1978 ANNUAL OPERATING REPORT

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE UNIT #1

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Abstract: This report is submitted in compliance with Technical Specifications 6.9.1.5, 4.4.11.3 and 10CFR50.59.

INDEX

DESIGN CHANGES

On the following pages are descriptions, including a summary of the safety analyses, of the design changes implemented at St. Lucie Unit #1 during the period January 1, 1978 through December 31, 1978 in accordance with 10CFR50.59.

Plant Change/Modification 39-76

PSL Unit #1

"BORIC ACID HEAT TRACING POWER SUPPLY FEEDER"

The power supply for the Waste Management Heat Tracing Subsystem of the Boric Acid Heat Tracing System was modified so that it would automatically be re-energized from the essential, safety related system (MCC 1A-5) when powered by the 1A diesel generator. This modification prevents boric acid precipitation in the Waste Management Heat Tracing Subsystem due to loss of power.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Failure of the Waste Management Heat Tracing Subsystem has the same probability when automatically re-energized from the 1A diesel generator as compared to the previous manual re-energizing from the 1A diesel generator.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The addition of the automatic re-energizing feature will not produce an accident or malfunction that has not already been evaluated.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The diesel generator has sufficient reserve capacity to accept the additional automatic load imposed by this modification.

Plant Change/Modification 2-76

PSL Unit #1

"INSTRUMENT AIR COMPRESSOR LOADED ON EMERGENCY MCCs"

The power supplies for the turbine building instrument air compressors were relocated from the non-vital power distribution system (buses MCC 1A1 and 1B1) to the essential, safety related system (MCC 1A5 and 1B5). Also, the cooling system for both compressors was modified. These changes allow the instrument air compressors to be powered from the diesel generators with minimal operator action.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This design change is similar to other non-class loads on safety related MCCs. (The instrument air system is not nuclear safety related.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The loading of these air compressors is within the design rating of the diesel generators and is not automatic, but is controlled by the operator in the manual load group.

Plant Change/Modification 5-76

PSL Unit #1

"BORIC ACID MAKEUP VALVE STATION MODIFICATION"

Abandoned electrical boxes were removed and four cables rerouted at the valve station in the reactor auxiliary building. This was done to improve access for maintenance at the valve station.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. There was no change in function or quality of any system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 91-76

PSL Unit #1

"REACTOR VESSEL HEAD SHIELDING FOR REFUELING"

A review of the steady state radiation levels during refueling indicated approximately 10 Rem/Hr around the bottom edges of the reactor vessel head. A ring-shaped radiation shield consisting of structural steel and lead was fabricated and attached to the top of the missle shield. The reactor vessel head is placed on this ring-shaped structure during refueling to greatly reduce the local radiation levels in the vicinity of the reactor head.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The shield is intended as a biological barrier for personnel during refueling and thus is not described in the FSAR; therefore, it has no relation to equipment malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No safety related equipment or design features were functionally altered during the installation of the radiation shield.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 115-76

PSL Unit #1

"ADDITION OF TOTALIZER TO PLANT VENT STACK FLOW TRANSMITTER"

A totalizer (requiring square root extractor) was added to the Plant Vent Stack Flow Transmitter (FT-26-1). This totalizer will aid operators in determining the activity released through the stack.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is an addition to existing equipment and in no way affects the probability of accidents. Consequences of an accident are not increased. Better resolution of total air volume released is now available.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. This change is not nuclear safety related.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 136-76

PSL Unit #1

"PRIMARY SAMPLE SYSTEM VALVE REPLACEMENT"

Thirty-nine valves in the Primary Sample System were replaced due to their unsuitability for the particular application required for primary sampling. These valves were malfunctioning due to damage caused by boric acid crystalization. The replacement valves are Nupro "UG" series bellow valves (or equivalent).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The replacement valves are of stainless steel construction, have suitable pressure and temperature design ratings, and in general are better than or equal to the existing valves for the application.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The function of these 3/8 inch diameter sampling valves was not changed.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 171-76

PSL Unit #1

"D/G CONTROL PANEL GASKETS AND DEMISTERS"

This added gaskets and demisters on relays and doors of diesel generator 1B control panel in proximity of the air intake. This change is to prevent moisture intrusion which could cause deterioration of components.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This modification is not nuclear safety related. The externally mounted demisters and the gaskets increase the reliability of the control panel and therefore the diesel generator itself.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 199-76

PSL Unit #1

"CONTAINMENT FAN COOLERS VIBRATION ALARMS"

A remote reset feature and time delay function was installed on the high vibration switch annunciator for each of the four (4) containment fan coolers. Prior to this installation, the vibration switch would be activated while starting the fan coolers due to the momentary high vibration level and required entry into the containment building to manually reset the local annunciator reset button. This installation will reduce the number of entries into the containment building, thus reducing operator radiation exposure.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Analysis Report has not been increased. The vibration switch provides only an annunciation function. Thus functionally, the modification is not a safety concern.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report

Plant Change/Modification 201-76

PSL Unit #1

"AUXILIARY FEEDWATER PUMP TURBINE GOVERNOR CONTROL BOX RELOCATION"

The auxiliary feedwater pump 1C turbine governor control and central panels were relocated away from their original high moisture area (next to the steam turbine). This will prevent circuitry failure in the panels due to moisture ingression.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The only possible accident that could occur is failure of the steam driven auxiliary feed pump 1C. This has previously been evaluated in Table 10.5-1 of the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. Only two of three auxiliary feedwater pumps are required to meet the bases of technical specification 3.7.1.2.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 215-77

PSL Unit #1

"METRASCOPE INPUTS"

Due to the new surveillance in technical specification 4.1.3.1.3 requiring a functional test on the CEA block circuit, this modification was submitted to install sliding-link terminal blocks and a multi-pin connector to allow connecting the CEA position simulator to the metra cathode ray tube scanner and the oscilloscope scanner on all input channels while allowing the backup scanner to remain in service.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The affected systems are not directly safety related and are used for regular surveillance only.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 217-77

PSL Unit #1

"COMPONENT COOLING WATER HEAT EXCHANGER STRAINER COVER"

The strainer cover gasket and stud washers on the component cooling water heat exchanger strainer (Intake cooling water side) were replaced with better quality material. The previous asbestos gasket material leaked requiring torquing of the stud bolts (to stop the leak) to the point where the stud bolt threads might strip and the stud washers would bend. Replacement material is 1/8" thick ethylene propylene gasket. material and mild steel stud washers.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The replacement strainer cover gasket material and stud washers are of better quality than the previous material.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The only equipment malfunction that could result from this change is a minor leak at the strainer cover gasket. Table 9.2-2 of the Final Safety Analysis Report already analyses loss of a strainer.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 220-77

PSL Unit #1

"CONDENSER LEAKAGE DETECTION AND RESPONSE MODIFICATIONS"

The following changes were made to provide immediate detection and response to condenser tube leaks:

a. Relocated blowdown flow controller to the control room.
b. Relocated the condenser quadrant cation conductivity recorder to the control room and added an alarm.
c. Modified control circuitry for condenser discharge valves, vacuum breakers, and circulating water pumps, to provide rapid drain capability for the condenser.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related. It will provide increased protection against plant equipment damage which could be caused by salt water intrusion.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 231-77

PSL Unit #1

"SODIUM HYDROXIDE CONTAINMENT SPRAY ADDITIVE SUBSYSTEM"

This documented the installation of the sodium hydroxide additive subsystem for containment spray system to satisfy condition I.1 of the Facility Operating License. The new subsystem is designed to operate in conjunction with the containment spray system to remove radio-iodines from the containment atmosphere following the postulated LOCA. This system also replaces the function of the trisodium phosphate dodecahydrate storage baskets by providing containment sump ph control.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Refer to section 6.2.6 of the FSAR. Also refer to FPL letter to NRC, L-78-178 dated 5/19/78.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This change is considered in the FSAR and was a condition of the operating license.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 270-77

PSL Unit #1

"SHIELD BUILDING VENTILATION SYSTEM HEATERS"

This change involved the installation of auxiliary heaters in each train of the shield building ventilation system. This item resolved Condition of License I.2. These additional heaters provide humidity control to increase the effectiveness of methyl iodine removal by the filters.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. These auxiliary heaters are considered in the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The backfit requirement for these heaters is described in the FSAR.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 295-77

PSL Unit #1

"CONTAINMENT HYDROGEN SAMPLING VALVES REPLACEMENT"

A review of the purchase order for Unit I hydrogen sampling valves revealed that the installed valves had not been type tested for the post LOCA containment environment per FSAR Section 3.11 requirements. This PC/M documents the installation of replacement valves to correct the deficiency. (Reference LER #335-76-28).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The replacement valves meet or exceed the applicable seismic class I, nuclear safety class, and post LOCA requirements.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No functional or quality requirements were changed.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 300-77

PSL Unit #1

"SPENT FUEL STORAGE RACK MODIFICATION"

Modified spent fuel storage racks were installed to increase the spent fuel storage capability from 310 to 728 spent fuel assemblies. The increase in on-site storage capacity provided by this modification is required because of the lack of off-site spent fuel storage and/or reprocessing facilities.

A detailed description and evaluation of this change was given in the "Spent Fuel Storage Facility Modification Safety Analysis Report" which was submitted to the NRC on August 31, 1977 (FPL letter L-77-273).

Technical Specification Amendment No. 22 dated March 29, 1978, authorized use of the modified storage racks.

Plant Change/Modification 304-77

PSL Unit #1

"APPLICATION OF FLAMEMASTIC TO ELECTRICAL CABLES"

Flamemastic 71A, a sprayable fireproof coating was applied to all electrical cables installed in cable trays located in the reactor auxiliary building, containment building, and turbine building. This fireproofing material was applied to preclude electrically initiated fires, to prevent fire spread, and to limit cable damage due to a fire in the vicinity of cable trays.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is in accordance with the fire protection evaluation sent to the NRC by letter No. L-77-102 dated March 31, 1977.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 323-77

PSL Unit #1

"ADDITION OF STANDOFFS TO NEW FUEL ELEVATOR"

Standoffs were installed on the new fuel elevator to positively eliminate the possibility of placing a fuel assembly adjacent to a loaded new fuel elevator.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change enhances safety during fuel transfers.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification NNS-324-77

PSL Unit #1

"REFUELING MACHINE MODIFICATION"

The vendor of the refueling machine recommended modifications to the bridge and trolley sections to improve the reliability and function of the machine. Spacers were added between the wheels and bearings on the bridge and trolley. Also, angle bracing was added under the bridge drive shafts.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The modification to the refueling machine is a minor improvement to existing equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The modification to the refueling machine does not change the basic equipment design or function.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The modification to the refueling machine improves the operation of the equipment. No structural changes on the machine were performed.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 326-77

PSL Unit #1

"INSTALLATION OF NEUTRON STREAMING SHIELDING"

A reactor cavity neutron shield, consisting of nylon-neoprene covered bags holding ordinary light water, was installed. FP&L letter to NRC, L-76-406, dated November 29, 1976 describes the design and analysis for this change. This change satisfied Condition D of the Facility Operating License.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Refer to FP&L letters to the NRC, L-76-406 dated November 29, 1976 and L-77-245 dated April 3, 1977.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This was addressed in the correspondence referenced above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 327-77

PSL Unit #1

"RELOCATION OF STARTUP TRANSFORMER SECURITY FENCE"

This change moved a security fence to exclude the startup transformers from the Unit 1 security area, added separate fences around each transformer, and sealed or locked cable duct manholes. This will allow Unit 2 construction work in the area of these non-safety related transformers to be accomplished with adequate security for the transformers but without affecting Unit 1 overall security.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change does not affect nuclear safety related equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The new fencing, gates, and administrative controls meet existing levels of security requirements.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 330-77

PSL Unit #1

"SERVICE BUILDING WAREHOUSE SPRINKLER ALARMS"

Alarm circuitry was installed on the service building warehouse sprinkler system to produce local and remote (Control Room) alarms whenever the warehouse sprinkler system is activated. Also, an alarm will sound whenever the isolation valve between the warehouse sprinkler system and the fire main comes off its fully open position.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This modification does not involve safety related equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 332-78

PSL Unit #1

"ONE TON HOIST ADDITION TO REFUELING MACHINE"

A one ton hoist with monorail was fabricated and installed on the refueling machine. The hoist is used to handle tools and light equipment that would otherwise require the use of the polar crane. This would divert the polar crane from other required heavy lifting services and possibly extend a refueling outage as a result of reduced polar crane availability.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. A report issued by the refueling machine vendor indicated that the refueling machine structural integrity is adequate to accept the additional loads imposed on it by the one ton auxiliary hoist and its load under all conditions including a seismic event.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 335-78

PSL Unit #1

"CEA GROUP INTERLOCK BYPASS"

A CEDS internal wiring change was made to permit temporary bypass of automatic functions which inhibit the regulating groups from being withdrawn in the group mode of control when the shutdown CEA's are not at their fully withdrawn position. This allowed implementation of a technical specification change concerning repositioning CEA's to minimize guide tube wear. (Reference FPL letter to NRC, L-78-7, dated January 4, 1978).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 336-78

PSL Unit #1

"ADDITION OF ISOLATION VALVES FOR PRESSURE SWITCHES"

Isolation valves were installed in the sensing lines for several pressure switches for the diesel generators. This change allows for calibration of the pressure switches without taking the entire diesel generator out of service. Small diameter (1/4 inch) tubing and valves were involved in this change.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. There is no change in quality or reliability of the affected switches and tubing.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No functional change was made.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 339-78

PSL Unit #1

"INTAKE COOLING WATER PUMP REPAIR"

Because of pitting corrosion on the interior wetted surfaces of intake cooling water pump column sections, the pump manufacturer provided recommendations for cleaning, repair welding, and coating the affected areas.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The repair procedures and materials were consistent with the intake cooling water pump design and quality criteria.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. NNS-340-78

PSL Unit #1

"ADDITION OF QUICK EXHAUST VALVES TO TURBINE NON-RETURN VALVE ACTUATORS"

Six quick exhaust valves were installed between existing solenoids and non-return valve actuator cylinders. The quick exhaust valves allow for the conventional installation of the existing solenoids and provide for a rapid flow of air through the solenoids; thereby, decreasing the valve stroke times.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 373-78

PSL Unit #1

"INSTALL HARD PIPING FOR SHUTDOWN COOLING PURIFICATION"

Permanent piping was installed for the bypass of a portion of shutdown cooling flow through the letdown portion of the chemical volume and control system for purification. Previously, temporary hoses were connected for purification during shutdown cooling operation.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The design for the permanent shutdown purification piping is consistent with that of the interfacing systems.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. Paragraph 9.3.5.2.2 of the FSAR considers the installation of temporary piping in the CVCS to bypass a portion of the cooling flow during shutdown cooling through the letdown portion of CVCS for filtration and ion exchangers. This permanent design improves the method for implementation.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 375-78

PSL Unit #1

"REFUELING EQUIPMENT SPEED CONTROL MODIFICATION"

The motor drive panels for the refueling machine, spent fuel machine and control element assembly change machine were replaced with new motor drive panels containing improved motor drive controllers. The new motor drive controllers are more reliable, easier to adjust and have finer speed control capabilities.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This modification improves the reliability of the equipment modified.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There is no functional change involved with this repair. This modification is not nuclear safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 378-78

PSL Unit #1

"ADDITION OF NOISE SUPPRESSION DIODE FOR FCV-2161"

An arc suppression diode was installed across the solenoid wiring for FCV-2161 to alleviate random noise generated when closing the valve. FCV-2161 is a boric acid makeup flow control valve.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The new component does not decrease the quality of the electric circuitry involved. The reliability of FCV-2161 operation is enhanced.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There is no functional change involved.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 390-78

PSL Unit #1

"INSTALL INSULATION ON CCW PIPING"

Anti-sweat type insulation was installed on the component cooling water supply piping to the reactor coolant pumps to alleviate corrosion. Insulation was added to approximately 10 feet of piping near each pump.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The affected piping is not safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The new insulation is non-combustible.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 391-78

PSL Unit #1

"DISCHARGE CANAL MODIFICATIONS TO PREPARE FOR RAISING DIKE"

This change involved maintenance excavation work at the exterior slope of circulating water system discharge canal banks. A gravel drainage layer was constructed and some "dressing up" of the inside slope of the dike was performed where erosion had reshaped the slope. This work was done in preparation for raising the dikes (Reference PC/M 430-78).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 397-78

PSL Unit #1

"WORK STATIONS FOR CEA GUIDE TUBE MODIFICATIONS"

Temporary equipment was installed at the spent fuel pool to support modification of selected fuel assemblies. This PC/M documents the tools, equipment, and procedures used to implement the modification. (Reference PC/M #421-78).

This field modification is discussed in Section V. of CEN-90(F)-P which was submitted to the NRC in April, 1978.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 398-77

PSL Unit #1

"REMOVAL OF PART LENGTH CONTROL ELEMENT ASSEMBLIES"

The eight part length control element assemblies (PLCEA) were removed from the reactor and guide tube plugs were installed in the locations previously occupied by the PLCEAs. These plug assemblies preserve the dynamic operating characteristics of the reactor. A description of this change was provided to the NRC with FP&L letter No. L-78-125 dated April 12, 1978. Technical specification change to support removal of the PLCEAs was authorized in Amendment No. 27 dated May 26, 1978.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The removal of these PLCEAs has no effect on the physics characteristics of the reactor. Also, there is no significant change in thermal or hydraulic effects by the use of the plug assemblies to replace the PLCEAs.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The PLCEAs served no nuclear safety function.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The use of PLCEAs had been prohibited by the technical specifications, and they had to be locked in the full out position during all reactor operations. Thus, this change was consistent with the technical specification basis.

Plant Change/Modification No. 405-78

PSL Unit #1

"TEMPORARY REACTOR CAVITY FILTRATION SYSTEM"

This change provided for temporary tie-in of a reactor cavity filter assembly to improve water clarity in the cavity during refueling operations. (A permanent design for this change could not be implemented because of material availability problems. The permanent change is still planned.) The system was restored to its original configuration following refueling operations.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The design and quality of the flanges used to implement this temporary change were consistent with original system requirements.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modifiaction No. 406-78

PSL Unit #1

"NOISE REDUCTION ON INPUT TO ESF BISTABLES"

A filter capacitor was added to the detector loops to reduce noise input to the bistables in the engineered safety features actuation system.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change was recommended by the equipment vendor as a noise reduction improvement. New components are equal in quality to the originally supplied components.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 415-78

PSL Unit #1

"REPAIR FLOW RESTRICTION ORIFICE FOR INTAKE COOLING WATER SYSTEM"

Restriction flow orifice SO-21-1B was modified as a flat plate orifice instead of a plate with vortex breaker. The cantilevered extension vortex breaker was deteriorated. This orifice is located in the "B" intake cooling water header downstream of the CCW heat exchangers.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The function and flow characteristics of the orifice were not changed. An engineering evaluation showed that the vortex breaker (flow cone) is not needed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 418-78

PSL Unit #1

"MAIN STEAM CHECK VALVE MODIFICATION"

The main steam check valves disc backstop and shaft bearings were modified to reduce stress levels, thereby augmenting service life. Examination of the valve internals revealed the need to reduce wear.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Welding procedures and materials conform to applicable codes and specifications.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This modification is consistent with the original valve design criteria, and does not alter the function or operating characteristics of the main steam check valves.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 419-78

PSL Unit #1

"FUEL HANDLING EQUIPMENT CHANGES FOR CEA PLUG COMPATIBILITY"

The long spent fuel handling tool and the CEA change mechanism guide plate was modified. These minor hardware modifications were required to make the existing fuel and CEA handling equipment compatible with the CEA plugs. These CEA plugs were installed after eight part length CEA's were removed (See PC/M 398-77).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Only minor hardware modifications were performed. Changes to CEA handling equipment are non-nuclear safety related only.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. On the basis that the modification to the spent fuel handling tool meets the intended design requirements for material compatibility, strength, testing and inspection, this change does not constitute an unreviewed safety question.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 420-78

PSL Unit #1

"INCORE INSTRUMENT THIMBLE REPAIR"

This documents repair of three (3) incore flux detector thimbles (protective tubes). Replacement of #10 coarse threads with #16 UNF threads, and tack welding of the collar to the thimbles reduces the likelihood of future failures. This item was reported to the NRC in Licensee Event Report 335-78-11 Update Report dated August 8, 1978.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The method of repair increased the reliability of the thimbles. (These thimbles are not pressure boundaries; they serve as "bushings" to support the small, flexible incore assemblies within the larger guide tubes).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There is no functional change involved with this repair. This change is not nuclear safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 421-78

PSL Unit #1

"RETURN TO OPERATION WITH CEA GUIDE TUBE SLEEVES INSTALLED"

This PC/M documents the reviews and approvals for reactor operation with selected fuel assemblies modified with control element assembly guide tube sleeves. (Implementation of the sleeving modification is covered in PC/M 397-78). The stainless steel sleeves are needed to alleviate a CEA guide tube wear problem reported in Licensee Event Report 335-78-12 dated April 28, 1978.

