10CFR50.59

BOSTON EDISON

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

In accordance with 10CFR50.59(b), Boston Edison is submitting this report of the changes, tests, and experiments at Pilgrim Nuclear Power Station for the period of January 1, 1992 through December 31, 1992.

A listing of reportable changes completed in the reporting period is attached. Each listing contains a brief description of the changes, a reference to the relevant Final Safety Analysis Report (FSAR) section, and a reference to the Safety Evaluation(s) that support each change.

No experiments were performed during the report period.

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MTL/nas/5059RPT

Attachment

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PLANT SECURITY AND YARD LIGHTING

PDC No.: 83-09
SE No.: 1559
FSAR Reference: Appendix P

A description of this change is not provided because it would contain Safeguards Information. The effects of the plant change in the FSAR - Appendix P "PNPS Security Plan" have been reported under 10CFR50.54.

POWER SUPPLY FOR PCI VALVES POSITION INDICATION

PDC No.: 85-31
SE No.: 1926
FSAR Reference: Figure 8.7-1

This design change upgraded the power supply for the Radwaste Collection and Drywell Equipment Sump isolation valves to the single failure criteria of Reg. Guide 1.97. Two redundant indication channels were provided for primary containment isolation valve position indication.

This change increased the reliability of the control room indicators for the system and did not involve an unreviewed safety question.

REACTOR WATER BACKUP GRAB SAMPLES IN RHR QUADRANTS

PDC No.: 87-47
SE No.: 2200
FSAR Reference: Figure 4.8-1

This modification provided additional grab sample lines from the RHR lines. These lines provide a "back-up" method of obtaining a reactor water sample if the primary sample lines are not in use or unavailable.

This change is not safety related in that the sample lines were attached outside of the safety related portion of the RHR system and therefore did not involve an unreviewed safety question.

FACILITY IMPROVEMENTS - BUILDING NO. 3

PDC No.: 89-36A
SE No.: 2467
FSAR Reference: Figure 8.4-4

This change is the first of a series to install a new security system. This particular change involved removal of the old "contractor gate house" and installation of the foundation for the new building to house the security computer. This work did not involve "Safeguards Information".

This change was "non-Q" and did not affect the operation of any safety system or safety equipment, therefore this change did not involve an unreviewed safety question.

PORTAL MONITOR REPLACEMENT AND ADDITIONS

PDC No.: 90-49
SE No.: 2490
FSAR Reference: 13.6.6.2

This change installed portal monitors at the Main Gate, Contractor Gate, and permanent fixed monitors at two locations in the Reactor Building. These changes were necessary because the previous monitors had limited capability for adjustments for background radiation and were unreliable due to the unavailability of spare parts.

The portal monitors do not affect any safety related systems or equipment, therefore, this change did not involve an unreviewed safety question.

DURALIFE 230 CONTROL BLADE REPLACEMENT

PDC No.: 90-61
SE No.: 2535
FSAR Reference: 3.4.5.1.1, 3.4.10, Figure 3.4-15

This change replaced original control blades (B4C) with General Electric model Duralife 230 control blades.

This modification did not involve an unreviewed safety question. The replacement blades (Model D230) are mechanically equivalent and are neutronically matched with the previous control blade design (B4C). Model D230 does not increase the probability of occurrence or consequences of an accident previously evaluated in the safety analysis. The accident analyzed in the FSAR is the rod drop insertion of reactivity. Although the relative worth was increased by 3%, this results in a very small increase in incremental rod worth of the limiting control rod in the core. The results of the rod drop accident in the FSAR are unaffected.

Analyses have been performed for control rod worths which bound the Duralife 230 control rod relative worth increases. Since the velocity limiter still functions identically as the original design and blades are matched (reactivity is within 5% $\Delta k/k$), control blade replacement with Model D230 does not increase the probability of - nor the consequences of malfunctions of equipment important to safety and does not create a potential for a new type of accident different from those already analyzed.

SECURITY: CCTV CAMERA IMPROVEMENT

PDC No.: 90-80
SE No.: 2585
FSAR Reference: Appendix P

A description of this change is not provided because it would contain Safeguards Information. The effects of this plant change on FSAR - Appendix P "PNPS Security Plan" have been reported under 10CFR50.54.

