure requirements. Allowances for piping external corrosion were made. The new liner exceeds the quality and design capabilities of the original rubber lining. Failures that could impact SSW flow are not expected over the life of the liner.

Temporary Procedure for Monitoring for Electrical Distribution System
Safety Evaluation: 3371

This TP monitors various points in the Electrical Distribution System (EDS) and provides data to verify results of the electrical load flow analysis. Monitoring of bus loading and voltage is required to validate a model of the AC distribution system. This analysis was performed to ensure that safety related electrical components are operating within their design ratings, under all expected operating conditions. Loading data was input into the model and the calculated output load flow was compared against the data collected by this TP. To accomplish these ends, various points in the 345KV switchyard, Startup and Unit Auxiliary Transformers, Main Generator, 4160V buses, 480V Load Centers and Motor Control Centers and Regulating Transformers were monitored.

This change did not involve an unreviewed safety question. The test equipment used was tested prior to its use to ensure it would have no adverse effect on the equipment they would monitor. The test equipment was isolated from ground and had very high input impedances. Recorders had isolated input modules, transducers acted as isolation devices, connections were secured with ring tongue connectors and personnel were trained to prevent shorting or opening of circuits. The TP did not alter the response characteristics of the EDS as designed to mitigate the consequences of design base accidents. The failure of the test equipment would have no adverse impact on the safety function provided by the EDS due to the precautions and controls applied. Test equipment power failure or electrical fault, or disengagement of test equipment connectors would have no adverse impact. The additional impact of a seismic event on test equipment is precluded by placing the test equipment on level floor areas to prevent interaction with safety-related panels.

Temporary Procedures to Verify Feedwater Level Control System Stability
Safety Evaluation: 3372

The feedwater regulating valves' internals were replaced in RFO 13. These TPs installed test probes and monitoring equipment on specific controller circuit locations to obtain operational data on control system performance during operator induced water level changes of up to six inches. This data was used to tune the control system as necessary.

These TPs did not involve an unreviewed safety question. The monitoring equipment and the test probes have high input impedance and have no impact on the feedwater control system when installed. If the test probes become disconnected and short together there will be no impact on the circuits monitored. A short to ground could cause a slight decrease in the loop signal. The impact on operation would be minor. The feedwater system response to plant transients analyzed in the FSAR is unaffected with the equipment installed. Power to the monitoring equipment is independent from other plant equipment. The monitoring equipment will be installed in the control room and precautions were taken to prevent any seismic interactions with control room devices.

Temporary Modification to Raise Alarm Set Points for "C" Safety Relief Valve Tailpipe Temperature
Safety Evaluation: 3334

This TM reconfigured the alarm circuitry in the SRV Temperature Recorder and the alarm setting for the computer data point. These changes served to eliminate continuous annunciation in the control room. This change was made due to leakage through the SRV pilot valve. The plant changes made are temporary and normal settings will be restored after SRV repair.

This change did not involve an unreviewed safety question. The proposed TM of the alarm set points does not involve any initiators associated with accidents described in the FSAR. The recorder and computer point affected by the proposed TM are not relied on to directly mitigate or provide operators information essential for the mitigation of any of the accidents evaluated in the FSAR. The recorder and EPIC computer point affected by the proposed TM provide information to operators on the current state of the SRV. The recorder also provides temperature trend information and alarms for all relief and safety valves. The proposed TM, by raising the alarm setpoint returns functionality of the alarms for each of the other safety and relief valves. The consequences of malfunction of equipment important to safety are not affected by this change. The changes to the tail pipe temperature measurement instrument loops do not affect the read out capability of the loops. Therefore, the loops are fully capable of providing their function of providing the operators with current tail pipe temperature information. The limiting conditions, surveillance requirements, bases, margins and required actions remain unchanged by the proposed alarm setpoint changes.

Procedure for Use of ShuffleWorks for In-Core Fuel shuffle During RFO 13
Safety Evaluation: 3374

ShuffleWorks is a computer code used to ensure that in in-core fuel movement meets reactor shutdown margin requirements of 1% at all times. The Technical Specification limit is 0.25%. This code is not available as safety-related so calculations were performed to benchmark results from ShuffleWorks against the results obtained from the fuel vendor's safety related code. The results from ShuffleWorks and administrative controls for its use during RFO 13 fuel shuffle formed the basis for this procedure.

This procedure did not involve an unreviewed safety question. The proposed procedure provides a conservative bias on calculations of shutdown margin by ShuffleWorks. This ensures that based upon a benchmark against a qualified, safety-related code shutdown margin is greater than 1% at all times for fuel shuffle during RFO 13. The analytical assumptions used in ShuffleWorks are cycle specific and include reload fuel types and fuel exposure. Use of the code ensures that the design limit of 1% shutdown margin is maintained.