A detailed report was provided in CEN-90 (F) - P, "St. Lucie Unit 1 Reactor Operation with Modified CEA Guide Tubes", submitted to NRC in April, 1978.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Section VI of CEN-90 (F)-P addresses this concern.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. Section VI.C of CEN-90 (F)-P addresses this.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. Section VI.D of CEN-90 (F)-P addresses this.

Plant Change/Modification No. 429-78

PSL Unit #1

"TURTLE PROTECTION NET"

A protective net was installed across the ocean intake canal to keep entrapped turtles away from the plant intake screens where they might be harmed, and to confine the entrapped turtles to a small area which will facilitate their removal.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 432-78

PSL Unit #1

"MODIFICATIONS TO OVERPRESSURE MITIGATING SYSTEM"

The relief setpoint for the power operated relief valve in the overpressure mitigating system was changed from a "sliding" setpoint for various temperatures to a constant setpoint of 465 psia for all temperatures. This modification was requested by the NRC.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This modification was a requirement of the NRC.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 433-78

PSL Unit #1

"INTAKE COOLING WATER STRAINER CHANGE"

This changed the bodies of the intake cooling pumps lube water strainers from carbon steel to type 316 stainless steel. This change was made to enhance corrosion resistance.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. There is no change in function or quality of the strainers associated with this modification.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased by this material change.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 446-78

PSL Unit #1

"RPS TM/LP PRETRIP SETPOINT"

The hardware in the Core Protection Calculator #1, TM/LP Pretrip Circuit was modified to change the TM/LP Pretrip setpoint from 100 pounds above the trip setpoint to 50 pounds above the trip setpoint. This was done to eliminate the numerous alarms which were occurring.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Neither the TM/LP Pretrip Circuit nor any device to which it sends a signal is safety related or taken credit for in the safety analysis.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The TM/LP trip setpoint has not been changed.

Plant Change/Modification No. 453-78

PSL Unit #1

"MODIFY PRESSURIZER RELIEF VALVES DISCHARGE PIPE VACUUM BREAKER"

A one-half inch diameter check valve was installed to replace the one-half inch diameter hole to serve as a vacuum breaker in line 10-RC-822. This piping vacuum breaker is located inside the pressurizer quench tank above the normal water level. The hole allowed steam to flow into the tank above the quenching water. The new check valve forces the leakage steam into the water, but still provides the vacuum breaker function.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related. The check valve will allow for improved cooling of steam leakage from the safety relief valves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The vacuum breaker function was not changed.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 455-78

PSL Unit #1

"DIESEL GENERATOR LOSS OF FIELD TRIP CIRCUIT MODIFICATION"

The emergency diesel generator loss of field trip circuitry was revised by defeating S2 lockout relay contacts 12 and 12C to eliminate loss of field relay chatter.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The loss of field trip circuit does not affect any accident evaluated in the FSAR. (The loss of field relay is taken out of the lockout circuit under accident conditions, including loss of offsite power.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This is a nonfunctional change made to increase the reliability of the relay.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 493-78

PSL Unit #1

"SI TANK FILL LINE BRACE MODIFICATION"

A 3 inch by 3 inch angle iron pipe brace for line 1-SI-123 was modified by using bolted flanges instead of welded construction. This facilitates removal of the brace, which is required periodically for entry to a steam generator manway.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The design load bearing capability of the brace was maintained.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No functional change is involved.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

The following summarizes changes in procedures and special tests conducted per 10CFR50.59 during the period January 1, 1978 through December 31, 1978:

PROCEDURE CHANGES

QI-3, PI/PSL-1, Design Control

This procedure was revised to allow in-plant approval of design changes that are not nuclear safety related. This was needed to avoid the time delay and expense of processing minor, non-safety related changes through the home office power plant engineering department. The method for processing nuclear safety related changes was not revised. Also, the method of control of design document updating for all changes was not affected.

This procedure revision affects only non-nuclear safety related design changes. (The depth of review to determine if a change is nuclear safety related or not was not lessened.) This procedure change does not constitute an unreviewed safety question.

SPECIAL TESTS

Low Power Feedwater Control System Test, Letter of Instruction T-06.

This test was performed to document the dynamic behavior of the steam generators to small level perturbations in the low power range, and also to verify the response of a proposed control system to changes in settings. The proposed system, which was designed by the NSSS Vendor, is to provide an automatic mode of control at low power levels to improve plant availability by minimizing reactor trips caused by steam generator level oscillations at low powers.

The test was conducted at low reactor power levels (less than 15%). Procedural controls were specified to keep operations within normal plant operating limits. The test equipment has been removed. This test was determined not to involve an unreviewed safety question. The test data is being evaluated by the NSSS Vendor to determine if the proposed system will increase the reliability of the low power feedwater control operations.

CORE BARREL MOVEMENT

Section 4.4.11.3 of PSL #1 Technical Specifications requires the results of all periodic Amplitude Probability Distribution (APD) and Spectral Analysis (SA) monitoring to be included in this report.

Routine monitoring in 1978 included weekly APD processing and SA processing was done in February, June and October. SA measurements in June included analysis at nominal thermal power levels of 20%, 50%, 80% and 100% at the beginning of fuel cycle 2. At no time were the Alert or Action levels exceeded.

As previously observed and reported in 1977 the RMS levels of all excore neutron detector signals continued to show a gradual increase throughout 1978 with the exception of a slight decrease following refueling.

The increase, confirmed by independent APD and SA analysis, amounts to about 50% above the levels of December 1977. As in 1977 the greatest portion of signal increase was in the frequency range of 1-4 hertz. This phenomenon has been observed at other PWR's and has been attributed to the increase in fuel element vibration worth due to the decreasing boron concentration with core burnup. Likewise the observed step decrease in the band of 1-4 hertz following refueling may be attributed to the step increase in boron concentration at the beginning of the new fuel cycle.

The observed net increase for 1978, if imputed to be entirely due to core barrel motion would signify less than 4 mils RMS motion at the core midplane.

"STEAM GENERATOR TUBE INSPECTIONS"

An inservice eddy current examination of selected tubes in the No. 1A St. Lucie Unit No. 1 Steam Generator was performed during the period of April 2 through April 7, 1978, by C-E Power Systems, Systems Integrity Services personnel. The inspection was conducted in accordance with C-E Test Procedures Nos. 00000-ESS-105, Revision 00 and 00000-ESS-070, Revision 01, and satisfied the requirements of the St. Lucie Plant Technical Specification 3/4 4-5 and the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition through the Summer 1976 Addenda.

The inspection program consisted of 400kHz testing for tube wall anomalies and 25kHz testing for sludge accumulation on the secondary side of the tube sheet. Table I details the number of tubes examined in each generator. Selection of tubes to be examined was based on an evaluation of data taken previously in other steam generators in service. Additionally, when requested by the Data analyst, certain tubes were re-examined at 100kHz for confirmation of the 400kHz data.

The data from the inspection was recorded on magnetic recording tape and strip charts. These recordings were evaluated and the results recorded on Eddy Current Examination Report Sheets. Of the tubes examined, none were found to have reportable wall degradation. (>20% of wall). No tubes were plugged.

A number of tubes were found to be dented at the point where the tube passes through one of the drilled hole support plates. A summary of the numbers and magnitude of these dents is included as Table I.

The 25kHz inspection indicated up to 4" of sludge on the Hot Side and up to 3.5" of sludge on the Cold Side of the secondary side of the tube sheet.

Information on the results of this first Unit 1 steam generator inservice inspection were reported to NRC in FP&L letter No. L-78-249 dated July 31, 1978. In addition, more detailed information is available at the plant site.

TABLE I

SUMMARY OF EDDY CURRENT TEST RESULTS

STEAM GENERATOR 1A

INSPECTION CONDUCTED APRIL 1978

TOTAL TUBES (by design)	8,519 'U' Tubes
TUBES THROUGH PARTIAL SUPPORT PLATE No. 9	2,225 (26.1%)
TUBES THROUGH PARTIAL SUPPORT PLATE No. 10	771 (9.1%)

TUBES EXAMINED

	No. of Tubes	% of Total	
HOT SIDE DEFECT DETECTION	583	6.8%	No Tube Wall Degradation Indications
COLD SIDE DEFECT DETECTION	100	1.2%	No Tube Wall Degradation Indications
HOT SIDE SLUDGE MEASUREMENT	62	0.73%	Maximum 4 Inches
COLD SIDE SLUDGE MEASUREMENT	62	0.73%	Maximum 3.5 Inches

APPENDIX A
STANDARD FORMAT FOR REPORTING NUMBER OF PERSONNEL AND MAN-REM BY WORK AND JOB FUNCTION

Work & Job Function	Number of Personnel (> 100 mrem)			Total Man-Rem		
	Station Employees	Utility Employees	Contract Workers and Others	Station Employees	Utility Employees	Contract Workers and Others
Reactor Operations & Surveillance						
Maintenance Personnel	0	0	0	0	0	0
Operating Personnel	21	0	0	8.200	0	0
Health Physics Personnel	7	0	0	3.370	0	0
Supervisory Personnel	2	0	0	1.680	0	0
Engineering Personnel	0	0	0	0	0	0
Routine Maintenance						
Maintenance Personnel	96	39	138	43.430	18.310	55.220
Operating Personnel	11	0	0	4.330	0	0
Health Physics Personnel	15	0	3	6.690	0	.740
Supervisory Personnel	8	1	1	2.190	.140	.120
Engineering Personnel	3	0	3	1.120	0	.460
Inservice Inspection						
Maintenance Personnel	4	5	3	1.590	2.510	.660
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	3	0	6	.820	0	1.690
Supervisory Personnel	6	0	0	4.050	0	0
Engineering Personnel	2	0	49	.740	0	27.700
Special Maintenance						
Maintenance Personnel	0	0	59	0	0	25.200
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	1	0	0	.150	0	0
Supervisory Personnel	0	1	2	0	.120	.770
Engineering Personnel	2	0	2	.330	0	.450
Waste Processing						
Maintenance Personnel	15	0	1	4.580	0	.140
Operating Personnel	11	0	0	3.740	0	0
Health Physics Personnel	6	0	0	2.060	0	0
Supervisory Personnel	1	0	0	.620	0	0
Engineering Personnel	0	0	0	0	0	0
Refueling						
Maintenance Personnel	38	53	0	17.960	38.430	0
Operating Personnel	10	0	0	2.120	0	0
Health Physics Personnel	4	0	20	1.410	0	8.350
Supervisory Personnel	6	0	1	1.810	0	.110
Engineering Personnel	1	0	2	.510	0	.720
TOTAL						
Maintenance Personnel	105	84	191	67.560	59.250	81.220
Operating Personnel	31	0	0	18.390	0	0
Health Physics Personnel	15	0	28	14.500	0	10.780
Supervisory Personnel	16	2	4	10.350	.260	1.000
Engineering Personnel	4	0	56	2.700	0	29.330
Grand Total	171	86	279	113.500	59.510	122.330

1977 ANNUAL OPERATING REPORT

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE UNIT #1

February, 1978

INDEX

SUMMARY OF OPERATING EXPERIENCE

The following is a summary of plant operations including pertinent items of interest chronologically for the period 1/1/77 thru 12/31/77.

1/1/77			at 50% power. (Power Ascension Testing)
1/3/77		3:15 PM	Rx trip. Loss of "B" S/G feed pump.
1/4/77		3:40 AM	Rx critical.
1/5/77		12:07 AM	Unit on line-increasing power to 50%.
1/9/77		11:50 AM	Increasing power to 60%.
1/10/77		5:45 AM	Increasing power to 70%.
1/11/77		1:00 PM	Increasing power to 80%.
1/24/77		2:00 PM	Decreasing power to 50% to clean condensate pump strainers.
1/25/77		6:52 AM	Increasing power to 80%.
2/1/77	(1)	10:48 AM	Planned reactor trip. Partial loss of flow test.
	(2)	11:48 AM	Rx critical.
	(3)	11:13 PM	Unit on line-increasing power to 40%.
2/2/77	(1)	4:04 AM	Planned Rx trip. Total loss of flow test.
	(2)	7:30 PM	Rx critical.
2/3/77	(1)	00:03 AM	Unit on line-increasing power to 20%.
	(2)	3:05 AM	Planned Rx trip-loss of offsite power.
	(3)	7:53 AM	Unit on line-increasing power to 90%.
2/4/77	(1)	11:15 AM	Dropped rod #60-reduced power to 50%. Recovered rod-increasing power to 90%.
	(2)	9:00 PM	Rx at ≈90% power.
2/20/77	(1)	10:15 AM	Planned Rx trip-manual turbine trip at 100% power.
	(2)	10:55 PM	Rx critical.

SUMMARY OF OPERATING EXPERIENCE

2/21/77	(1)	12:05 AM	Unit on line.
	(4)	2:25 PM	Planned Rx trip. Manual loss of load at 60% power.
2/22/77	(1)	12:16 AM	Rx critical.
	(2)	8:47 AM	Unit on line-increasing power to 100%.
3/4/77		8:15 PM	Reduce power to 50% to clean 1B S/G feed pump strainers.
3/5/77		1:30 PM	1B S/G feed pump back in service-increasing power to 100%.
3/7/77	(1)	2:04 AM	Discovered CEA #20 stuck at 125" while performing FLCEA periodic test-decreasing power to 70%.
	(2)	2:50 AM	Dropped rod #23 while aligning it with Rod #20-decreasing power to 60%.
	(3)	4:15 AM	Rods #20 & #23 repaired-increasing power to 100%.
3/11/77	(1)	5:00 PM	Containment emergency air lock interior door failed leak rate test-reducing power to enter containment and inspect.
	(2)	10:04 PM	Unit off line to replace airlock gasket.
3/12/77	(1)	6:38 AM	Unit on line-increasing power to 30%.
	(2)	5:30 PM	Completed load swing test at 30%-increasing power to 50%.
3/18/77		2:45 PM	Completed load swing test at 50% - increasing power to 90%.
3/19/77		6:40 PM	Completed load swing test at 90% - increasing power to 100%.
4/3/77		6:08 AM	Rx manual trip-fire in generator lead box. Fire caused by H_2 leak on generator lead box seal. Seal replaced.
4/11/77	(1)	7:04 AM	Rx critical.
	(2)	2:09 AM	Unit on line-increasing power to 100%.
4/15/77		3:39 AM	Manual Rx trip due to loss of cooling water to RCP seals caused by loss of containment air compressor. See PCM 258-77.

SUMMARY OF OPERATING EXPERIENCE

4/29/77	(1)	11:40 AM	Rx critical.
	(2)	1:10 PM	Unit on line-increasing power to 100%.
4/30/77	(1)	1:54 AM	Channel "A" pressurizer pressure transmitter failed high.
	(2)	3:18 AM	Started reducing power to 12% to make containment entry to change transmitter.
	(3)	7:30 AM	Replacement of transmitter completed-increasing power to 100%.
4/30/77	(1)	1:04 PM	Reactor tripped on loss of load. Indication in Control Room of loss of voltage regulator. Operator opened exciter field breaker. Indication caused by failure of B heater drain pump L.C.V. which caused extreme vibration in discharge line. Line is near the north wall of turbine building switchgear room and vibration caused the 6.9 KV undervoltage relays to trip. RCP's tripped.
	(2)	3:30 PM	Reactor critical.
	(3)	5:42 PM	Unit on line-increasing power to 100%.
5/10/77	(1)	6:25 PM	Started reducing power to take unit off line due to small leak on 1A2 S.I.T. line.
	(2)	9:50 PM	Unit off line.
5/12/77		4:15 PM	Maintenance completed repair of 1A2 S.I.T. drain valve line.
5/13/77	(1)	7:58 AM	Rx critical.
	(2)	1:23 PM	Unit on line-increasing power to 100%.
5/16/77	(1)	10:23 AM	Rx trip due to system voltage fluctuations.
	(2)	8:04 PM	Rx critical.
	(3)	9:58 PM	Unit on line-increasing power to 100%.
5/19/77		9:30 PM	Reducing power to 50% to clean steam generator feed pumps 1A & 1B suction strainers.
5/20/77		5:55 PM	Started increasing power to 100%.
5/27/77	(1)	3:10 PM	Dropped rod #39-turbine runback to ≈ 70%. 1A SGFP tripped. Rx tripped on low S/G level.
	(2)	6:28 PM	Rx critical.
	(3)	8:13 PM	Unit on line-increasing power to 100%.

SUMMARY OF OPERATING EXPERIENCE

5/31/77		3:40 PM	Turbine/reactor trip-loss of generator excitation caused by failure of permanent magnet generator. Unit shutdown to replace permanent magnetic generator (PMG).
6/5/77	(1)	11:13 AM	Reactor critical.
	(2)	8:38 PM	Unit on line-increasing power to 100%.
6/7/77	(1)	4:16 PM	Reactor trip-operator error. Tripped operating motor generator set.
	(2)	6:00 PM	Reactor critical.
	(3)	8:04 PM	Unit on line-increasing power to 100%.
6/13/77	(1)	5:04 AM	Reduce load to ≈50% to replace 1B turbine cooling water pump seal.
	(2)	11:59 PM	Seal repaired-increasing power to 100%.
6/25/77		10:39 PM	"A" condensate pump expansion joint failed, A condensate pump tripped, A SGFP tripped on low suction, Rx tripped on low S/G level. Replaced expansion joint.
6/26/77	(1)	2:10 AM	Rx critical.
	(2)	1:17 PM	Unit on line-increasing power to 100%.
6/27/77		10:20 PM	Reducing power to come off line to isolate PCV 1100F. (Pressurizer Spray Valve)
6/28/77		3:40 PM	Unit back on line-increasing power to 100%.
7/4/77		10:18 PM	Reducing power to come off line to repair PCV 1100F.
7/5/77		8:00 AM	Unit back on line-increasing power to 100%.
7/8/77	(1)	2:30 AM	Reducing power to come off line to repair PCV's 1100 F & E.
	(2)	6:41 AM	Rx shutdown.
	(3)	3:23 PM	Rx critical.
	(4)	4:52 PM	Unit on line-increasing power to 100%.
7/10/77		5:52 PM	Reducing power to come off line to repair PCV 1100F.
7/11/77		5:59 AM	Unit back on line-increasing power to 100%.
7/12/77		8:56 PM	Reducing power to come off line to repair PCV 1100 E & F.