RHR VALVE ELECTRICAL INTERLOCK

PDC No.: 91-09
SE No.: 2605, 2552
FSAR Reference: 4.8.5.2, 4.8.5.4

INPO SOER 87-02 described several events involving improper valve lineups in the RHR system during shutdown and refueling modes of reactor operation. This modification installed electrical interlocks which are designed to improve system reliability and physically prevent RHR system valve misalignment.

This modification did not involve an unreviewed safety question because it did not alter the basic design of the RHR system and enhanced the safety and reliability of the system.

INCREASE OVERVOLTAGE TRIP SETPOINTS ON HPCI AND RCIC INVERTERS

PDC No.: 91-34
SE No.: 2606, 2634
FSAR Reference: Figure 8.6-1

This change increased the input overvoltage detector trip setpoints on the HPCI and RCIC Inverters from 140V dc to 150V dc to prevent the inverters from unnecessary tripping when larger 4kv ac motors are started.

This modification did not involve an unreviewed safety question as the availability of the inverters was improved. The inverter and its downstream loads will not experience any loss of function or significant degradation due to the infrequent voltage perturbations associated with larger motor starts.

RWCU PIPING REPLACEMENT

PDC No.: 91-39
SE No.: 2598
FSAR Reference: 4.9.3, 0.2, 0.6.4.1, Figure 0.6-22, Figure 0.6-24

This modification replaced piping in the Reactor Water Cleanup System between drywell penetration X-14 and the Cleanup Demineralizer Regenerative Heat Exchanger. The replacement piping is IGSCC Resistant Type 316L stainless steel.

This change did not involve an unreviewed safety question because it did not affect system operation, did not affect the function or configuration of the RWCU, and did not reduce the margin of safety associated with the RWCU system.

REACTOR WATER LEVEL REFERENCE LEG EQUALIZING LINE PIPING REPLACEMENT

PDC No.: 91-40
SE No.: 2603, 2609
FSAR Reference: Figure 7.8-2

This change increased the line size for Reactor Water Level equalizing line piping between Reactor Vessel Nozzle 15A(B) and Condensing Chambers 12A(B) and 13A(B) from the existing 1" NPS to 2" NPS. The reliability of the reactor water level measurement was enhanced and the safety-related functions of the Reactor Water Level Instrumentation piping and supports was ensured. This change did not involve an unreviewed safety question.

The changes did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety. The possibility of creating an accident or malfunction other than evaluated in the FSAR was not increased. The margin of safety, as defined in the basis for the technical specifications, was not impacted by this new installation because no technical specification was affected.

MODIFICATION FOR SAMPLING AUXILIARY HEATER BOILER AND RESIN STORAGE TANK

PDC No.: 91-52
SE No.: 2626
FSAR Reference: Figure 10.9-2, Figure 11.7-2

This modification made permanent, two sample sinks that were installed on a temporary basis for sampling the Auxiliary Heating Boiler outlet water and for sample lines associated with the Resin Storage Tank T-114.

Installation of these sample sinks did not involve an unreviewed safety question because no safety system or safety equipment is involved.

X55 and X56 TRANSFORMER REPLACEMENT

PDC No.: 91-59A
SE No.: 2706, 2676, 2664
FSAR Reference: 8.8.4, Figure 8.7-1

This change replaced two 480-120 volt distribution transformers to provide a greater margin of voltage between the manufacturer's minimum published rating and the available voltage from distribution panels Y3, Y4, Y31 and Y41. The new transformers are of the regulatory type and provide 120V AC $\pm$4% control power to the distribution panels.

This modification did not involve an unreviewed safety question because the replacement transformers increase reliability of the 120V AC power system by eliminating potential degraded voltage conditions on the 120V AC system.

SALT SERVICE WATER DISCHARGE HEADER PRESSURE SWITCH SETPOINT

PDC No.: 91-70
SE No.: 2648
FSAR Reference: 10.7.5

This modification reduced the setpoint for the Salt Service Water Discharge Header pressure switches. These pressure switches are required to start one Salt Service Water Pump in each loop with a Loss of Offsite Power and a Loss of Coolant Accident. The existing setpoint would have caused the start of two pumps per loop instead of one. Starting two pumps was unacceptable for diesel generator loading.