Procedure Describing Decay Heat Removal Methods and Practices to be used during Refueling Outage 13
Safety Evaluation: 3375

This evaluation supports the decay heat removal methods and practices to be used during RFO 13. These activities will utilize various modes of operation of the RHR and Fuel Pool Cooling (FPC) systems during cold shutdown and refueling conditions to provide decay heat removal for the Reactor and Spent Fuel Pool (SFP). This evaluation also considers a temporarily larger off-load of additional fuel assemblies and the possibility for an unplanned full core off-load during the RFO. By using these considerations, this SE provides a bounding evaluation for RFO 13. An additional change included in this evaluation is a revision to the Augmented Fuel Pool Cooling (AFPC) Mode 1 operating configuration in station procedures.

This procedure did not involve an unreviewed safety question. The primary method of decay heat removal must have sufficient cooling capacity such that the bulk temperature in the SFP will be at or below a nominal 125°F with the peak not to exceed 142°F at the point of maximum heat load. Also, the minimum time to boiling without cooling must be greater than 6.4 hours. RFO 13 decay heat removal methods and practices described in the procedure are consistent with these requirements.

Replace 125V DC Breaker
Safety Evaluation: 3376

This modification replaced a 125V DC magnetic-trip only breaker with a new thermal magnetic breaker. The breaker was replaced as a precaution because it may not have met the test criteria and magnetic only breakers are no longer available.

This change did not involve an unreviewed safety question. The new breaker added a time-overcurrent protection feature that was not present with the original breaker. The breaker will now trip on overload fault conditions as well as on short circuit conditions. Breaker size was evaluated based on load at the feeder breaker. Breaker coordination characteristics were improved. The new breaker will improve DC system reliability. Certain overload faults can now be contained to the breaker leaving other systems and components available to perform their functions.

Outdated Pipe Break Data Deleted from FSAR Section 5.3.4
Safety Evaluation: 3377

Several paragraphs of FSAR Section 5.3.4 were deleted which contained outdated information concerning high-energy pipe breaks in the Reactor Building. The reader is now referred to FSAR Appendix O, which contains the current analysis of pipe breaks.

The change did not involve an unreviewed safety question. The steam release data in FSAR Section 5.3.4 had not been updated and did not represent the latest pipe break analyses. The current detailed steam release values are listed in the Subcompartment Modeling Data Book. Because the analysis and consequences of high energy line breaks are now discussed in Appendix O of the FSAR, the steam release rates were deleted from Section 5.3.4 and the reader is referred to Appendix O.

Isolate Turbine Building Heating and Ventilation when CRHEAFS is placed in Service
Safety Evaluation: 3378

The CRHEAFS operating procedure was revised to require isolation of the turbine building ventilation system supply and exhaust fans when CRHEAFS is in service during a design basis event. If a turbine building exhaust fan were to fail with the supply fans in operation while CRHEAFS is in service, pressure in the turbine building could be greater than the positive pressure in the control room pressure envelope. This could result in a condition whereby the control room would be at a negative pressure relative to an adjacent ventilation zone, resulting in infiltration of unfiltered air into the control room. The purpose of the procedure change was to preclude such a condition.

The change did not involve an unreviewed safety question. The procedure change does not impact any system needed for accident mitigation. New equipment or operating modes of safety-related equipment were not introduced. Isolating the turbine building ventilation requires new manual actions affecting non-safety-related equipment. These actions are performed in the same general area as those required for establishing CRHEAFS. Six switches need to be operated and can be performed easily by operations personnel. The operator would not receive any additional thyroid dose to perform these actions. The new actions are not compensatory actions for normally automatic actions. The response time to place CRHEAFS in its operating configuration is not affected by the proposed activity since the actions are not required to initiate CHREAFS and can be performed within pre-established time frames. The procedure change serves as a contingency to eliminate a potential source of unfiltered inleakage into the control room during a design basis accident.

Procedure to Connect Temporary Battery Charger to 125V "A" DC Bus During Battery Testing
Safety Evaluation: 3379

The125V DC bus has three possible sources of supply: a battery, a battery charger and a backup battery charger. This TP provided detailed instructions necessary to keep the 125V "A" DC system energized while the battery was being tested and both battery chargers were not available. A temporary battery charger and temporary battery was connected to the 125V DC bus. This allowed loads on the "A" side to be available but not operable. This work was performed in RFO 13 when the reactor cavity was flooded and fuel shuffle was in progress.