SUMMARY OF OPERATING EXPERIENCE

7/13/77		6:54 AM	On line-increasing power to 100%.
7/29/77		5:25 PM	Reducing power to come off line to repair PCV 1100 E & F. (See PCM 268-77)
7/30/77		2:18 PM	Unit back on line-increasing power to 100%.
8/30/77		8:45 PM	Reducing power to 50% to work on 1A feedwater pump.
8/31/77	(1)	1:11 PM	Rx tripped on low S/G level. 1B S/G feedwater pump tripped when 1A S/G feedwater pump start was attempted.
	(2)	3:11 PM	Rx critical.
	(3)	4:47 PM	On line-increasing power to 100%.
9/24/77		6:25 PM	Reducing load to take unit off line for planned outage. See PCM 264-77.
10/10/77	(1)	5:40 AM	Rx critical.
	(2)	11:59 AM	Unit on line-increasing power to 100%.
10/28/77	(1)	12:45 PM	Dropped Rod #56.
	(2)	1:30 PM	Rx power <70%.
	(3)	2:38 PM	Rod #56 retrieved.
	(4)	6:00 PM	Due to flux tilt maintaining power at ≈ 75%.
10/30/77	(1)	1:55 AM	Reactor Engineering calculated flux tilt to be within Tech Spec limits-increasing power to 100%.
11/17/77		7:05 PM	Reducing power to 50% to clean 1A SGFP strainer.
11/18/77		6:50 AM	Increasing power to 100%.
11/22/77	(1)	10:32 AM	Rx trip due to loss of 1A SGFP.
	(2)	12:40 PM	Rx critical.
	(3)	2:27 PM	Unit on line-increasing power to 100%.
12/20/77	(1)	5:31 AM	Rx trip-loss of load caused by loss of excitation.*
	(2)	1:49 PM	Rx critical.
	(3)	3:59 PM	Unit on line-increasing power to 100%.
12/31/77			at 100% power.

*Cause later found to be intermittent open in fuse in excitation circuit.

DESIGN CHANGES

On the following pages are descriptions, including a summary of the safety analyses, of the design changes implemented at St. Lucie Unit #1 during the period January 1, 1977 through December 31, 1977.

Plant Change/Modification 4-76

Unit #1

"INSTALL ACCESS PLATFORMS AT PRESSURIZER CUBICLE"

Access platforms were installed at the pressurizer cubicle to facilitate maintenance and operation of valves and instruments near the top of the cubicle.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The platforms are not required for safe shutdown of the plant.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new platforms were designed and installed to Seismic Class I requirements to preclude failure as a result of a postulated earthquake.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 37-76

Unit #1

"STEAM GENERATOR HI WATER LEVEL TURBINE TRIP CIRCUIT"

Control circuitry was added to provide a trip signal on steam generator hi water level to prevent the possibility of water carry over to the steam turbine.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Where safety related components were replaced (level indicator controllers), devices of the same basic type and equivalent qualification were used. The low water level trip features were not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. This modification does not affect any technical specification basis.

Plant Change/Modification No. 83-76 Unit #1

"RELOCATION OF 4.16 KV AND 6.9 KV SWITCHGEAR RELAYS"

Hinged armature relays were originally mounted on the 6.9 KV and 4.16 KV switchgear door panels in such a manner that inadvertent relay contact actuations could possibly result from slamming closed the switchgear doors. To prevent this, those relays with control or trip functions were relocated to the side walls of the breaker cubicles or to separate boxes outside the switchgear cubicles. Approximately 51 Westinghouse type "SG" and 6 General Electric type "HGA" relays were relocated.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Replacement materials meet or exceed the requirements of the original components. The reliability of the subject relays is increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This modification does not change the function of any safety related equipment.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 86-76 Unit #1

"BUTTERFLY VALVES CONTROLS MODIFICATION"

The control circuitry for 31 motor operated valves was revised to utilize a limit switch rather than a torque switch to stop the motor operator in the valve closed position. This change conforms to valve vendor recommendations that rubber seated butterfly valves be position seated instead of torque seated. The torque switch, which actuates when the operator reaches its mechanical stop, was retained as a backup to the limit switch control.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The new limit switches will stop the valves at the same relative position as the original torque switches. To provide a backup to control valve closure, the torque switches were retained.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There is no change in control scheme involved. The use of a limit switch with torque switch backup for control will increase the reliability of each valve closure operation and reduce operator maintenance.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 108-76 PSL Unit #1

"INSTALL SPARE 5 KV CABLES"

These cables were installed to meet Technical Specification 4.8.1.1.3.b.2 which requires three (3) spare cables for insulation resistance tests.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   These cables, which will not be used for plant equipment, are designed and installed to the same requirements as the original cables. They allow testing required by the NRC to demonstrate the satisfactory condition of the Class 1E underground cable system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change is required by the Technical Specifications.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 169-76 PSL Unit 1

"EQUIPMENT DRAIN TANK 1A STRAINER MODIFICATION"

The single element equipment drain tank strainer (S6904) was replaced with a dual basket type strainer with a differential pressure alarm. This change allows the cleaning of a clogged strainer without interrupting the ability to pump into the equipment drain tank from the reactor cavity sump or the engineered safeguards room sump.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related. New components are compatible with the original systems affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This change provides operational flexibility without a change of function.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 170-76 Unit #1

"MODIFICATION TO REACTOR CAVITY SUMP LEAK DETECTION SYSTEM"

This change involved replacement of 1 inch tubing from collection boxes to wier tank with stainless steel gutters, modification of the weir tank, and relocation of a flow transmitter to outside of the sump. These changes were made because of problems experienced with maintenance and calibration of the reactor cavity sump leak detection system.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Replacement components were designed and installed to equivalent or better standards as the originals.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There is no change in overall function or quality of the system.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased. The modified system is calibrated to detect RCS leakage within Technical Specification limits.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 175-76 PSL UNIT #1

"REDUCE NEEDLE FLUCTUATION ON RADIATION MONITORING INSTRUMENTS"

This change was performed for the plant radioactive waste and process radiation monitors. It involved changing capacitors in the log count ratemeter circuit board to increase the circuit time constant. This eliminated the many spurious alarms caused by needle (signal) fluctuation.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The time response of the instruments is not significantly changed. These instruments are not discussed in the Accident Analysis. Eliminating spurious alarms enhances equipment reliability and removes a source of distraction for the operators.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   An equivalent part of changed capacitance is installed. No new accidents or malfunctions are created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 187-76 Unit #1

"ADDITION OF VENT VALVES FOR CCW PUMPS"

Manual globe valves were installed at the 3/4 inch threaded vent connections for each component cooling water pump casing. This was done to provide a convenient means to vent the pump casing section. The vent connections were originally provided with screwed plugs only.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. New materials were purchased and installed to standards which meet or exceed original pump design specifications. Because of the small mass of the new 3/4 inch valves and lightweight tubing, original seismic design was not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. Failure of a vent valve would be similar to failure of the originally installed vent plugs. The FSAR single failure analysis for the CCW system is not affected.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased. This modification reduces the time required for venting component cooling water pump casings following maintenance activities.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 188-76

PSL Unit #1

"ADDITION OF CHECK VALVES IN ICW PUMP LUBE WATER PIPING"

Check valves were installed in each 1 inch bearing lube water supply line to the intake cooling water pumps. This was done to prevent the draining of the lube water piping when lube water supply is shutdown. The draining could lead to the formation of air voids in the piping causing pump bearing overheating.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. All new materials meet or exceed the original design requirements for the intake cooling water system. The new check valves will improve the reliability of the bearing lubrication system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There are no functional changes associated with this modification.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 189-76

PSL Unit #1

"INSTALLATION OF VALVE FOR ILRT PRESSURE CONNECTION"

Gate valve I-V00101(612) and associated supports were installed at containment penetration number 54 in accordance with Amendment No. 2 to the Facility Operating License. This valve was part of original plant design for ILRT pressurization. Installation was deferred because of material availability.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This containment penetration valve is part of the plant design as described in the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

MODIFY INTAKE STRUCTURE CRANE PLATFORM

The original crane platform and ladder allowed personnel to work too close to transmission lines over the crane. The access ladder and platform, needed to allow maintenance on the crane, were relocated to ensure compliance with OSHA requirements.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The crane is not safety related but is located near some safety related equipment. The new platforms were designed and built to the same as or better specifications as the original.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 193-76 PSL Unit 1

"INSTALLATION OF REACTOR COOLANT PUMP OIL DRIP PANS"

Oil drip pans were installed for the reactor coolant pump motors to reduce the possibility of a fire due to uncollected oil leakage. Also, upper oil reservoir gaskets were replaced in accordance with the vendor's recommendations.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related. The oil drip pans were fabricated and installed to Seismic Class I requirements, however, due to the mounting location.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This modification does not affect the function or quality of any plant system.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 195-76

PSL Unit #1

"STEAM GENERATOR BLOWDOWN TREATMENT FACILITY ELECTRICAL TIE-IN"

This change tied in several Steam Generator Blowdown Treatment Facility electrical interfaces to Unit 1. Six valve positions indicating lights and annunciation for high radiation levels were added in the Unit 1 control room. The blowdown treatment facility is a requirement of our license and is described in the FSAR.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. These tie-ins implement the design described in the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The systems affected are not nuclear safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 210-76

PSL Unit #1

"ROUTE STEAM DRAINS AWAY FROM AUXILIARY FEEDWATER PUMP"

Several drains associated with auxiliary feedwater pump 1C steam turbine were routed away from the pump area. This change was made to decrease the possibility of corrosion caused by collection of moisture in control components.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The drain line piping is not nuclear safety related. No functional changes were made.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The new drain piping is small in diameter and will not carry high energy fluids.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 211-76 PSL UNIT #1

"LETDOWN CONTROL VALVES SEAL RING CHANGE"

The pressure seal rings for letdown control valves LCV-2110 P&Q were changed from teflon coated rings to silver plated rings. This change was recommended by the valve vendor to alleviate leakage problems caused by corrosion of the seal rings.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The replacement parts were provided by the original vendor to the same specifications as the original parts.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The change does not affect the function or quality of the letdown control valves.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 213-77 PSL Unit #1

"TURBINE RUNBACK CIRCUITRY REVISION"

The heater drain pumps turbine runback input was revised to a setting of 92% of full load upon tripping of both pumps. Also, a feedwater/steam flow mismatch contact was deleted from the steam generator feedwater pumps turbine runback circuit because of a relay chatter problem. This modification will provide automatic runback to required levels utilizing a simpler, more reliable circuit.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change affects only the non safety related turbine runback circuits.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 216-77 PSL UNIT #1

"PRESSURIZER SPRAY VALVE PLUG TO STEM CONNECTION REPAIR"

This change added a retention weld at the plug to stem connection of the pressurizer spray valves. This was recommended by the valve vendor to alleviate problems with plug and stem separation caused by internal vibration in the valves.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The function and operating characteristics of the valves were not changed. Qualified welders and procedures were used and the valve pressure boundary integrity was not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 217-77 PSL Unit 1

"INTAKE COOLING WATER STRAINER COVER GASKET CHANGE"

This change allows the use of alterate materials for gaskets and washers on the Component Cooling Water Heat Exchanger salt water side inlet strainers. Rubber gaskets were installed and mild steel full washers were used in place of "C" type stud washers. These changes were made to eliminate minor leakage problems.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The quality and function of the replacement materials are the same as the original components.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This is a minor nonfunctional change.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 218-77

Unit #1

MOMENTARY PICKUP - "C" AUX. FEEDWATER PUMP TRIP SOLENOID

This change modifies the trip solenoid controls so the solenoid is picked up momentarily instead of continuously. This allows solenoid to relatch so visual observation better informs operators pump is operable and clears a continuous annunciator window which blocks another needed alarm.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No functional changes are made. Failure of the steam inlet valve controlled by this solenoid is already evaluated in the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments in 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 219-77

PSL Unit #1

"RESIN DEWATERING PUMP PIPING MODIFICATION"

The resin dewatering pump discharge piping was rerouted from the Holdup Tanks to Equipment Drain Tank 1A. This change was made to prevent chemical contamination of the Holdup Tanks with waste resin sluice water.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The new piping, valves and fittings used to implement this change are similar in quality and design to materials originally installed in the spent resin handling system. This modification does not affect safety related components.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 223-77 Unit #1

"CHARGING PUMP CONTROL RELAY MODIFICATION"

This change added diodes to the charging pump pressurizer level control relay coils to prevent electronic noise which was giving interference with the Reactor Cooling System temperature indications.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   There is no change to the design intent or function of the involved circuits.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 224-77 PSL UNIT #1

"MODIFICATION TO RESTRAINT FOR BLOWDOWN VALVE FCV-23-6"

The restraint for blowdown valve FCV-23-6 was modified to improve allowance for thermal growth by line I-2-B-2. This change was needed to prevent the possibility of subjecting the operator structure of the valve to binding.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Components used were designed and built in accordance with applicable codes and FSAR seismic design criteria. This change reduces the probability of a valve failure already analyzed in the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No functional changes are involved.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 226-77

PSL Unit #1

"EQUIPMENT DRAIN PUMPS DISCHARGE PIPING MODIFICATION"

Piping, valves and fittings were installed to connect lines 2-WM-5 and 1½-WM-B02 to provide the capability to process liquid waste from the 1A Equipment Drain Tank and 1A Chemical Drain Tank through waste ion exchangers and the waste filter while a liquid release from the 1B Equipment Drain Tank is in progress. These operations originally required use of the same piping and thus could not be conducted simultaneously.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No safety related components are involved in this change. All new materials used are compatible with the original design and quality group classification of the waste management system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 227-77 Unit #1

"FEEDWATER PUMP DISCHARGE VALVES CONTROL MODIFICATION"

These valves are required to close on both Safety Injection Actuation Signal (SIAS) and Main Steam Isolation Signal (MSIS). It was noted (Refer to Licensee Event Report 335-77-5, dated February 25, 1977) that if a MSIS were present with the feed pump running and no SIAS present the valves would close but then would reopen, close again, reopen, etc. This PC/M corrected that situation so the valves would not reopen on MSIS (or SIAS).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter the valve functions; it just corrects the as-built system to comply with original design intent. It should be noted that the probability of receiving MSIS without SIAS is low.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 228-77

PSL Unit #1

"DELETION OF DROPPED CEA INITIATION OF TURBINE RUNBACK"

The automatic feature of a turbine runback on the occasion of a full length dropped CEA was eliminated by disconnecting the runback circuit.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The runback circuit is not nuclear safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The dropped CEA analysis in FSAR Section 15.2.3 was made on the basis of no turbine runback.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 229-77 PSL UNIT #1

"REPLACEMENT OF VENTS AND DRAINS ON INTAKE COOLING WATER HEADERS"

This PC/M allows the replacement of 1 inch vent and drain valves with caps or plugs on the 30-inch intake cooling water headers. This change was needed to promptly repair leaks that have occurred in the 1 inch vent or drain connections.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The vent and drain valves involved are not required to safely shut the plant down or to prevent or mitigate the consequences of accidents as evaluated in the FSAR. Replacement materials are consistent with original FSAR design criteria. This change does not affect the seismic design of the intake cooling water headers.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The failure of a cap or plug is not different than the failure of a 1 inch vent/drain valve.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 232-77 PSL UNIT #1

"MODIFICATIONS TO INTAKE COOLING WATER PUMP"

The following repairs were made to an intake cooling water pump. These changes were required because of corrosion and wear problems:

a. The pump shaft was repair welded and straightened.
b. Cutless rubber bearings were replaced with polypenco acetal bearings of similar design.
c. A small crack in the pump impeller was repaired.
d. Corrosion attacks in bearing support struts, casing welds, and the o-ring area of shaft tube were repaired using materials selected to increase corrosion resistance.

This change is not an unreviewed safety question because:

1. The probability of occurrrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The repair materials and methods were consistent with original equipment specifications and FSAR criteria for the intake cooling water pumps.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The changes do not affect the function, design bases, reliability, or single failure analysis of the intake cooling water system and components as evaluated in the FSAR.

3. The margin of the safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 236-77 PSL Unit 1

"RELOCATION OF RCP INSTRUMENTATION ON RTGB"

The reactor coolant pump vibration reset switches and motor ammeters were relocated on the control board (RTGB 103). The original placement of these instruments was inconsistent with the location of other reactor coolant pump instruments and was a source of unnecessary confusion. The control wiring schemes for these instruments were not changed.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The affected components are not nuclear safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No functional change was made.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 242-77

PSL Unit #1

"REACTOR REGULATING SYSTEM CEA CONTROL MODIFICATION"

The RRS was designed to automatically control RCS temperature by insertion/withdrawal of CEA's as needed. Due to the vendors recommendations to operate with all CEA's normally fully withdrawn and not to use CEA's for power increases it was decided to modify the system by removing the CEA automatic withdrawal function. Also, an audible indication was added to inform the operators of an automatic CEA insertion. This change in consistent with the vendor's fuel operating guidelines.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The Reactor Regulating System is not a nuclear safety related system. This change provides a conservative approach to prevent exceeding the T-inlet Limiting Condition of Operation (LCO= 542°F).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 246-77 Unit #1

"CONTAINMENT SPRAY PUMP AMMETER REPLACEMENT"

The containment spray pumps ammeter was changed from a 0-75 amp scale to a 0-100 amp scale to conform with original design specifications. This change provided an increase in accuracy by extending the range of pump motor amps indication.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The replacement components conform to original specifications.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The pump control system was not altered. This change only involved the range of an indicator.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 251-77

PSL Unit #1

"RCP SEAL COOLER VALVES REMOTE OPEN SWITCHES"

Remote open switches were installed in the control circuit of the Reactor Coolant Pump seal cooler supply valves. This change was made to avoid unnecessary entries to the containment building to re-open these valves and to reduce the time duration for loss of seal cooling water due to inadvertent valve closures.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This modification decreases the possibility of malfunction of a Reactor Coolant Pump by allowing timely remote resetting of the seal cooler supply valves. This control circuit change does not affect any automatic control features.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 258-77

PSL Unit #1

"CONTAINMENT BUILDING INSTRUMENT AIR SYSTEM MODIFICATIONS"

Modifications were made to improve the reliability of the instrument air system inside containment. Changes include the addition of spring-loaded check valves in the compressor discharge lines, installation of a pressure regulator in the turbine building instrument air supply to containment, and the addition of a low pressure alarm.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Containment instrument air is not a safety related system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No changes in function of the instrument air system were made. These modifications improved the reliability of the system.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 259-77 PSL Unit 1

"INSTALL CURBING AROUND STATION SERVICE TRANSFORMERS"

Six inch high concrete curbs were installed around the station service transformers to contain any possible leak or spill of askarel (PCB). This change was made because of recent concerns about the adverse affects on the environment by the askarel (PCB's).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This is a non-nuclear safety related change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No nuclear safety related equipment is affected by this change.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as decribed in the Final Safety Analysis Report.