The modification did not involve are unreviewed safety question because it returned the Salt Service Water Pumps automatic start sequence to within the intent of the FSAR and the original design basis.

SECURED VENTILATION SYSTEM TEST FOR THE SWITCHGEAR, BATTERY AND VITAL MG SET ROOMS AT PILGRIM STATION

Temp Procedure No. 92-26
SE No.: 2665
FSAR Reference: none

The purpose of this test was to provide data on the effects of a loss of ventilation within the Switchgear areas, Battery Rooms and the vital MG Set Room. This information was used as an input to the determination of the operability of safe shutdown electrical equipment within the subject areas following a Loss of Coolant Accident with or without a Loss of Offsite Power.

This test did not involve an unreviewed safety question because the design maximum room temperatures were maintained - tests were limited to a maximum of 103°F which is less than the FSAR limit of 105°F.

CORE SPRAY PUMP DISCHARGE VALVES

SE No.: 2736
FSAR Reference: 7.4.3.4-4, Table 5.2-4, Table 7.3-1

The purpose of this change was to make the valve operating time in FSAR section 7.4.3.4.4 consistent with the analytical valve operating time used in the design basis analysis for ECCS performance as described in FSAR Section 6.5.4. Specifically, Reference 17 (see FSAR section 6.5.6) of this analysis indicates that 22 seconds is the maximum allowable stroke time used in the LOCA analysis for the low pressure core spray system injection valve (p. 4-6 of Reference 17). The term "pump discharge valves" is analogous to the term "injection valve" as used in the LOCA analysis described in Reference 17 of FSAR section 6.5.6.

This change increased the indicated design stroke time of the core spray pump discharge valve(s) from 18 or 20 to 22 seconds in the FSAR. The change makes this value consistent with the actual value used in the safety analysis for the core spray system as described in FSAR Section 6.5.4 and reference 17 of Section 6.5.6. This analysis was provided to the NRC in BECo Letter 2.90.154 dated December 10, 1990. The change did not involve an unreviewed safety question because it makes the valve stroke times consistent with the value used in the existing analysis.

EMERGENCY DIESEL GENERATOR DAY TANK FULL LOAD CAPACITY

SE No.: 2745
FSAR Reference: 8.5.2, Table 8.5-3

This change revised the diesel day tank "full load" capacity from 4 hours to a minimum of 2.5 hours. Also, estimates of actual tank volume were revised from 800 gallons to on the order of "greater than 600" gallons.

This change does not involve an unreviewed safety question because the emergency diesels are not affected nor are the bases for the level settings of the tank volumes affected.

RADIATION MONITORING AUXILIARY SAMPLING OF THE MAIN STACK AND REACTOR AUXILIARY EXHAUST VENT EFFLUENT

SE No.: 91-40
FSAR Reference: 7.12.3.3, 7.12.5.3

This change allows the use of auxiliary sampling equipment to perform required maintenance on the Main Stack and Reactor Building Vent Radiation Monitoring Systems or to provide backup monitoring capabilities if the Main Stack or Reactor Building Vent Monitoring Systems fail.

This change did not involve an unreviewed safety question because these systems do not mitigate the consequences of an accident, are not safety related and do not impact safety related systems.

GET LEVEL I AND VISITOR DOSIMETRY

SE No.: 92-23
FSAR Reference: 7.15, 13.6.6.2

This change removed the procedural requirement for GET Level I personnel and visitors to wear TLDs at PNPS by deleting the requirements to perform radiation dose monitoring which is not required by 10CFR20.

This change eliminates unnecessary use of dosimeters for dose monitoring which is not required by 10CFR20, does not affect any plant system or component, and therefore does not involve an unreviewed safety question.

ANNUAL REVISION TO THE PNPS EMERGENCY PLAN

SE No.: EP-92-01
FSAR Reference: Appendix N

This change involves the PNPS Emergency Plan only. These changes have been previously submitted to the NRC under 10CFR50.54q.