The change did not involve an unreviewed safety question. One battery was discharged tested at a time allowing the other battery and associated bus to be maintained operable throughout the duration of the test. Available, but not operable, equipment is not credited to meet minimum system operability requirements of Technical Specifications. Separation between 125 V DC divisions was maintained at all times. This modification did not adversely affect the ability of systems and components to respond to transients and accidents applicable to the plant configuration at the time of this evolution.

Revise HPCI and RCIC Test Procedures
Safety Evaluation: 3380

The HPCI and RCIC pump test procedures at less than 150 psig reactor pressure were revised to modify acceptance criteria and simplify the test procedures whereby testing of HPCI or RCIC can be performed at rated flow using the normal system lineup. Requirements to remove the test line orifices prior to testing at rated flow were eliminated from the procedures. An adjustable test line throttle valve installed during RFO13 replaced the test line orifice in the HPCI system. Regarding RCIC, a test was conducted during the last cycle during power ascension and it was demonstrated that RCIC could meet the test criteria with the test line orifice installed and reactor pressure under 150 psig. Based on the proposed procedure changes, orifices will no longer need to be removed and re-installed to allow testing at rated flow, and alternate testing at reduced flow will no longer be utilized.

This change did not involve an unreviewed safety question. The Technical Specifications require testing of HPCI and RCIC once per operating cycle with reactor pressure less than 150 psig. Each system can be tested by drawing water from the condensate storage tank and returning it through the installed test line. This change modified the system lineup used during these tests, but did not change the fundamental method used for these tests as described in the FSAR. FSAR changes were made to remove the description of orifice removal and replacement to enable testing at less than 150 psig reactor pressure.

Acceptance of Raised Fuel Support Piece in Control Cell
Safety Evaluation: 3381

During core verification from RFO 12, four fuel assemblies in a control cell were found to be elevated compared to other fuel assemblies. This evaluation assessed the impact of a raised fuel support piece in the cell for future operating cycles.

This change did not involve an unreviewed safety question. This safety evaluation addressed the following effects of an increase in bypass flow as a result of increased clearance between the guide tube and the fuel support piece and had no adverse effect on thermal limits design margin. The flutter of a LPRM string due to proximity to the increased bypass flow region and channel wear on surrounding bundles was shown by observation to be inconsequential. The effect of reduced in-channel flow on core flow distribution was determined to be insignificant because it was estimated to be well within the uncertainty of core physics calculations. The effect on MCPR was also determined to be insignificant due to the wide margin to the MCPR limit in this peripheral bundle. There is no impact on the ability to monitor the core using LPRMs because of the negligible effect of the bypass flow on neutron moderation. The fuel mechanical integrity of the bundle in the event of a seismic event was maintained. The orientation of the fuel support piece was shown to have no effect on the ability to move or scram the associated control rod. The evaluation concluded that there are no adverse safety consequences as a result of the raised fuel support piece.

Revision of Core Operating Limits Report for Cycle 14
Safety Evaluation: 3382

The Core Operating Limits Report (COLR) augments the Technical Specifications by describing the fuel operating limits that may vary from cycle to cycle based on cycle specific analyses performed by the fuel vendor. The COLR was revised to reflect the new core design that was evaluated in Safety Evaluation 3358. The COLR revision also includes a penalty factor of 3.2% on the Linear Heat Generation Rate (LHGR) limit and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) on GE 11 fuel. These penalties were necessary to accommodate errors discovered in General Electric's accident analysis codes.

This change did not involve an unreviewed safety question. The changes decrease the probability of fuel failure because decreasing LHGR and MAPLHGR limits the stored energy and decay heat in the fuel mitigating the rise in clad temperature after a LOCA. A reduction in maximum fuel power output would lower the calculated Peak Clad Temperature (PCT) following a LOCA. The change increases the margin of safety to the PCT limit of 2200° F.

Freeze Seal Section of Control Rod Drive Hydraulic Control Unit Piping
Safety Evaluation: 3383

A freeze seal will be installed in accordance with station procedure. The freeze seal involved a ¾ inch stainless steel pipe section on a CRD Hydraulic Control Unit scram outlet line to the scram discharge volume. The seal was necessary in order to replace a manually operated valve in the line. The seal was installed with the reactor shutdown and depressurized.

This activity did not involve an unreviewed safety question. The freeze seal procedure defines the necessary controls and processes for proper application of freeze seals. The procedure and NUMARC 91-06 guidelines provide controls and contingencies to ensure that there is an extremely low probability of a loss of CRD and reactor water inventory. The freeze met the requirements of these documents. The associated control rod was made inoperable by this seal but was acceptable because the rod was fully inserted. Also, the integrity of the piping is not affected during and following the application of the freeze seal. This will be verified by a surface examination following removal of the freeze seal.