Plant Change/Modification No. 260-77 Unit #1

"INSTALLATION OF POWER SUPPLY TEST JACKS FOR NUCLEAR INSTRUMENTATION DRAWERS"

15-volt power supply test jacks were installed in the linear range and wide range nuclear instrumentation drawers. This change was made to provide easier access for monitoring and maintaining equipment and to eliminate a potential personnel safety hazard.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Materials used to implement this change are consistent with the original vendor supplied equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This modification does not affect the function or reliability of the Nuclear Instrumentation System.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 264-77 Unit #1

"MODIFICATION OF ATMOSPHERIC DUMP VALVES AND CONDENSATE STORAGE TANK"

During an engineering design review, an error in the sizing of the atmospheric steam dump valves was discovered (Reference Reportable Occurrence 335-77-22). This could have caused problems in plant cooldown to the shutdown cooling window in the unlikely event of loss of offsite power for an extended time period. The following modifications were made to correct the deficiency:

a. The valve internals for I-HCV-08-2 A & B were changed from 6x3 to 6x4 to provide increased flow capacity. New valve actuators, silencers, and modified restraints were installed to accommodate the increased flow characteristics.

b. The Seismic Class I capacity of the Condensate Storage Tank was increased to 160,000 gallons by relocating a piping nozzle.

c. * Digital indicators for RCS temperature were added in the Control Room to provide increased accuracy for transition to shutdown cooling.

d. ** The shutdown cooling piping support designs were re-evaluated for a proposed increase in shutdown cooling entry temperature. Several minor support modifications were made. Some additional piping support changes are expected to be made in 1978.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Replacement components are purchased and installed to the same or better specifications as the original components. New materials are compatible with the as-built systems. Present FSAR analyses envelope the consequences of any accident or malfunction related to this modification.

Plant Change/Modification No. 264-77 (Cont)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. There is no change in function of any safety related system. Replacement components are of the same generic design as the originals.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased. Design bases are met with changes to shutdown cooling entry temperature. (Technical Specification revisions are under consideration at this time.

* The digital temperature indicators are not yet operable due to material (RTD's not yet available). The RTD's will be installed and the indicators made operable during the April 1978 refueling.

** The remainder of the restraints will be adjusted/ modified during the 1978 refueling. A request for Technical Specification change to allow operation at higher temperatures has been submitted but not yet approved by the NRC.

The system will not be operated at the higher temperature until the items above are completed and the Technical Specification change is approved by the NRC.

Plant Change/Modification No. 268-77 PSL UNIT #1

"PRESSURIZER SPRAY VALVE INTERNALS MODIFICATION"

Redesigned pressurizer spray valve plugs were provided to correct lateral vibration problems experienced when operating the valves partially open. The design of the new valve plugs redistributes the forces of the process fluid on the valve plug such that the net force on the plug at low lifts will be directed in one direction and thus eliminate the vibration.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new valve plugs were provided by the valve vendor and were built to equivalent specifications as the original parts.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The redesigned plug has equivalent capacity and characteristics as the original plug.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

The change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 274-77

PSL Unit #1

"SECURITY LIGHTING STANDARD MODIFICATION"

Installation of a Unit 2 construction tower crane deadman required removal of a Unit 1 lighting pole foundation and re-routing of conduit for lighting. It was determined by measurement that this light is not required for security illumination.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change does not affect nuclear safety related systems.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 275-77

PSL Unit #1

"RTGB ANNUNCIATOR FLASH RATE CHANGE"

A resistor was changed in the Rochester Instrument Systems DC Momentary Alarm and Flasher Module to increase the flash rate. This was done to improve operator awareness of the alarming window.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The annunciator system is not nuclear safety related. The replacement resistor is of equivalent quality as the original part. The increased flash rate will make this system more effective to the operators.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. No changes in function were made.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The affected components are not discussed in the technical specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 280-77

Unit #1

"RCS TORNADO PROTECTED MAKEUP WATER SOURCE"

Installed piping and valves between the Safety Injection Tanks drain line and the Volume Control Tank to allow the use of safety injection tank borated water inventory as an emergency shrink makeup water source to the Reactor Coolant System. This modification resolves condition I.4 of the St. Lucie Unit 1 Operating License.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change does not affect actuation or functional performance of plant safety related systems. The new piping and valves are located in the auxiliary building, which is designed for tornado missiles. Material selection is consistent with the design of the interfaced systems and with FSAR quality group classifications.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. All materials added by this modification are passive low pressure and low temperature components.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 281-77 PSL Unit 1

"ANNUNCIATE INVALID DISCHARGE CANAL LEVEL AND TEMPERATURE ALARMS"

Circulating Water System discharge canal level and temperature indication circuitry was revised to provide alarms in the control room upon loss of data. This change was made to alert the operators that the data channel from the discharge canal is inoperative.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. No safety related components are affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 285-77

PSL Unit #1

"MODIFICATION OF CONTROL ROOM DOORS"

Five control room doors were modified to improve the operation of security system card reader controls and to improve leak tightness. The direction of swing was reversed for four doors and magnetic type seals were installed for three doors.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This is not a nuclear safety related change. The leak tightness of the doors was improved.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This is a non-functional change made to improve the reliability and operation of the doors.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 286-77 Unit #1

"ANNUNCIATION OF EMERGENCY COOLING CANAL VALVE POSITION"

The annunciator scheme for the isolation valves in the flow barrier between the intake structure and Big Mud Creek was modified to provide control room annunciation when the valves start to open. Originally annunciation occurred only when the valves reached the full open position.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This is a minor wiring change for annunciation of valve position only.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. This change does not affect valve control scheme, characteristics, or function.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased. Annunciation when a valve starts to open will assist in meeting the Environmental Technical Specification limits on flow through these valves.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 293-77 Unit #1

"TEMPORARY TEST CONNECTIONS FOR CHARGING SYSTEM"

Pressure taps were installed at the suction and discharge of each charging pump and at the common pump discharge line to facilitate testing. These test connections were used to monitor and record pressure pulsations in the charging system, and were removed following completion of testing. (See Page 59 of this report for a summary of the test procedure).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Although these test connections were temporary, materials used (valves, fittings, etc) were consistent with the design and quality of the charging system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. Charging pump controls and permanent indication and alarms were not affected by this temporary change.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 299-77 Unit #1

"INTERIM RCS OVERPRESSURE MITIGATING SYSTEM"

The pressurizer power operated relief valves control scheme was revised by installation of a variable low pressure setpoint. This modification is designed to mitigate pressure transients in the Reactor Coolant System during plant startups and shutdowns. This installation is considered to be an interim fix until NRC approval is received and verification of the water relief capability of the power operated relief valves is complete. A description of this interim solution was forwarded to the NRC in our letter of August 23, 1977 (L-77-257).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Electrical interlocks and administrative controls are provided to prevent an inadvertent PORV actuation signal. The mechanical characteristics of these valves are not changed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The equipment added by this modification is not nuclear safety related. Where a safety related circuit is interfaced, suitable isolation is provided.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

Plant Change/Modification No. 301-77

PSL Unit #1

"REPLACEMENT OF INTAKE STRUCTURE WARNING LIGHT"

The intake structure warning signal and supporting components were removed and a new lighted marker buoy was installed. This change was approved by the United States Coast Guard.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. This change is not nuclear safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 302-77 Unit #1

"GOULD INVERTER CAPACITOR JUMPER WIRING MODIFICATION"

The commutating and power factor jumper wires were revised to separate the parallel connected capacitor banks into smaller parallel loops in order to reduce the maximum currents in each jumper wire. These wires had shown signs of premature aging. The 10 KVA and 15 KVA inverters were affected.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. New materials were specified as equal to or better than those originally installed. Wiring was sized adequately to increase component reliability.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The function of the inverters was not altered.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 308-77 PSL UNIT #1

"MAIN STEAM CHECK VALVES BACKSTOP MODIFICATION"

This change added a valve disc backstop to the main steam check valve cover plate to relieve loading on the original stop. Examination of the original backstop area revealed excessive wear. This modification improved the mechanical advantage and increased the backstop surface area to prevent deterioration of the valve internals.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This modification is consistent with the original valve design criteria, and does not alter the function or operating characteristics of the main steam check valves. Calculated loadings on the cover and bolts are within allowable limits of the applicable code.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new stop plate is a non pressure boundary apertenance welded to a pressure boundary part. Welding procedures and materials conform to applicable codes and specifications.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 309-77 PSL UNIT #1

"INSTALLATION OF PIPING RESTRAINT FOR SAFETY INJECTION LINE I-24-SI-506"

As a result of our program for inspecting concrete anchor bolts used in pipe hanger installations, it was discovered that support number SIH-57 was not installed in conformance with the as-built design drawings. This PC/M provided for installation of that support. (Reference FPL letter to NRC, #L-77-312 dated October 7, 1977).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change restored line I-24-SI-506 to the original design configuration. The support materials, location, and design loading, satisfy the FSAR design criteria.

2. The possiblity for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 311-77 PSL Unit 1

"REACTOR CAVITY SUMP LEVEL INDICATION MODIFICATION"

The reactor cavity sump liquid level transmitter was changed from a float type to a bubbler type instrument. Also, the instrument was relocated to outside the secondary shield wall. These changes will facilitate maintenance and improve the reliability of the level indication (LT-07-06).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. The instrumentation affected in not nuclear safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created. The change does not affect the function or quality of this level measuring instrument.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. This modification improves the reliability of the reactor cavity sump level monitoring system.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PROCEDURE CHANGES
January 1, 1977 - December 31, 1977

The following summarizes changes to procedures as listed in the FSAR in accordance with Title 10, Code of Federal Regulations, Part 50.59.

EMERGENCY AND OFF NORMAL PROCEDURE NO. 0030142
RCS COOLDOWN DURING BLACKOUT

This procedure provided instructions to ensure the capability of reaching cold shutdown in the event of loss of offsite power considering the design deficiency of the installed atmospheric steam dump valves as reported in LER 335-77-22 dated April 22, 1977. Instructions were provided to bypass the low vacuum interlock and to remove a flange in the secondary steam dump system for additional steam dumping capacity. Also, this procedure pointed out that other sources of condensate storage tank make-up water are useable if necessary (city water transferred by fire pumps and hose). Permanent corrective actions for the atmospheric steam dump design error are summarized in the design change section of this report (See PC/M #264-77). This new procedure does not impact the safety analyses of the FSAR or the Technical Specifications and was determined not to involve an unreviewed safety question.

TESTS

The following list summarizes special tests performed under the provision of Title 10, Code of Federal Regulations, Part 50.59 during the period January 1, 1977 through December 31, 1977.

Charging System Pulsation and Vibration Testing

Pressure pulsation, strain and vibration data were recorded at different RCS pressures and charging pump operating configurations (1,2 and 3 pump operation) to provide an extensive data base for determining the causes of charging system pulsation problems. Damage to instruments had been experienced during low pressure operating modes.

This test was performed on September 24 and September 25, 1977 prior to and during plant cooldown for a scheduled maintenance outage. Southwest Research Instiutte (SWRI), a consultant organization assisted in the collection and evaluation of data. Recommendations to resolve the apparent problems will be considered following receipt of a full report from SWRI.

The charging system was operated within design limits and in accordance with approved procedures at all times. This test did not constitute an unreviewed safety question.

Concrete Expansion Anchor Bolt Verification

At the request of the NRC a program for inspecting anchor bolts and concrete expansion anchors utilized in seismic class I pipe hanger installations was completed. The inspection procedure included checks for anchor bolt size and engagement as well as verification for proper installation. (Letter of Instruction T-03, revision 1). The results of this program were summarized in our letter to the NRC, L-77-312, dated October 7, 1977.

This inspection program did not involve an unreviewed safety question.

RDT Sensor Response and Process Noise Data Testing

Signal measurements were recorded for control channel temperature instruments to obtain steady state and transient data on RTD signals to develop a technique for on-line verification of RTD sensor time response. Also, the test was to obtain steady state data on process input signals, such as feed the core protection calculator, to provide the NSSS vendor with typical noise levels. This was done to support RTD response time testing as described in the FSAR and required by Technical Specifications.
Test cables were attached to selected points in the C-E patch panel. Data was collected on May 10, 11 and 19, 1977.

The plant was operated within design limits and in accordance with approved procedures at all times during this test. This test did not involve an unreviewed safety question.

CORE BARREL MOVEMENT

Section 4.4.11.3 of the PSL #1 Technical Specifications requires the results of all periodic Amplitude Probability Distribution (APD) and Spectral Analysis (SA) monitoring to be included in this report.

In March, 1977 the baseline measurements were completed and a full report of levels observed was submitted to the Commission in April, 1977. This report also detailed the determination of alert and action setpoints pursuant to specification 4.4.11.1.

During routine monitoring on June 21, 1977 the RMS levels of noise on four of the excore detector channels exceeded their baseline values by 10%. A special report describing measured levels and identifying likely causes for the increases was submitted pursuant to specification 3.4.11c.

As anticipated, the RMS levels of all detector signals have continued to increase on a gradual trend throughout 1977. This trend has been observed on both APD and SA monitoring and represents an increase of about 30% over the baseline levels. The greatest portion of this increase remains in the lower frequency band (1 - 4 Hz) (normally observed due to changes in coolant temperature coefficient of reactivity and fuel element vibration coefficient of reactivity).

The observed increases in RMS values, if attributed entirely to core barrel motion, would indicate an increase of less than .001" RMS motion at the snubber level.

STEAM GENERATOR TUBE INSPECTIONS

Section 4.4.5.5.b of the PSL #1 Technical Specifications requires reporting all Steam Generator Tube Inspections in this report. For the period January 1, 1977 through December 31, 1977 no tube inspections were performed.

It is planned that tube inspections, as specified by Section 4.4.5.3.a of the Technical Specifications, will be performed during our first refueling in 1978 and reported in the Annual Operating Report for that year.

WORK & JOB FUNCTION	NUMBER OF PERSONNEL (>100 mrem)			TOTAL MAN-REM		
	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT WORKERS & OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACT WORKERS & OTHERS
Reactor Operations & Surveillance:						
Maintenance Personnel	0	0	0	0.00	0.00	0.00
Operating Personnel	32	0	0	7.90	0.00	0.00
Health Physics Personnel	14	0	0	4.51	0.00	0.00
Supervisory Personnel	1	0	2	0.15	0.00	0.21
Engineering Personnel	2	0	0	0.26	0.00	0.00
Routine Maintenance:						
Maintenance Personnel	78	0	67	43.69	0.00	12.53
Operating Personnel	2	0	0	0.33	0.00	0.00
Health Physics Personnel	11	2	1	5.57	0.22	0.13
Supervisory Personnel	1	1	6	0.15	0.11	0.79
Engineering Personnel	0	0	1	0.00	0.00	0.14
Inservice Inspection & Special Maintenance:						
Maintenance Personnel	7	0	21	3.35	0.00	5.94
Operating Personnel	27	0	0	15.63	0.00	0.00
Health Physics Personnel	8	0	0	6.47	0.00	0.00
Supervisory Personnel	2	1	3	0.71	0.11	0.41
Engineering Personnel	2	0	3	1.24	0.00	1.65
Waste Processing:						
Maintenance Personnel	36	0	0	10.55	0.00	0.00
Operating Personnel	18	0	0	2.91	0.00	0.00
Health Physics Personnel	17	0	0	4.91	0.00	0.00
Supervisory Personnel	0	0	0	0.00	0.00	0.00
Engineering Personnel	0	0	0	0.00	0.00	0.00
Refueling:						
Maintenance Personnel	0	0	0	0.00	0.00	0.00
Operating Personnel	0	0	0	0.00	0.00	0.00
Health Physics Personnel	0	0	0	0.00	0.00	0.00
Supervisory Personnel	0	0	0	0.00	0.00	0.00
Engineering Personnel	0	0	0	0.00	0.00	0.00
TOTAL:						
Maintenance Personnel	80	0	84	57.59	0.00	18.26
Operating Personnel	37	0	0	26.79	0.00	0.00
Health Physics Personnel	21	2	1	21.46	0.22	0.13
Supervisory Personnel	4	2	10	1.01	0.22	1.41
Engineering Personnel	4	0	3	1.50	0.00	1.79
GRAND TOTAL	146	4	98	108.35	0.44	21.59

ABBREVIATIONS USED

A/C	Air Conditioner
AOV	Air Operated Valve
B.A.	Boric Acid
CCP	Coolant Charging Pump
CCW	Component Cooling Water (for Rx plant components)
Ch	Channel (i.e. one of four channels of the RPS)
CVCS	Coolant and Volume Control System (Charging and letdown)
CWD	Control Wiring Diagram
Disch	Discharge
FCV	Flow Control Valve
FW	Feedwater
FWP	Feedwater Pump
Hdr	Header
HPSI	High Pressure Safety Injection
Hx	Heat exchanger
ICW	Intake Cooling Water (sea water cooling for CCW, Turbine Cooling Water)
ISO or ISOL	Isolation (valve)
Ix	Ion exchanger (demineralizer)
LCV	Level Control Valve
LPSI	Low Pressure Safety Injection
MOV or MV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NI	Nuclear Instrumentation
PCV	Pressure Control Valve

ABBREVIATIONS (cont)

PRZR or PZR	Pressurizer
RCP	Reactor Cooling Pump
RV	Relief Valve
Rx	Reactor
SDC	Shutdown Cooling (decay heat removal system)
S/G or S.G.	Steam Generator
SIT or SI Tank	Safety Injection Tank (Accumulator)
TM/LP	Thermal Margin-Low Pressure
TX	Transmitter
VCT	Volume Control Tank
V/I	Voltage to Current (signal) converter

1976 ANNUAL OPERATING REPORT

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE UNIT #1

February, 1977

INDEX

SUMMARY OF OPERATING EXPERIENCE

The following is a summary of Plant Operations including pertinent items of interest chronologically for the period 3-1-76 (operating license issued) to 12-31-76. The plant did not reach 100% power during this period.

3-1-76	1)	Full security plan implemented
	2)	Commenced preparations for initial core load
	3)	Operating license issued, effective 3-1-76
3-2-76	1)	Dummy with neutron source loaded in X-11 for instrument response check
3-3-76	1)	2:45 A.M. core load commenced
3-4-76	1)	Fuel loading secured (11:45 P.M. 3-3-76 - 2:37 A.M. 3-4-76) Containment integrity was breached due to decreasing level in transfer canal - canal refilled to proper level. Reported per LER 335-76-1, dated April 2, 1976.
3-5-76	1)	Surveillance check of Spent Fuel Machine performed 6:35 A.M. Overload did not function properly - fuel loading secured. Reported via LER 335-76-2, dated April 5, 1976
	2)	1:30 P.M. - NRC issued (per telephone) Amendment #1 to license DPR-67 allowing operation without the overload for a period of two weeks (for non-irradiated fuel).
	3)	3:30 P.M. - Fuel loading recommenced
3-11-76		2:26 P.M. Last fuel assembly loaded into core
3-12-76	1)	5:50 A.M. Upper guide structure installed
	2)	11:45 P.M. Completed latching CEA's
3-14-76		12:22 A.M. Vessel head in place
3-15-76		6:45 A.M. Entered Mode 5, Vessel head torqued
3-16-76		Commenced fill and vent and heat-up preparations
3-20-76		4:40 P.M. CEDM cable connections completed
3-21-76	1)	Fill and vent - 3:50 A.M. - water issued from Pzr vent - vent closed, increased RCS to 200 psi.
	2)	10:59 P.M. - Commenced 30 sec. pump runs (RCP) for venting Reactor Coolant System (RCS).
3-23-76		3:13 P.M. RCP runs complete for fill and vent.
3-25-76		4:30 P.M. - Commenced heating Pzr to draw bubble

3-26-76	1)	2:35 A.M. - Started 1B1 and 1B2 RCP's for heat-up
	2)	5:45 A.M. - RCS O_2 at 2.8 ppm @ 200°F - stopped 1B2 RCP, began reducing O_2 conc.
	3)	4:15 P.M. - Commenced Pre-Op 0110081 CEDM/CEA Testing
	4)	10:11 P.M. - O_2 @ .05 ppm - heat-up recommenced.
3-30-76		1:00 A.M. - 1.23 GPM Leak rate from RCS - found to be RCP Bleedoff relief - repaired and returned to service 11:30 A.M.
4-1-76	1)	Entered Mode 4
	2)	4:30 P.M. - Declared CEA 44 inoperable during preoperational test
	3)	11:00 P.M. - Drew vacuum @ 26" in Turbine condenser
4-2-76		Continued CEDM testing
4-3-76		6:00 A.M. Entered Mode 3 RCS @ 300°F
4-4-76	1)	6:00 A.M. Commenced heat-up to 470°F, 1800 psia
	2)	Performed OP 0110081B, CEDM/CEA 44 Lower Gripper check
	3)	Completed filling fuel pool
4-5-76	1)	1:15 A.M. Commenced Heat-up to 510°F and 2150 psia
	2)	9:15 A.M. - CEA #44 declared operable
	3)	2:45 P.M. - Started 4th RCP - Commenced heat-up to 532°F - 2250 psia
4-6-76		Pre-Op 0110081, CEDM Drop Testing in progress at 532°F, 2250 psia.
4-10-76		7:00 A.M. - CEA Drop testing per Pre-Op 0110081 complete
4-11-76		11:23 A.M. - Conducted RCS Flow Coastdown Test
4-12-76		9:00 A.M. - OP 0110081 In progress - Fuse Block on 1A2 CWP Breaker caused discharge valve on 1A2 CWP to close with pump operating - Rapid action by NPS (opening breaker locally - manually - would not open from control center) avoided damage to 1A2 CWP.
4-13-76	1)	12:35 P.M. - Secured all RCP's for no-flow CEA testing
	2)	8:32 P.M. - Started 3 RCP's
	3)	11:30 P.M. - Commenced cooldown to Mode 4 for testing CEA 44.
4-14-76	1)	5:00 A.M. - Entered Mode 4
	2)	11:30 A.M. - CEA 44 Inoperable
	3)	4:00 P.M. - Heat-up to 420°F to Mode 3
	4)	9:30 P.M. - CEA 44 operated @ 420°F 1260 psia
4-15-76	1)	12:55 A.M. - Entered Mode 4 RCS 280°F, 470 psia
	2)	4:05 A.M. - Retested CEA 44, Test Satisfactory
	3)	4:45 P.M. - Heat-up, entered Mode 3
	4)	11:36 P.M. - 4 RCP's in service
4-17-76		Conducted S/G Feedwater Hammer Testing

4-18-76	Normal Plant Ops (NPO) - making preparations for initial criticality
4-20-76	7:00 P.M. - Commenced CEA testing IAW App 3, Addendum I of Initial Criticality Procedure
4-21-76	Continued Rod Testing 1) 7:00 P.M. - Completed App. 3 of Initial Crit. Procedure. 2) CEDM 44 would not operate properly cold but did function hot. Amendment 4 to license DPR-67 has been issued, deleting approval to go critical cold for testing, thus allowing hot operations and criticality with CEDM 44 to be repaired/replaced at a later date. 3) 7:56 P.M. - Began diluting RCS IAW Initial Crit. Procedure
4-22-76	1) 7:10 A.M. - Entered Mode 2 2) 8:30 A.M. - St. Lucie #1 Critical, Boron Conc., 935 ppm 3) 3:40 P.M. - Commenced Low Power Physics Testing
4-26-76	1) 6:00 A.M. - Rx subcritical for CEA drop test 2) 7:43 A.M. Rx critical 3) 10:51 A.M. - Use of part length CEA's to control power - Inserted below 90% withdrawn limit. Reported in LER 335-76-18, dated May 26, 1-9-76 4) 3:10 P.M. - Reactor shutdown - In Mode 3 for repair of CEDM Reed Switches
4-27-76	1) 11:15 - Rx critical - Entered Mode 2 2) 8:27 P.M. - Recommenced Low Power Physics Testing
4-30-76	1) 2:55 P.M. - Completed Low Power Physics 2) 3:30 P.M. - Rx shutdown, Mode 3
5-2-76	Attempted to withdraw CEA's - Timers bad on several - manual trip for repair.
5-3-76	11:37 P.M. - Reactor critical
5-4-76	1) 5:35 A.M. - Rx Trip - Operator error while testing Nuclear Instrumentation 2) 9:32 P.M. - Rx Critical
5-5-76	1) 1:30 A.M. - Rx S/D to repair CCW Line to RCP motor 2) 7:51 A.M. - Rx Critical 3) 10:42 A.M. - Commenced power ascension to 5%
5-7-75	1) 4:00 A.M. - Rx Trip while conducting Steam Bypass Control System (SBCS) Test due to Low S/G level 2) 6:00 A.M. - Rx critical 3) 6:24 A.M. - Rx Trip (Low S/G level) 4) 7:05 A.M. - Rx Critical 5) 12:24 P.M. - Rx Trip (Low S/G Level) 6) 1:55 P.M. - Rx Critical 7) 8:48 P.M. - PSL Unit 1 on Line @ 100 MW Note: The FW Control System was later adjusted (at power) to minimize recurrence of these problems.
5-8-76	1) 12:15 P.M. - Rx Trip (Low S/G Level) while performing Turbine overspeed tests

*Indicates forced power reduction of 20% or more per Reg. Guide 1.16

5-8-76	2)	1:40 P.M. - Commenced Rx Start-up
	3)	1:44 P.M. - Dropped CEA 32 due to blown SCR, replaced SCR. (See Note 2)
	4)	5:18 P.M. - Rx Critical
5-9-76		2:18 A.M. - Performed turbine overspeed test - Sat @ 1971 RPM- Unit on Line @ 100 MW
5-10-76	1)	3:51 P.M. - Manually tripped unit due to closure of 1B MSIV - cause DC ground
	2)	7:04 P.M. - Rx Critical
	3)	11:05 P.M. - Unit on line
5-12-76	1)	2:55 P.M. - Conducted 20% Trip test
	2)	8:40 P.M. - Rx Critical
5-13-76	1)	2:27 A.M. - Unit 1 on line @ 160 MW (20%)
	2)	6:00 P.M. - Power ascension to 30%
	3)	7:30 P.M. - @ 30%
5-14-76	*1)	5:09 A.M. - Rx Trip due to loss of Feedwater (feed pump tripped), performed S/G Water Hammer Test (See Note 1)
	2)	3:11 P.M. - Rx Critical
	3)	11:30 P.M. - Unit on line - Ascension to 50% plateau commenced
5-15-76	1)	9:30 A.M. - 40% plateau
	*2)	8:00 P.M. - Problems with water hammer in feedwater heaters while placing moisture separator/reheaters (MSR's) in service - Unit manually shut down from 50% - Entered Mode 3 for addition of restraints to secondary plant (60 hr. S/D)
5-16-76		Start-Ups performed by hot license candidates for training
5-18-76		9:10 Rx Critical - performed training start-ups
5-19-76		6:10 A.M. - Secondary plant restraint addition complete, Unit 1 on line and going to 40%
5-20-76		Ascension to 50% power
5-21-76		*5:23 P.M. - Rx Trip - Low S/G Level, 1A MFWP Tripped, 1B MFWP did not start. (See Note 1)
5-22-76	1)	4:20 A.M. - Rx critical
	2)	8:25 A.M. - Rx Trip - Low S/G Level During turbine startup
	3)	10:22 A.M. - Rx Critical
	4)	1:41 P.M. - Unit on line, increasing power to 50%
5-23-76	1)	4:00 P.M. - Lightning strike in switchyard caused 4 loss of load pretrips
5-24-76		Normal Plant Operations (NPO) @ 50%
6-1-76.		NPO @ 50%

*Indicates forced power reduction of 20% or more per Reg. Guide 1.16

6-6-76		NPO @ 50% Set up for Moderator Temperature Coefficient (MTC) and Power Coefficient Test
6-7-76	*1)	9:40 A.M. - Manual Trip of Rx @ 50% due to dropped CEA #26 during MTC and Power Coefficient Test. Cause - power switch on CEA - repaired. (See Note 2)
	2)	5:24 P.M. - Rx Critical
	3)	7:44 P.M. - on the line
6-8-76		1:30 A.M. - Reached 50% plateau
6-12-76		4:56 P.M. - Tripped Reactor @ 50% - To perform Control Room Inaccessibility Test
6-13-76		Cooling Down RCS for a scheduled maintenance shutdown
	1)	8:00 A.M. Entered Mode 4
	2)	12:52 P.M. - On Shutdown Cooling (SCD)
	3)	6:35 P.M. - Entered Mode 5
6-17-76		Fill and Vent of RCS
	1)	When first RCP was started, there was a pressure transient to 300 psi above the pressure temperature limit (at 100°F) for less than 5 minutes. Reported in LER 335-76-30, dated 7-17-76.
	2)	4:30 P.M. - Commenced drawing Pzr Bubble
6-18-76		11:53 A.M. - Entered Mode 3
6-19-76	1)	9:53 A.M. - Rx Critical
	2)	5:25 P.M. - On the line - Commenced Power Ascension to 80%
6-20-76	1)	2:02 P.M. - Increasing Power to 60%
	2)	6:40 P.M. - @ 60%
6-21-76		9:10 A.M. - Reached 70%
6-22-76	1)	9:27 A.M. - Dropped CEA #20 - Turbine load reduced to match Rx power @ 60% - recovered CEA
	2)	2:28 P.M. - Reached78% plateau
	*3)	5:00 P.M. - Reduced load to 50% to clean Feedwater Pump (FWP) strainers
6-23-76		*11:28 P.M. - Rx trip @ 50% - Cause low S/G level due to closure of 1 A MSIV - MSIV closed due to opening of its DC Breaker. The breaker was opening while investigating cause of loss of AB DC Bus. AB DC Bus was lost while trying to secure A Battery Charger which was oscillating.
6-24-76	1)	9:45 A.M. - Rx Critical
	2)	12:18 P.M. - On line, increasing power to 70%
6-25-76		6:15 A.M. - @ 70%
6-26-76		*3:30 A.M. - Reduced load from 70% to 50% to clean 1B FWP strainer

*Indicates forced power reduction of 20% or more per Reg. Guide 1.16

6-28-76		Increased power to 78%
7-1-76	1)	1:35 A.M. - Commenced load reduction from 78% due to high usage of H_2 in Generator (Note - leak found & repaired)
	*2)	2:55 A.M. - Reactor Trip (TM/LP) Thermal Margin/Low Pressure-While Borating to reduce power and transferring from single to sequential valve control, generator picked up 60 - 70 MW - rapidly decreasing RCS pressure caused trip.
7-2-76	1)	3:05 A.M. - Rx Critical
	2)	8:10 A.M. - Unit on line
	3)	8:33 A.M. - Rx Trip @ 10% - Low S/G level
	4)	11:20 A.M. - Rx Critical
	5)	1:11 P.M. - Rx Trip - DEH Malfunction @ 10%
	6)	3:25 P.M. - Rx Critical
7-3-76		Increased power to 78%
7-4-76	1)	3:23 A.M. - Turbine Runback 78% - 70% - due to malfunction in Runback circuit - repaired
7-5-76		NPO @ 78%
7-6-76	*1)	7:20 P.M. - Reduced Power from 78% to 50% to clean condensate pump strainer. At this time we first became aware of a possible flux distribution anomaly. See the write-up attached to PCM 176-76 in this report. Also reported in LER 335-76-35 dated July 23, 1976.
7-7-76		NPO @ 48%
7-9-76		NPO @ 48%
	*1)	Reduced power at 1:15 A.M. to off line for testing. This was to help determine cause of the flux distribution anomaly.
	2)	11:27 A.M. Generator off line
	3)	@ 10^{-2}% power for physics testing
7-14-76		NPO @ 10^{-2}%
	1)	2:06 A.M. - Rx Trip - Operator error - during RPS Logic Matrix Test
7-15-76	1)	8:02 P.M. - Rx Critical @ 10^{-2}%
7-18-76		NPO @ 10^{-2}%
	1)	1:19 - Tripped Rx - Borating to Refueling Concentration
	2)	9:45 A.M. - On SDC
	3)	12:30 P.M. - Entered Mode 5
7-26-76		Decision to remove some fuel for inspection confirmed by Company management.
7-27-76		6:00 A.M. PZR solid

*Indicates forced power reduction of 20% or more per Reg. Guide 1.16

8-1-76	Entered Mode 6 - 8:30 A.M.
8-3-76	Uncoupling CEA's
8-4-76	Dual CEA problems - See PCM 203-76
8-5-76	Drilled the "7" Slots on Dual CEA shafts. All CEA's uncoupled.
8-8-76	Commenced removing fuel for inspection of fuel assemblies
8-18-76	Commenced defueling core
8-23-76	12:14 A.M. Core defueled
8-23-76 - 11-15-76	Fuel Reconstitution - See PC/M's 176-76, 192-76, 200-76 and discussion attached to PC/M 176-76
11-4-76	Commenced core loading Cycle 1A
11-15-76	Fuel reconstitution complete
11-16-76	Core loading complete, started reassembly of Reactor vessel
11-25-76	Reassembly complete, entered Mode 5
11-26-76	On Shutdown Cooling (SDC)
11-28-76	6:30 P.M. - RCS Solid
11-29-76	Started RCP Runs for fill and vent
11-30-76	Fill and Vent 1) 8:15 A.M. - Commenced drawing Bubble in PZR 2) 3:37 P.M. - Entered Mode 4 3) 10:30 P.M. - Entered Mode 3
12-2-76	Mode 3-CEA Testing including testing of CEDM 44, replaced during the fuel reconstitution shutdown.
12-3-76	RCS Flow Test and low flow trip setpoint verification
12-4-76	1) 2:10 A.M. - Commenced diluting to critical 2) 3:37 P.M. - Rx Critical
12-5-76	@10^{-2}% CEA Symmetry Test In progress
12-6-76	@10^{-2}% CEA Testing
12-7-76	@10^{-2}% CEA Testing
12-8-76	@10^{-2}% Isothermal Testing
12-9-76	10:25 PM - increased Power to 3%

12-10-76 1) 5:58 A.M. - Mode 1
2) 7:53 A.M. - On the line at power, 20% plateau

12-11-76 1) 6:00 A.M. - @ 30% plateau

12-16-76 1) 3:50 P.M. @ 40% plateau

12-17-76 1) 8:56 P.M. @ 50% plateau

12-22-76 Still at 50%. Declared PSL #1 Commercial at 12:01 A.M.

12-27-76 NPO @ 50%. Conducted Moderator Temperature Coefficient and Power Coefficient test

12-31-76 Continuing operations at 50% power

Note 1 We have experienced some difficulty with MFW pumps tripping while running and tripping off immediately after automatically starting when the running pump has tripped. This appears to be caused by signals from the feed pump protective trip circuits (low suction pressure and flow). We have modified the Feed Pump recirculation valve controls and are continuing to gather data and evaluate the problem so any further corrective action necessary can be defined and implemented.

Note 2 Shortly before licensing the CEDM power supply vendor and the NSSS vendor recommended a modification to a power supply module to improve reliability. This was approved and drawings issued before licensing. As modified modules became available, throughout the power ascension, the new modules were installed. Since return to operation in December with all modules replaced, there have been no dropped CEA's due to these modules. The modification primarily consisted of upgrading the voltage rating from 400 to 600 volts.

OUTAGE SUMMARY MARCH 1, 1976 - DECEMBER 31, 1976

The following summarizes the three plant shutdowns performed for maintenance from initial criticality on April 22 through December 31, 1976.

5/15/76 - 5/19/76

This 60-hour outage was for installation of restraints on the secondary plant due to vibration and water hammer experienced during the first attempt to place the Moisture Separator/Reheaters in operation. The restraints prevented plant damage until sufficient operating experience allowed formulating and implementing permanent corrective action which has been satisfactorily completed. No major corrective safety related maintenance was performed.

6/12/76 - 6/19/76

This 7-day shutdown was to clear up several problems, mostly in the secondary plant, in anticipation of power ascension to 80% and beyond. Major safety related corrective maintenance consisted of: repairing the pressurizer spray valves (leaking through as discs had loosened from the stems); replacing the Auxiliary FWP crossconnect valve stems; repacking the SDC loop isolation valves, the power operated relief valves and 1B MSIV; replacing Cell #31 in 1B battery; and performing PC/M 116 to reduce the RPS temperature indication noise.

7/9/76 - 12/10/76

This 155-day shutdown was for fuel poison pin replacement (fuel reconstitution). See PC/M's 192-76 and 176-76 with attached discussion for details. Other major safety related items were: repair of 1C ICW pump bearings, repair of two small leaks in ICW lines and replacing the motor on V-2501, Volume Control Tank outlet isolation valve.

Following is a summary of other Safety Related corrective maintenance.

MECHANICAL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
1 A CCW ICW Strainer	0187	High ΔP	Cleaned
Fuel Pool Purif. V07188	0194	Bonnet leak	Replaced gasket
B Diesel Gen. Air Start Sys. Lubricator	0201	Low oil flow	Adjusted
B Diesel Lube Oil Pump	0211	Oil leak	Replaced seal
B Charging Pump	0213	Seal Leak	Replaced plungers & packing
1 B 1 Diesel Gen. Lube Oil Pump	0214	Cracked brg. housing	Replaced housing
A Diesel Gen. Air Comp. (Diesel)	0218	Ran out of fuel	Prime fuel system
A Diesel Gen. Air Start Sys. Valve	0284	Faulty Air Relay valve	Cleaned
A CCW P Inboard Seal	0298	Seal leaks	Replaced seal
Control Rm HVA-3A, -3B, -3C	0313	Dust filters dirty	Changed filters
A Diesel Gen. Fuel Filters	0315	High ΔP	Changed filters
A Charging Pump Disch. RV2326	0320	Plug leaking	Replace gasket
B Cont. Spray Pump Suct. V7124	0321	Bonnet leak	Replaced gasket
R.A.B. SAFSDS Rm. Drain V25-6	0322	Packing leak	Packing tightened
1C Gas Decay Tank Disch V6703	0323	Leaks past seat	Replace diaphgram
A Emerg. Diesel Fuel Transfer Pump	0328	Packing leak	Repacked
A CCW Hx. ICW Strainer	0333	High ΔP	Cleaned
Przr. Spray Bypass V1236	0380	Packing leak	Repacked
Przr. Spray Bypass V1236	0381	Packing leak	Repacked
B I.C.W. Pump Piping	0432	Seal Water Line Plugged	Cleaned
B Diesel Gen. Cooling RV	0450	Leak Past seat	Removed, reset, replaced
B CCW Hx ICW Strainer	0456	Gasket leak	Replaced gasket
B CCW Pump Inbrd Brg.	0461	Oil leak	Replaced gasket
Fuel Pool Purif. Pump Suction V07170	0466	Body to bonnet leak	Replaced gasket
Letdown LCV-2110Q	0468	Body to bonnet leak	Replaced gasket & seal
A Cont. Sump Check V07174	0479	Leaks past seat	Cleaned seats
Cont. Escape Hatch	0482	Various loose bolts in operating mechanism	Tighten and stake threads
B ICW Pump	0483	Packing leak	Repacked
Aux. HPSI Hdr. to CVCS, V-2340	0486	Bonnet leak	Replaced gasket
ACC 3 A, B, C Control Room Air Cond's.	0490	Condenser cooling air dampers seized	Freed and lubed
C ICW Pump	0492	Bearings failed	Installed spare pump
Pressurizer Vent V-1239	0493	Valve leaks past seat	New plug & stem - lapped

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Containment Personnel Door	3273	Misalignment - hard to operate	Adjusted
Pressurizer Valves 1436 & 1440	3289	Bonnet leaks	Tighten bonnet
1A Aux. Feed Pump Valve MV-9	3290	Bonnet leak	Replaced bonnet gasket
RCP Bleed Off to Quench Tank R.V.2199	3293	Leaks past seat	Remove,relap, retest, replace
RV 3483 to Hold Up Tank	3294	Flange leaks	Replaced gasket
1A Purif.I Ex. V2380, V2372	3465	Valves not closing	Replaced diaphragm in air operator
N2 to Gas Comp. V6059	3504	Body to bonnet leak	Tighten bonnet
RCP Controlled Bleed Off V6107	3508	Valve leaks thru	Relapped seat
1A Waste Gas Comp. V6573	3522	Valve leaks thru	Adjusted stem stop nuts
1A Charging Pump Coupling	3523	Excessive Grease	Removed excess grease
1C HPSI Inboard Seal	3524	Leaks	Replaced
Regen Hx V2810	3559	Valve leaks thru	Relapped seat
1B & C HPSI Pump	3570	Seal cooling lines leak	Tightened unions
A Loop S.D.C. Isolation V3480	3580	Packing leak	Repacked
Letdown Iso. Valve 2515	3582	Packing leak	Tightened packing
1A Shutdown Hx RV3431	3593	Leaks past seat	Remove,rework, retest, replace
Letdown Hx RV2345	3595	Leaks past seat	Remove, relap, retest, replace
Gas Decay Tank V 6592 - 6701	3605	Stems pulled loose	Replaced diaphragms
C.V.C.S. PCV 2201	3606	Packing leak	Repacked valve
B I.C.W. Pump	3609	Packing leak	Adjusted packing
Aux. Feed Pump Cross Connect MV09-14	3610	Stem damaged during MOV test	Replaced stem
1 C Gas Decay Tank V6597	3614	Leaks past seat	Replaced diaphragm
C.V.C.S. Let Dwn L.C.V. 2110P	3620	Packing leaks	Repacked valve
C.V.C.S. Let Dwn L.C.V. 2110 Q	3621	Packing leaks	Repacked valve
B Gas Decay Tank V6578	3635	Leaks past seat	Replace diaphragm
1B Charging Pump Vent V2805	3638	Leaks past seat	Installed blank flange
B.A. Sys. F.C.V. 2161	3643	Packing leak	Tightened packing
1 A Charging Pump	3646	Plunger packing leak	Replaced center plunger-repacked
1 A BA Make-Up Pump	3683	Seal leak	Replaced seal
Przr. Spray V1100E	3685	Bonnet leak	Replaced gasket
1 A Charging Pump	3699	Brass Chips in brg. housing	Cleaned and inspected o.k.
1 B Emer. Diesel Coolant Tk. RV	3705	Leaks past seat	Lapped and reset
CVCS Letdown LCV2110P	3720	Bonnet leak	Replaced gasket
1 A MSIV Check Hinge Pin	3730	Steam leak on hinge pin cover	Repaired by Furmanite process
B SG Blowdown Orifice Flng.	3815	Steam leak	Replaced flexitallic gaskets

MECHANICAL CORRECTIVE MAINTENANCE CN SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Flash Tank Gas Vent Trap	3820	Trap leaks past	Cleaned internals
1 B Aux. Feed Pump Piping	3878	Pipe plug leak	Tightened
CVCS Letdown LCV2110Q	3892	Steam leak	Repacked
CVCS Letdown LCV2110P	4002	Bonnet and packing leak	Replaced gasket & repacked
CVCS Letdown PCV 2201P	4015	Bonnet leak	Replaced gasket
1 C Gas Decay Tank V6597	4020	Diaphragm leak	Replaced diaphragm
Flash Tk. Gas Vent T6909	4030	Trap leaks past	Replaced plug & seat
Cont. Personnel Hatch	4042	Loose pin in operating mechanism	Replaced pin
1 A Diesel 12 Cyl. Starter Valve	4070	Air start supp. valve	Cleaned
MSIV Bypass I-MV-08-1B	4082	Stem broken	Replaced
MSIV Bypass I-MV-08-1A	4083	Stem broken	Replaced
Przr. Relief Iso. V1403	4092	Packing leak	Repacked & new gasket
Przr. Relief Iso. V1405	4093	Packing leak	Repacked
1B MSIV HCV-08-1B	4097	Steam leak	Repacked
Aux. Feed Pump Cross Connect MV-09-14	4098	Damaged stem during MOV test	Replaced stem
Aux. Feed Pump Cross Connect MV-09-13	4099	Damaged stem during MOV test	Replaced stem
1 B Hot Leg S.D.C. V3651	4100	Packing leak	Repacked
1 B Hot Leg S.D.C. V3652	4101	Packing leak	Repacked
Steam Flow Tx FT-08-1B Iso. V8135	4105	Packing leak	Repacked
Steam Flow Tx FT-08-1B Iso. V8136	4106	Packing leak	Repacked
Steam Flow Tx FT-08-1B Iso. V8137	4107	Packing leak	Repacked
1 B Charging Disch. R V2325	4125	Leaking plug	Inspected & cleaned
Steam Flow Tx. FT-08-1B Root V8134	4126	Packing leak	Repacked
Przr. Condensate Pot. Iso. V1437	4129	Steam Leak	Tightened plug
Gas Decay Tank V6592	4141	Stripped stem	Replaced diaphragm
C Charging Pump Seals	4145	Seals leaking	Replaced plungers & packing
A Charging Pump Seals	4161	Seals leaking	Replaced plungers & packing
1 A Charging Pump Disch. R V2326	4186	Leaks past seat	Removed, lapped, retested, replaced
Przr. Safety V1201	4187	Leaks past seat	Removed, relapped, retested, replaced.
A Loop S.D.C. V3470	4193	Packing leak	Repacked
A Loop S.C.D. V3481	4194	Packing leak	Repacked
B CCW/ICW V21250	4244	Broken weld	Threaded (temp fix) - See PWO 712
Control Room Vent Fan HVA 3C (Damper)	4250	Seized damper	Freed and lubed
1 A CCW Pump Ind. Brg.	4260	Excessive oil usage	Inspected, replaced seal - PWO 4621
1 A CCW Pump Otbrd Seal	4261	Seal leak	Replaced seal
BA Makeup Relief RV2141	4276	Defective stem	Replaced stem
BA Makeup Relief RV2133	4277	Defective stem	Replaced stem
Przr. Spray Bypass V1236	4281	Packing leak	Repacked

MECHANICAL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Przr. Spray Bypass V1237	4282	Packing leak	Repacked
B.A. Make-Up Sample Sys. A V2128	4407	Seat leak	Cleaned - o.k.
1 A MSIV	4409	Steam Leak at plug	Repacked
1 B Charging Pump Packing	4413	Packing leak	Replaced plungers & packing
1B1 SI Tank V3631	4422	Body to bonnet leak	Retorqued bonnet bolts
Sample Ht. Ex. Iso V1211	4426	Body to bonnet leak	Replaced bonnet gasket
ACC. 3A, B, C Control Room A/C's	4600	Condenser cooling air dampers seized	Freed-up & lubed
1 A Charging Pump Disch. R V2315	4602	Leaks past seat	Removed, relapped, reset, replaced
V.C.T. Piping Gasket	4647	Bad gasket - gas leak	Replaced gasket
V.C.T. Piping Gasket	4649	Bad gasket-gas leak	Replaced gasket
1 A G.D.T. Inlet V6584	4650	Leaks past seat	Replaced diaphragm
V.C.T. Inlet Chk. V2112	0002	Bonnet leak	Replaced gasket
C I.C.W. Pump	0003	Packing leak	Adjusted packing
1B ICW Pump Lube Water Str. Valve	0004	Leaked thru badly	Replaced valve
HVS-1C Cont. Cooling Fan	0007	Bad bearing	Replaced
Przr. Spray 1100E	0009	Leaks thru seat	Checked - o.k.
CCW Hx. Salt Water Str. V21339,21334	0010	Not closing properly	Repaired - adjusted
A CCW Hx Salt Water Str	0011	Strainer plugged	Clean
Cont. Per Hatch Outer	0012	Gasket leaks	Replaced
1 A LPSI Suction R V3483	0018	Leaks past seat	Removed, retested, replaced
1 B LPSI Suction R V3468	0019	Leaks past seat	Machine and lap seat
Przr. Quench Tank Rupture Disc.	0024	Damaged	Replaced
Diesel Gen. Lube Oil Pump 1A2	0027	Noisy bearing	Realigned
Diesel Gen. Lube Oil Pump 1B1	0029	Noisy bearing	Realigned
Przr. Spray V1100F	0030	Leaks past seat	Found loose seat - repaired
A Waste Gas Comp.	0039	Leaking Diaphragm	Replaced
B.A. Make-Up Strainer S2903	0061	Flange leak	Cleaned, replaced flange
B CCW I.C.W. Strainer 1B	0063	High ΔP	Cleaned
C Charging Pump Hdr V2504	0166	Leak under lagging	Replaced bonnet gasket
Cont. Personnel Hatch Mech. Dr. Shaft	0171	Misaligned bearing	Realigned flange bearing
Gas Decay Tank Valve 6579	0174	Leaks past seat	Replaced diaphragm
A Diesel Gen. Air Start Sys. Lubricator	0176	Low oil flow	Adjusted
1 C CCW Pump	0178	Seals leak	Replace seals

MECHANICAL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
SD Hx Relief V3431	0498	Leaking plug	Removed, cleaned, replaced
Radiation Monitoring Sys. FCV-26-02,04	0499	Leak past seats	Cleaned seats
B Diesel Gen. Air Comp (Diesel)	0501	Will not run	Primed fuel system
B Diesel Gen. Fuel Transfer Pump	0502	Packing leak	Repacked pump
C ICW Pump	0505	Packing leak	Repacked pump
A Charging Pump Disch. RV2315	0514	Body to nozzle leak	Replaced gasket & reset
H2 Return Chk. VI-27102	0515	Leak past seat	Cleaned seats
Primary Water Chk V-15328	0516	Leak past seat	Machined and lapped
S.I.T. Test Line VI-07009	0518	Packing leak	Repacked
B CCW Hx. ICW Strainer	0520	High D.P.	Cleaned
Letdoyn LCV2110P	0525	Bonnet Seal leak	Replaced seal & gasket
Aux. FW Crossconnect Valve, HV-09-13	0527	Bonnet leak	Replaced gasket
Cont. VacuumRelief, V-2520	0661	Springs damaged while installing leak test flange	Replaced springs
A CCW Hx Salt Water Disch. Line	0663	Requires inspection by Engineering	Opened line and cleaned
A CCW Hx Salt Water Disch. Line	0664	Branch connection leaks	Removed for repair (See PWO 678)
A CCW Hx Salt Water Disch Line	0666	Line cracked	Welded crack
A CCW Hx Salt Water Disch Line	0667	Connection leaks	Welded in new sockolet
1A LPSI Pump Suction Relief RV-3483	0668	Body to nozzle leak	Installed gasket
Equip. Drain Tank Inlet Strainer	0669	Plugged	Cleaned
MSIV Bypass MV-08-1A	0676	Improper torque	Reset
A CCW Hx Salt Water Disch Line	0678	Branch connection leaks	Installed new sockolet
B CCW Hx	0682	Inspected & cleaned	Cleaned
A CCW Hx Salt Water Strainer	0684	High ΔP & loose bolts	Cleaned, replaced bolts
1 B Diesels Air Start Relay V's	0690	Suspect dirt	Cleaned
B CCW Hx. Salt Water Strainer	0696	High ΔP	Cleaned
A BA Make-Up Tank RV2132 Flange	0699	Flange leak	Replaced gasket
B I CWP Disch. Chk. V21208	0700	Flange leak	Tighten bolts
B CCW Hx Salt Water Disch Line	0703	Line leak	Removed for repair (See 0712)
Cont. Personnel Hatch	0709	Shaft bent in operating mechanism	Straightened shaft
B CCW Hx. Salt Water Disch. Line	0712	Hole in pipe	Welded (See 0703)
ICW Pump Disch. Hdr.	0713	Hole in pipe	Welded
Letdown LCV2110Q	0714	Bonnet leak	Replaced seal ring

MECHANICAL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
B CCW Hx. Salt Water Line	0717	Pin hole leaks	Welded
1 C Aux. Feed Pump Turbine	0718	Low oil flow	Change filter
1 C ICW Pump	0722	Seized	Replaced bearings
A & B SDC Loop Inst. Iso. Valves	0731	Packing leaks	Repacked
A Charging Pump Disch. RV2326	0734	Relieves at too low a setting	Removed, inspected, retested
B Charging Pump Disch RV 2326	0735	Leaks through	Reset, retested
Przr. Spray V1100E	0707	Stem pulled from plug	Replaced pin
C ICWP "Y" Strainers	0814	Low flow of lube water	Cleaned
Przr Spray Valve V-1100E	0815	Packing leak	Repacked
Letdown Hx Inlet RV2346	0817	Lifts early	Replaced valve
B Charging Pump Packing	0843	Packing leak	Replaced packing & plungers
Charging Line to 1B1 Loop	0898	Blind flange leak on V2805	Tighten flange
CEDM 13,44	NA (by CE)	44 didn't operate when cold 13 showed signs of possible failure (didn't withdraw properly on last startup)	Replaced, retested

ELECTRICAL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Diesel Gen. 1A Annunciator (Ann.)	4122	Spurious Alarm	Replaced relay
1A CCW Pump Ann. S-51	4301	No isolate switch alarm	Replaced lifted leads
1A Aux. FWP	3324	In/Bd Bearing heating up	Replaced motor filters
1B CCW Pump	3971	Sealtite conduit broken	Repaired conduit
1B HPSI Pump	3395	Overcurrent relay not functioning	Adjusted contacts
1A & 1B LPSI Breakers	4015	Isolate switch deleted from design	Lifted lead to agree with latest CWD revision
MV-09-13 Aux.. FW Header Cross-Connect	4036	Indicates both open and closed	Replaced Beleview washer in operator
1A Charging Pump	3915	Motor leaking oil	Repaired piping
SDC Isolation Valves MV-3480/3481	3387	Received spurious "not open" alarm	Corrected wiring
Shield Bldg. OutsideAir Supply Valve FCV-25-11	3989	No open indicating light	Adjusted limit switch
MOV 3617 (HPSI)	2754	Damaged conduit	Repaired conduit
1A Battery Charger	3391	Amp meter indicating low	Adjusted charger voltage
1A Battery Charger	3320	Spurious alarm	Replaced phase sequence relay
1A Battery Charger	4307	No amp meter indication	Change circuit bd. MBC 1971
MV-3656 HPSI	2765	Spurious valve not open alarm	Corrected wiring to agree with CWD
MV-21-3 ICW Non-Emerg. Header Isol.	3380	Won't open & close	Adjusted operator clutch mechanism
Aux. HPSI/Loop 1B1 HCV 3737	3396	Opens beyond throttling setting	Adjusted limit switch
SI Tank Disch. Valve 1A1 MV3624	3381	Won't operate	Replaced fuse
MV 07-1A RWT Outlet	2253	Defective torque switch	Replaced switch
Containment Sump	4009	Valve won't close fully	Reset torque switch
Boron Control Valve 2525	3957	Overload tripped	Repaired limit switch
MV 2504 B.A. Make-Up	2775	Indicator Wrong	Adjusted indicator
Volume Control Tank Iso. V-2501	4063	Motor bad - excessive cycling	Replaced motor (See PC/M 181-76)
VC Tank Discharge V2501	3322	Overload tripped	Reset (See PC/M 181-76)
1B Rx Sump Pump	4311	Penetration bad electrically	Move to spare penetration
MV 3645 RTGB Indicator - HPSI	3389	Position ind. defective	Repaired wire in meter
B Battery Charger	3314	Spurious alarm	Replaced phase failure relay
Control Rm A/C ACC3C	3949	Disconnect switch bad	Replaced switch
Aux. FWP 1C	3343	Tripped (overspeed)	Repaired wiring
MV09-11 Aux. FW Pump 1C Disch.	3917	Won't close	Cleaned contacts
1B Battery	3988	Cell No. 31 bad	Replaced cell
1A LPSI Pump	4109	Leaking oil sight glass motor	Replaced sight glass
1C Aux. FW Pump Iso. MV08-3	4282	Valve starts open - then stops	Replaced torque switch
Aux. FWP MV09-11	4293	Valve won't close	Removed sand from torque SW
B.A. Heat Trace CVCS	4131	Heater off	Repaired shorted section
B.A. Heat Trace CVCS	3333	Grounded	Repaired grounded section
B.A. Make-Up Pump	3907	Low temp. alarm (spurious bad thermocouple)	Repaired thermocouple
1C Aux. FWP	4142	Control junction box has excessive moisture	Sealed control box
1C Aux. FWP Steam Valve MV08-3	3346	Will not open	Repaired wiring lug

ELECTRICAL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Brkr 60308 Aux. FWP 1C	2789	Ground	Removed jumper to make circuit agree with CWD
1C Inverter	3960	Won't take load	Chance oscillator board
1C Inverter	3996	Low voltage alarm (spurious)	Adjusted alarm relay
DC Bus 1A	3321	Ground on Ckt D114	Repaired
1C Aux. FWP	4317	Solenoid stuck	Lubricated latch
1C Inverter	4094	No AC output	Replaced burnt wiring
120 VAC Vital Bus Inverter	3950	Tripped	Replaced blown fuse
120 VAC Instrument Bus "MC" Inverter	4303	Hi Voltage Alarm	Adjusted contact on relay
Charging Pump Seal Lube Pump 1B	3397	Motor won't run	Replaced motor bearings (See PC/M 75-76)
Charging Pump Seal Lube Pump 1B	4008	Motor won't run	Replaced motor (See PC/M 75-76)

INSTRUMENT & CONTROL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Nuclear Instrumentation Detectors	3083	Incorrect detector high voltage	Reset & measured the voltage on wide range and linear power channels
Remote Wide Range Start Up Rate Meter	3084	Ch. "B" not responding to input signal	Replaced transistor and op. amp. in sigma meter
Nuclear Instrumentation	3693	Wide range "A" failed functional test	Period meter adjustment pot. was adjusted
Nuclear Instrumentation	4035	Inoperative rate ckt. on Ch. "C" wide range	Reset pretrip set point-rechecked period ckts.
Wide Range Nuc. Pwr. Ch. "B"	4059	Indicator inaccurate	Calibrated indicator "B"
NI Channel C	4256	SUR pretrip and DPM appeared to be off setpoint	Checked set point - no abnormal deflections
Nuclear Instrumentation	4463	Replace feedback resistor.	Reset extended wide range bi-stable-replaced resistors on all four channels
NI	4476	Readjust linear amplifier gains	Readjusted upper & lower gains for Ch. B,C & 10
NI	6039	Replace all trip test pots. on linear pwr drawers	Replaced pots and resistors in all drawers
NI	6055	Replace R-7 & R-26 on all linear pwr drawers	Replaced pots and adjusted the output on R-7
Wide Range Log Channels	6143	Ch. A & B off on cal. check	Adjusted DPM meter and bi-stable
NI	6151	Linear pwr. control Ch. #9 reads 2% pwr @ no voltage input	Adjusted span resistor, verified proper readings
Wide Range Ch. "C"	6205	All calibrate positions read high	Adjusted and calibrated
Wide Range Ch. "C"	6512	Repair faulty high voltage connector	Replaced bad male connector & ray chem heat shrink tubing
Pressurizer Temperature	3278	Failed RTD - requires replacement	Installed spare RTD
RCS - 1B1 Cold Leg Temp. Element	3288	Erratic signal from detector TE 1125	Replaced RTD - calibrated
Pressurizer, HIC-1100 Spray Valve Control	3533	Continuous open signal to spray valves	Adjusted lower limit on controller
Stm. Gen. ΔP Transmitters	3558	Valves leaking PDT-1111 A,B,C	Replaced valves and leak checked
RCP 1B1 Oil Lift Pumps	3591	1B1 oil lift pumps running in auto (speed switch out)	Adjust sensitivity to provide proper ckt. action
Pressurizer Level Control	3735	LIC 1110Y does not give indication	Replaced deviation amp, cleaned meter, calibrated
RCP 1A1	3848	TIA 1159 & 1158 inputs are reversed	Reversed elements-verified normal indication
RCP 1A2 Upper Seal	4009	Low alarm failed PIA 1162 & 1173	Installed retro-fit kit & re-calibrated
Pressurizer Spray 1B2	4017	Temp. output varying causing alarm	Tightened lugs on RTD-Satisfactory performance
RCS H1 Cold Leg Temp	4024	Alarm is set too high	Replaced temp. indicator & recal. of setpoint
RCP 1B2	4056	Spurious hi vibration alarm (Ann. J-28)	Adjusted sensitivity of switch
RCS ΔT Power Ch. "D"	4089	ΔT Power signal is abnormal	Installed pwr supply-checked against Ch. A,B & C
Quench Tank Pressure	4109	Failed to alarm at setpoint	Replaced blown fuse
Containment RTD's	4119	Check for loose leads at RTD's	Tightened leads at containment penetration
RCS Cold Leg Temperature	4142	Different readings on temp indicators	Calibrated sigma temp. indicator
RCS Cold Leg Temperature on Hot Shut-down Control Panel	4458	Out of calibration specs.	Failed calibration - replaced with spare

INSTRUMENT & CONTROL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
RCS Pressure Low Low Setpoint	4462	Press. ind. will not allow SDC valves to open	Replaced press. ind. controllers-calibrated & installed. Verified indications agreed.
RCS-PDI 1124X	4468	Failed specs during RCS flow test	Replaced servo meter & calibrated meter
RCS-PDI 1112	4469	Failed specs during RCS flow test	Meter calibrated
RCS Loop 1A Cold Leg Temp.	4473	TIC-1111Y is non linear & out of tolerance	Replaced scale & calibrated-reinstalled meter
RCS Loop 1A Cold Leg Temp.	4480	Transmitter output cycles to max & locks in	Replaced circuit board and recalibrated unit
Quench Tank	6030	Check cal. on quench tk instrumentation	Adjusted zero on PT-1116-replaced oscillator on CT-1116
RCS-PDT 1121A	6040	Bad zenor diode	Replaced diode and tested unit
RCS Transmitter Inst. Valves	6369	Leaking past packing	Inspected & repacked around A & B S/G
DC Power Supplies	6376	Change wiring to agree with CWD	Ran new wire
RCS-PDT-1121D (Loop ΔP)	6385	Repair transmitter	Could not be repaired-returned to factory
Pressurizer Pressure	6470	PIC 1100Y Indicating incorrect	Installed replacement oscillator amplifier
Pressurizer Spray Valve 1100E	6479	Remove & replace instrumentation for valve rpr	Replaced instrumentation & stroked valve
Quench Tk. Level Indicator	6483	Level fluctuates between 42% & 58%	Replaced faulty meter with calibrated spare
Pressurizer Level Control LIC 1110X	6489	Output meter sticks	Rebuilt output meter and reinstalled
Engineered Safeguards Cabinet Mod.	3133	Isolation module wiring changes	Performed work per PC/M #31-76
Engineered Safeguards	3549	Actuation module SB cab. trip light dim	Corrected bad solder joint
Engineered Safeguards	3550	ATI przr. press. function not resetting	Repaired bad connection & module in SIAS
Actuation Module (safeguards)	3575	Module does not reset	Replaced IC V3 - retested satisfactory
ATI Module (safeguards)	3576	Does not sequence through logic check	Replaced IC V1 - retested satisfactory
Low Ptzr. Pressure Trip Ann.	3654	Low press. trip comes in before pre-trip alarm	Made setpoint change per CE letter F-SP-835
ESC-MA Bi-stable Module	4065	Setpoint dial pot. has broken locking device	Replaced pot. & readjusted bi-stable
ESC Cab. "A"	6077	Current adjust for test ckt. erratic	Replaced pot. - tested satisfactory
ESC Przr. Press. Meter PIA-1102 All	6353	Low alarm is in-operative	Installed retro-fit kit - checked cal & alarm
RPS - Stm. Gen. Level	3255	LT-9013B Setpoint drift	Installed retro-fit kit - cal'd. satisfactory
RPS CIP Panel	3687	T Hot Digital readout too high	Repaired loose leads at TE 1112 HB
RPS Indicating Relay	3589	PIS-07-2A Pegged Down scale	Repaired bad torque switch-test satisfactory
RPS Adder & Multiplier Modules	3659	Voltage greater than 10 MV	Replaced modules - tested satisfactory
RPS Trip Unit Ch. "D"	3845	Resistors overheated	Replaced and retested
RPS-TMLP Ch. "A"	3870	TMLP Bi-stable would not open	Reworked module-re-test operation satisfactory
RPS - Power Supply Ch. "D"	3871	-15 V failed to -21.00 V	Adjusted power supply
RPS Cabinet "D"	4043	Wires have wrong polarity to metro scope	Reversed leads

INSTRUMENT & CONTROL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
RPS	4124	Noise on instrument loops (Check)	Checked for grounds on lifted shields
RPS T Hot Ch. "D"	4130	Temp. transmitters erratic	Removed grounded lead on TE-1122 HD
RPS	4248	Readjust core protection calculator setpoints	Adjusted coefficients-Set pwr. ratio calculator
RPS Ch. "A"	4251	Comparator module failed	Replaced defective diode
RPS Cab. "D"	4451	Positive differential alarm is inop.	Installed retro-fit kits-cal'd-adj. set-points.
Reactor Protection Syst. Cabinet "A"	4488	Nuclear Pwr. ΔT pwr. % meter failed	Installed retro-fit kit and calibrated
Reactor Protection Syst.	6046	Relay socket on AD2 logic matrix relay	Replaced socket - resoldered shields
Reactor Protection Syst.	6206	S/G low press. Bi-stable will not activate	Installed spare bi-stable-checked trips
Reactor Protection Syst.	6312	TT-1122 HA Produces no output	Replaced zener diode & cal'd within specs.
RPS Ch. "B"	6284	CPC-1 G-2 module exhibits 10 MV offset	Replaced module and retested
RPS Ch. HA TH/LP	6347	Hi alarm inoperative	Installed retro fit kits, cal'd & set alarms
RPS Wide Range Ch. C	6366	DPM Meter indicating incorrect	Replaced meter and re-cal'd.
RPS Wide Range Ch. C	6377	Hi start-up rate trip setpoint is above limits	Adjusted setting on trip and pre-trip
RPS Cab. C	6384	Repair 15V power supply C-1	Resoldered leads-checked
Control Element Drive Motors (CEDM)	3527	Replace defective reed switch ass'y.	Replaced two defective switches with spares
CEDM CCP Timer Modules	3530	Replace Defective diodes in 12V pwr. supply	Replaced diodes and tested
Control Element Drive System (CEDS)	3727	CEA #33 intermittently drops	rearranged wire to eliminate noise problem
CEDM	3859	Lower limit light on Rod #18 did not clear	Removed position transmitter & replaced with spare
CEDS CPP	4120	Repair damaged connector sockets	Removed modules-replaced damaged sockets
CEDM CCP Timer Modules	4492	Replace 12V pwr. supply diodes	Performed per CE Letter F-SF-920
CEA #50	6047	Dropped - cannot retrieve	Replaced 15V power supply
CEDM and Reed Switches	6092	Replace damaged connectors	Replaced connectors per procedure 8770-8202 R-0
CEDM	6093	#10 blows fuses-#13 will not raise correctly	Located grounded pin-replaced damaged cable
CEDM Coil Stacks	6289	Remove for repair (position indication)	Repaired coil stacks & reed switches
CEDS Control System	6404	Troubleshoot & repair in support of start up	Replaced CEA module #7 & 34 - Timer Module #55 & 61
CEA	3163	Adjust reed switch position transmitters	Measured resistance & adjusted (8770-6947)
CEA #29	3704	Lower electrical limit switch shorted	Replaced defective switch-tested Ops. Proc. 0110081
CEA #31 Reed Switch	3748	Improper positions indications & actuations	Soldered broken wire in position transmitter
Boric Acid Make Up Tank 1A	3191	Low low level alarm failed	Install retro-fit kit & cal'd. LIA 2206
Charging Pumps Header Pressure	3640	PT-2212 Internal leak	Replaced bourdon tube & re cal'd.
Charging Pumps Header Pressure	3657	PT-2212 leaking	Replaced with spare-installed snubber-re cal'd.
1A Charging Pump	3690	Seal Water Tank Level Indication Inoperative	Freed float in the indicating assembly
Volume Control Tank	6002	Pressure Regulator not working correctly	Reset regulator
Boric Acid Make-up Tank 1B	6089	Level indicator failed high	Disconnect fittings-cleaned flow meter-reconnected
1A & 1B BAM Tanks	6140	Reset low level alarm	Installed retro-fit kits-reset alarms
1B BAM Tank	6236	Recirc. valve indicates intermediate position	Adjusted limit switch
1A BAM Tank LIT/LIA 2206	6418	Level transmitter air line plugged	Flushed lines-returned to service
BAM Isolation Valve	6459	No indication of valve position	Snap lock switch sticking-freed arm

INSTRUMENT & CONTROL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Volume Control Tank	6509	Low alarm not functioning	Wiring error on sigma contact-corrected
"A" Charging Pump	6523	Low level alarm on seal tank	Corrected indicator alignment
Volume Control Tank	6524	Low level alarm not operating	Replace sigma instrument with calibrated spare.
Safety Injection Tk. 1B1	3146	Abnormal fluctuation on LIA-3331	Replaced with spare - recalibrated
S.I. Tank 1B1	3564	High-high level alarm.	Vented transmitter LT-3331
S.I. Tank 1A2 & 1B1	3565	High and low level alarms	Vented transmitters-readjusted setpoint on LIA-3331
S.I. Check Valve Leakage to RWT	3566	Valve leaking by seat.	Re-zeroed valve positioner
Shutdown Heat Exch. 1B	4484	Temperature indication incorrect	Replaced servo-motor-re-calibrated
Safety Injection Tank 1B2	6497	High-high level alarm	Adjusted transmitter-zero
Gas Decay Tank	3554	Flow indicator incorrect FIT-6648	Removed unit-cal'd. spare & installed
Waste Gas Compressor	3599	Setpoints need adjustment	Adjusted PS-6647 2 & 3
Gas Decay Tank	4025	Flow meter gives inaccurate flow rate	Added 3/8" orifice in gas release line
Waste Gas Comp. "A"	6495	Not cycling properly	Re-calibrate pressure switches
Waste Cond. Tank LIC-6640 & 6641	6290	Cannot pump tanks lower than 22%	Checked calibration & verified setpoint actuation
Containment Spray FCV 07-1B	3706	Valve indicates closed when open	Adjusted limit switch & valve stem collar
Reactor Containment Bldg. Pressure	6026	PIS-07-2D oscillates	Tightened all terminal screws
Reactor Containment Bldg.	6357	PIS-07-2A Servo sticks	Replaced servo - ca'ld. & adjusted setpoints
Reactor Containment Bldg.	6364	PIS is connected wrong for lower alarm	Corrected wiring
Main Steam Dump Valve	3632	PIC-08-1B1 reads above indication	Repaired and re-installed
1A Main Stm. Isolation Valve	3671	Diaphragm leaking	Replaced diaphragm and tested
1A Main Stm. Isolation Valve	3672	Solenoid light stays on	Repaired switch arm & tested
1B Main Stm. Isolation Valve	4032	Ground in controls	Ground in LS-10 - Replaced switch
"A" Main Stm. Dump Valve	4108	PIC-08-1A Setpoint problem	Balanced controller to hold setpoint input
1B Main Stm. Dump	6122	PIC-08-1B1 indicates too high	Replaced capacitor & re cal'd. unit
Main Stm. Pressure	6235	PT-08-1A Output saturates up scale	Checked mech. linkage-replaced detector coil
Main Stm. Pressure	6476	PT-08-1B fails intermittently	Replaced with calibrated spare transmitter
Feedwater Reg. ByPass S/G 1B	3115	LIC-9006 inoperative	Corrected remote & local process linearly problems
Stm. Gen. 1A Level	3251	Unable to determine actual level	Filled ref. legs on transmitters & vented D/P blocks
"B" Aux. Feed Pump	3510	No flow indication with pump in operation	Replaced transmitter force motor & recalibrated
"C" Aux. Feed Pump	3590	Control problem-overspeeds when started	Terminal connection on servo actuator loose
Aux. Feedwater Flow Header "B"	3738	Gauge reading zero with flow through header	Reset zero to 4 ma DC & returned to service
1A Main Feedwater Flow	4118	Indication failure	Replaced PE-09-1A1- Cal'd - returned to service
S/G Level "B"	4192	Level reads high on L-9023 A & L-9021	Tightened fittings & valve stems
Feedwater ByPass Control	4490	LIC-9006 has erratic operation	Replaced resistor & adjusted-stroked valve
1B B.A. Makeup Tank	6469	Spurious level alarms	Replaced remote amplifier, calibrated

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Stm. Gen. 1B Level	6370	Hi level controller actuation inoperative	Installed mod. kit-cal'd. & re-installed
Stm. Gen. 1A Level	6371	Hi level controller actuation inoperative	Installed retro kit-cal'd. & re-installed
Stm. Gen. 1A Level	6372	Transmitter out of cal.	Removed,aligned & cal'd., replaced & tested
1C Aux. Feedwater Pump	6387	Controls are rusty and full of water	Cleaned, removed moisture, lubricated
Stm. Gen. 1B Level	6468	Incorrect level indication on LIC-9023C	Vented instrument & tightened equalizer valve
Stm. Gen. 1B Level	6484	Hi controller contact is on constantly	Installed retro kit and calibrated-tested
Stm. Gen. 1A Level	6521	Hi level alarm with normal level	Installed retro-kit and cal'd - tested
CCW HX-1B Outlet Pressure	3098	PIS 14-8B indication is erroneous	Replaced force motor in transmitter & cal'd
CCW HX-1B Outlet Pressure	3887	PIS 14-8B Indication is erroneous	Replaced force motor and re-cal'd.
CCW From Letdown HX Flow	6390	Flow indication on FIS 14-6 & valve is closed	Replaced oscillator amplifier
CCW Header B Flow	6403	Hi/Lo Alarm on with normal flow	Installed retro-fit kit and cal'd.
Shutdown Heat Exch. CCW Outlet 1B (HCV-14-3B)	6450	Air supply line to valve broken	Repaired broken line
Primary Water Valve to Containment	4271	Annunciator for HV15-1 not working	Checked & installed arming screw
1A Emergency Diesel	6356	Starting air low press alarm won't clear	Switch sticking- replaced-functionally checked
Intake Water Level	3574	Alarm does not actuate on low level test	Recalibrated
ICW Disch. Header A	3691	Transmitter failed	Replaced internal parts with spares
Intake CW Disch. Header A & B	6065	No bleed off valves between isolation valves and press. transmitter	Installed valves as req'd per Tech Specs
CCW Hx Inlet D/P Ann.	6088	Doesn't alarm	D/P Indicator beyond repair-replaced indicator
Intake Cooling Water Pump A	6303	Flow indicator inaccurate FI-21-3A	Removed, cleaned & repair flow meter
CCW Ht. Exch. 1B	6367	Spurious Low Flow Alarm FIS 21-9B	Lifted leads, cleaned corrosion, alarm cleared
Intake Cooling Pump Lube Water "C"	6378	Failed Low Flow Alarm PS 21-46	Cleaned switch, operation satisfactory
CCW Ht. Exch. 1B	6480	Barton pegged low FIS-21-9B	Contacts closed - replaced switch
S-G 1A Blowdown PS-23-6	4481	Press. Switch has ruptured bellows	Installed new switch-tested satisfactorily
Containment Vac. Relief Valve	3114	FCV 25-7 failed open	Removed water from sensing lines-test ok
ECCS Room D/P	3168	Transmitter failed high	Replaced oscillator amplifier assembly
Shield Bldg. Vent D/P Alarm	3228	Ann. has alarm-indicator reading normal	Installed retro-fit-reset setpoints & cal'd.
Containment Personnel/Escape Locks	3286	Repair monitors	Replaced bad I/C and cal'd. monitor
Personnel Hatch	3854	Outer seal switch actuation arm loose	Fabricated new pin & installed-tested sat.
ECCS Emerg. Fan HVE-9A	4255	Spurious Alarms on low flow or motor overload	Adjusted arm contact screw on flow switch
Shield Bldg. Vacuum Alarm - "A"	6240	Low vacuum alarm condition	Replaced retro-kit - operation satisfactory
Fuel Pool Exhaust Fan HVE-16A	6415	Low flow/overload alarm	Adjusted flow switch
Containment Purge Sample Valves	6421	Sample valves indicate midway position	Readjusted limit switches - tested satisfactory

INSTRUMENT & CONTROL CORRECTIVE MAINTENANCE ON SAFETY-RELATED EQUIPMENT

EQUIPMENT	PWO #	MALFUNCTION	CORRECTIVE ACTION
Control Rm. A/C HVA-3C	6460	Received trouble failure alarm	Adjusted FS-25-10C (flow switch)-test sat.
Shield Bldg. Vent. Vacuum Alarms	6514	Setpoint too low on PDIS-25-7A & B	Increased setpoints adjustment per set point list (was 4" Wg vacuum; now is 4.5" Wg Vac.)
Containment Isolation Monitor #4	3086	Non-functional	Replaced defective V/I unit
			test signal
Containment Isolation Ch. 3	4050	High MA readings	Problem in MA detector-cal'd. & returned to service
Ch. "A" ARMS Recorder	6232	Recorder & ECCS meter indication differ	Adjusted V/I to correspond with proper input & output voltage
ARMS Monitor Cab. "A"	6299	Take up reel broken on chart recorder	Soldered broken take-up reel
Cont. Rad Monitor RIS-26-3-2A	6345	MR/HR Reading varies from ESFAS panel	Replaced V/I -adjusted & cal'd new V/I to loop
Containment Isolation FCV-26-04	6352	Valve failed leak test	Adjusted limit switch-stroked valve
Containment Isolation Ch. 5	6399	High alarm light not working	Repaired lead & resoldered
H2 Analyzer I-FSE-27-6	3265	Valve Stuck in closed position	Adjusted stroke
H2 Analyzer Sample Panel	6401	No closed indication for valve 27-09	Adjusted micro switch & replaced fuse
H2 Analyzer Sample FSE-27-10	6451	Does not indicate open when cycled	Repaired bulb holder-adjusted micro sw.
Refueling Machine	6170	Setpoint load adjustment	Adjusted setpoints per CE Letter F-SF-0981
Spent Fuel Handling Machine	6348	Hoist stuck in upper elevation position	Readjusted overload & fuel load limit switches
Refueling Machine	6375	Setpoint change	Reset underload setpoint
Upend carriage	6388	Overload occurs during transfer	Corrected static zero on load cell-checked setpoints
Post Accident Panel Recorders	6239	Chart drive mechanisms not functioning properly	Adjusted all chart retainer assy-replaced motor in #159

DESIGN CHANGES

On the following pages are descriptions, including a summary of the safety analyses, of the design changes implemented at St. Lucie Unit #1 during the period March 1, 1976 (issuance of operating license) through December 31, 1976.

Plant Change/Modification 1-76

Unit #1

"CEA GUIDE TUBE IRRADIATION TEST PROGRAM"

This Plant Change/Modification provides for installation of 3 Zircaloy test specimens in the St. Lucie Unit #1 core. The specimens are patterned after the neutron source assemblies employed in several currently operating Combustion Engineering reactors. The test specimens are installed in fuel assembly guide tubes in an arrangement identical to that used for the neutron source assemblies. The program will confirm the growth of cold worked Zircaloy material under actual "in-core" conditions. This was done at the request of Combustion Engineering.

This change is not an unreviewed safety question because:

(1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

Design, construction and installation were done under the same criteria and procedures as were other in-core assemblies such as neutron sources, surveillance capsules and in-core instrument thimbles. The Nuclear Regulatory Commission has reviewed the Combustion Engineering program on a generic basis and concluded that "the health and safety of the public and plant personnel will not be affected by the program." (Letter Olan B. Parr to A. E. Scherer dated November 21, 1975).

(2) The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

(3) The margin of safety as defined in the basis for technical specifications had not been decreased.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 7-76

PSL Unit #1

"SAFETY SHOWER AND EYEWASH STATION AT CVCS CHEMICAL ADDITION STATION"

This change installs a safety shower and eyewash station at the Chemical and Volume Control System chemical addition station to comply with OSHA reguations.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The water supply being tapped is not safety related and the area (Reactor Auxiliary Building) involved has been evaluated regarding possible flooding.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The non-safety related electrical junction box in the area was waterproofed as part of this PC/M.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 10-76

PSL Unit #1

"FUEL HANDLING BUILDING RADIATION MONITORING FLOW CONTROL VALVE"

This bleed-off valve motor was wired so that it was always energized. Thus, when the sampling pump was turned off, the valve was run against its travel stop trying to raise system flow by shutting off bleedoff flow. The motor was rewired so it is de-energized with the sampling pump and won't damage itself. Also an extra ground on a shield cable was removed by this PC/M. The one ground recommended by the vendor is still installed.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accidental or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Any leakage past this valve is filtered in the sampling unit and is returned to the filtered ventilation system anyway. This ventilation sampling system is not safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No functions or design intents were changed and no new components were added.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 11-76

PSL Unit #1

"WIRING CORRECTION FOR PDIS-25-16A and B"

This Plant Change/Modification changes the terminals for alarm relay wiring to give a closed contact for the low vacuum (high pressure) alarm. As originally wired the alarm was set up for low pressure instead of low vacuum. Changing the terminals for two wires causes the alarm to function properly. This was discovered during preoperational testing shortly before receiving our operating license and correction reviewed/processed via the Plant Change/ Modification program.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter any functions of the instruments; it just corrects the as-built system to comply with the original design intent (alarm on low differential pressure between Emergency Core Cooling Systems pump rooms and remainder of Reactor Auxiliary Building).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 12-76

Unit #1

"COVERS FOR EX-CORE NEUTRON DETECTORS"

Aluminum covers were fabricated and installed on the top of the wide range excore neutron detectors to prevent dust and debris from entering the detectors through the top opening. The debris had been reducing the insulation resistance (outer shield to ground) below the required value (1×10^6 ohms). The detectors were cleaned before installation of the cover plates.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The cover plates are passive elements which prevent the entrance of dust or debris and thereby reduce the probability of detector failure due to electrical grounds and insulation deterioration. These failure modes are already considered in the Final Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The materials and methods of construction used are the same as the original detector/housing/lift mechanism assembly which has not been changed by this addition.

3. The margin for safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 13-76

PSL Unit #1

"ALLOCATION OF A PERMANENT CABLE TO CONTROL ROOM FOR THE RADIATION CALIBRATION FACILITIES"

This change designates a previously abandoned (spare) cable, which goes to the Control Room Radiation Monitoring Cabinet, as a permenent "Testing Facility Cable" and reroutes the last 30 feet of the cable from a cable tray in the Electrical Penetration Room to the testing facility via conduit.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The cable will not be permanently terminated but used only for test/ calibration purposes.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This item is not discussed in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 14-76

PSL Unit #1

"SOLENOID VALVE REPLACEMENT ON REACTOR DRAIN TANK VALVES"

This Plant Change/Modification changed the air line solenoid valves for Reactor Drain Tank Isolation valves V-6301 and V-6302. These 2 valves are Containment Isolation valves and must close within 5 seconds. The orifices in the original solenoid valves were too small to meet this specification so solenoid valves with larger orifices were installed and tested satisfactorily. Need for this change was determined during preoperational testing before receiving our operating license and the change processed via the Plant Change/Modification procedure.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new solenoid valves are direct functional replacements for the original valves with the only difference being larger orifice size needed to meet closure time specification on V-6301 and V-6302.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new solenoid valves are direct functional replacements for the original valves with the only difference being larger orifice size needed to meet closure time specification on V-6301 and V-6302.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change was necessary in order to meet technical specifications regarding closure time for 2 containment isolation valves.

This change is not a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 15-76

PSL Unit #1

"REMOVAL OF RELIEF VALVE RV-2185"

This change removed RV-2185 and blind-flanged its inlet and discharge. This relief sometimes lifted when a boric acid makeup pump started and passed boric acid solution into the primary water system (non-borated system). While evaluating the valve and its setpoint it was determined that the valve was not needed due to another relief, not discharging to primary water, located in the same section of line (within the same isolation valves). This was determined shortly before our operating license was issued and reviewed/processed per the PC/M procedure.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The section of line involved still has overpressure protection.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The section of the boric acid makeup system involved is non-safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 16-76

PSL Unit #1

"ADD FUNCTIONAL TEST CIRCUIT TO AREA RADIATION MONITORING SYSTEM"

This change added a pushbutton, fixed resistor and a variable resistor to the ARMS Containment Isolation Signal circuitry. This allows internal generation of a test signal on demand and varying it to test the alarm functions and setpoints. Previously, testing of the circuit would have required opening the circuit and supplying a test signal from some external source.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The components involved are supplied by the original vendor to the original design specifications. Proper testing of the circuits will improve overall reliability.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This modification does not cut the detector out of the circuit. As was the original circuit, it is designed to "fail safe". If the test circuit "fails off" it has no effect; if it "fails on" it gives a high (conservative) signal.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   Technical Specifications require testing of this circuitry to verify its operability.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 19-76

PSL Unit #1

"CONTAINMENT INSTRUMENT AIR DRYER OUTLET VALVES CONTROLS"

As originally wired the valve remained open at all times (never closed). This change corrected the problem so the valves close when the associated compressor/dryer is not running. The valves still fail open.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change corrects a wiring error to meet original design intent and, this system is not safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comments under (1) above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 20-76

PSL Unit #1

"RPS POWER RATIO AND CORE PROTECTION CALCULATORS"

This change reverses the polarity of inputs to the summing amplifiers, increases the gain coefficient on the multiplier circuit, and changes the labels on 3 terminals in the Reactor Protective System Power Ratio and Core Protection calculators. This change was determined to be necessary at a similar plant of another utility, reported to us before granting of our operating license and reviewed/processed by the Plant Change/Modification program.

This change is not an unreviewed safety question bacause:

1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change ensures the system will function as intended by the original design.

2) The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change does not add any new design features or functions; it just corrects the design implementation so the system will function per the original design intent.

3) The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 21-76

PSL Unit #1

"REACTOR REGULATING SYSTEM CEA STATUS LIGHTS"

This change removes 2 interposing relays (RTGB-104) in the "high" and "low" CEA insertion rate status light circuitry due to electronic noise interference with other instruments. These relays were intended for functions which previously had been deleted from the design and now serve no function.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The affected circuits do not have any safety functions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   A superfluous active element has been removed from each circuit and no new functions have been added.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

"SAFETY INJECTION LINES (2) PRESSURE TEST"

This PCM was for connecting jumpers and documenting pressure testing of two (2) vent valve lines on the SI headers to the Reactor Coolant System. Jumpers were installed (and later removed) by a PCM as the lines were isolated from the RCS only by check valves and the RCS had to be hot and pressurized to perform the pressure test. This was done after core loading but before initial criticality.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   System was restored to normal after pressure test. Had the (one inch) lines or jumpers failed, the SI header check valves would have prevented any leakage from the RCS itself. Safety injection tank levels were monitored during the test to ensure they remained within limits.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   Change was temporary; jumpers were removed and restored to normal after the pressure test.

This change does not represent a change to the facility as described in the Final Safety Analysis Report

Plant Change/Modification No. 23-76

PSL Unit #1

"REROUTE DRAINS TO REACTOR CAVITY SUMP"

This change reroutes certain drains from the Reactor Drain Tank to the Reactor Cavity Sump. The drains involved (Reactor Coolant Pump seal cooler CCW relief valves and containment instrument air compressors moisture separators) contain chromated Component Cooling Water which could eventually be returned to the Reactor Coolant System when Reactor Drain Tank water was processed. The drains now go to the Reactor Cavity sump which is waste water and is not reused.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The drain lines are non-safety related and have no effect on the non-safety related components served by the lines (RCP and instrument air compressor Component Cooling Water).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 24-76

PSL Unit #1

"TEMPERATURE INSTRUMENTATION NOISE REDUCTION"

This change adds capacitors to the Reactor Regulating System (RRS) temperature circuits to remove electronic noise interference. The noise created undesirable oscillation in the output indications (up to $\pm 5^{o}F$).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The RRS is not safety related. And, this change does not alter any design functions or setpoints of the involved system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The RRS is not mentioned in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modifcation 25-76

PSL Unit #1

"BYPASS CONTAINMENT EVACUATION ALARM DURING CIS A.R.M. CHANNEL TESTING"

This change installs a bypass switch to allow functional checking of Containment Isolation System Area Radiation Monitoring channels without actuating the Containment Evacuation alarm. Functional testing is done by injecting a high radiation signal to test the entire circuit and, previously, this actuated the Containment Evacuation alarm. Frequent actuation of this alarm during testing distracts personnel from their duties and may create a complacent attitude toward the alarm.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The containment evacuation alarm is not safety related. In addition, use of the key operated bypass is administratively controlled.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The alarm itself is not safety related and the modification was designed and tested to ensure no other circuits were affected.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The alarm is not discussed in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Rbport.

NOTE: This PC/M is closely related to and was processed/ performed at the same time as PC/M 26-76.

Plant Change/Modification 26-76

PSL Unit #1

"CONTAINMENT EVACUATION ALARM BYPASS DURING A.R.M. PANEL ALARM TEST"

The Area Radiation Monitoring Panel alarm annunciators are tested by opening the circuits to ground which causes the annunciators to alarm. This change connects previously installed relays which bypass the test contacts for the containment evacuation alarm to prevent false evacuation alarms. Frequent actuation of this alarm during testing distracts personnel from their duties and may create a complacent attitude toward the alarm.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The Containment Evacuation alarm is not safety related. Also, this change does not alter any functions; it just corrects the as-built system to include the relays already installed to meet the original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The alarm itself is not safety related and the modification was designed and tested to ensure no other circuits were affected.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The alarm is not discussed in the technical specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

NOTE: This PC/M is closely related to and was processed/performed at the same time as PC/M 25-76.

Plant Change/Modification 27-76 PSL Unit #1

REPLACEMENT OF V/I CONVERTERS ON AREA RADIATION MONITORING CONTAINMENT ISOLATION SIGNAL CHANNELS

This change installed larger capacity V/I converters, supplied by the original equipment vendor on the A. R. M. - C. I. S. channels. The system impedance was too large for the removed converters and loaded them down excessively.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Replacement part meets or exceeds the standards of the original part.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Channel failure is already covered in the Final Safety Analysis Report.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 28-76

PSL Unit #1

"TURBINE CONDENSER ΔT RECORDER WITH ALARM"

This change installs instruments (RTD's) with printout and alarm in the control room to monitor and record condenser ΔT.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change adds temperature indications and alarms to a non-safety related system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change aids operators in meeting requirements of the environmental Technical Specification on condenser ΔT.

   NOTE: This equipment has been installed but is not yet fully operational as functional testing is not satisfactorily completed.

Plant Change/Modification 31-76

PSL Unit #1

"ESFAS CABINET MODIFICATION"

This change modifies the Engineered Safety Features Actuation Signal Cabinets so that upon loss of power to a cabinet all channels except Recirculation Actuation and Containment Spray Actuation trip (formerly all channels bypassed).

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter any functions of the system; it corrects the as-built system to agree with original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report. The change is necessary to conform to the Facility Safety Analysis Report.

NOTE: This change was reported as the corrective action for Licensee Event Report 335-76-3.

Plant Change/Modification 33-76

Unit #1

"INSTALLATION OF GAGE LINE SNUBBERS"

This change installed snubbers to protect gages from pressure surges on pump starting and stopping. The safety-related pumps involved were the flash tank pumps and reactor drain pumps. Other pumps involved were in the waste treatment systems.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The pulsation dampers (snubbers) used met the same or better standards as the original tubing and gages and are located adjacent to the pressure gages outside the system isolation valves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   These systems are not evaluated in the FSAR accident analysis and, in addition, addition of passive components in a location isolable from the system is very unlikely to add a malfunction mode.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 34-76

PSL Unit #1

"INSTRUMENT AIR PRESSURE SWITCHES-MAIN STEAM ISOLATION VALVES"

This change replaces the present switches with new ones. The new type has a stainless steel bellows. The old type failed several times due to vibration induced bellows failure.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new bellows will improve reliability of instrument air to the MSIV's as it is less susceptible to failure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new switch meets or exceeds the original specifications and is a direct functional replacement for the original switch.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comments under (1) and (2) above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 36-76 PSL UNIT #1

"COOLING FANS FOR STATIC UNINTERRUPTIBLE POWER SUPPLY CABINETS"

This change adds cooling fans and high temperature alarms to the SUPS cabinets to supplement the natural circulation cooling originally provided for. The natural circulation cooling was adequate but caused shorter than desired preventative maintenance schedules to ensure SUPS reliability.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not change any design intent or function of SUPS and will enhance operational reliability.

2. The possibility for an accident of malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Reliability of SUPS will be enhanced due to better service life of components. If a fan should fail, the cabinets will still be cooled by natural circulation as originally intended.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 40-76

PSL Unit #1

"MOVABLE INCORE FISSION CHAMBER AND AMPLIFIER"

This change installed a fission chamber instead of a self-powered rhodium detector for the movable incore neutron flux monitoring system. The fission chamber is smaller and more sensitive and can better measure detailed flux patterns and fuel densification gaps if any should exist. The amplifier was necessary to power the chamber and electronically process the chamber output.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No design intents or functions were changed. The new equipment better implements the original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 41-76

PSL Unit #1

"CURRENT TRANSFORMER GROUNDS"

This change removed superfluous ground connections (in RTGB-101) on the CT (amperage monitoring) circuits for 4160V and 6900V switchgear. The vendor supplied grounds on the CT circuits at the switchgear. It is company policy for personnel safety reasons to have one and only one ground on these circuits so the redundant grounds were removed.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Only redundant ground wires were removed. No functions or circuits were changed. The remaining grounds are adequate for personnel/equipment protection.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Only redundant ground wires were removed. No functions or circuits were changed. The remaining grounds are adequate for personnel/equipment protection.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   These grounds are not discussed in the technical specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 43-76

PSL Unit #1

"MODIFICATION OF FUEL POOL PURIFICATION LOOP SIPHON BREAKER"

This change plug welds the ½" siphon breaker in the spent fuel pool purification loop. This will allow interim use of the spent fuel pool as a source of RCS makeup in the event the RWT is unavailable due to a tornado or other causes. A permanent makeup source will be provided and the siphon breaker reinstalled before spent fuel is placed in the fuel pool.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The siphon breaker will be reinstalled before spent fuel is placed in the fuel pool. This is administratively controlled per the plant backfit list.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The siphon breaker will be reinstalled before spent fuel is placed in the fuel pool. This is administratively controlled per the plant backfit list.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The siphon breaker will be reinstalled before spent fuel is placed in the fuel pool. This is administratively controlled per the plant backfit list.

Plant Change/Modification 47-76

PSL Unit #1

"CHANGE MECHANICAL SNUBBERS FROM LOCKING TO NON-LOCKING TYPE"

This Plant Change/Modification changes the original INC locking type mechanical snubbers to Pacific Scientific non-locking type. The locking snubbers will not release until the force which caused the acceleration is removed. If a pipe experienced "jerky" movement due to thermal growth, the locking snubbers could lock and thereafter act as a restraint (until the next cooldown). Although we had a thorough monitoring program to ensure no problems on initial heatup, the snubbers were changed to non-locking type to eliminate this concern for future operations.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The replacement snubbers meet or exceed the requirements of the original specification.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The replacement snubbers meet or exceed the requirements of the original specification.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   These snubbers are not discussed in the technical specifications.

This change does not/does represent a change to the facility as decribed in the Final Safety Analysis Report.

Plant Change/Modification 50-76

PSL Unit #1

"C.E.A. CHANGE MECHANISM TRANSVERSE DRIVE MOTOR"

This change removed the ¼ horsepower transverse drive motor and replaced it with a ½ horsepower motor (and larger motor overload protection). The original motor was too small and continually tripped out during preoperational testing. This was determined before receiving our operating license and reviewed/processed by the PC/M program.

This change is not an unreviewed safety question because;

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This system is not safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This system is not safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This system is not discussed in the technical specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 51-76

PSL Unit #1

"INCREASE SENSITIVITY OF FLOW ALARM SWITCHES-
CONTROL ROOM VENTILATION FANS"

This change installed new flow switch paddles and balance springs on the Low Flow Alarm switches for the Control Room Ventilation fans. This increased the sensitivity so the switches respond to the air flow and clear the (previous) continuous low flow alarm.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The switches sense low flow; they cannot cause it. The new parts were ordered from the original vendor per the original specifications.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No functional changes were made and the switches now meet original design intent.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The low flow alarm is not required by the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 52-76 PSL Unit #1

"MODIFICATIONS TO CONTROL ROOM AIR CONDITIONING COMPRESSORS"

To get the desired capacity the vendor originally converted standard (Seismically qualified) 6 cylinder compressors into 5 cylinder units (one cylinder was blanked). When repairs were needed on one unit we discovered that this approach was no longer used by the vendor and repair parts were unavailable. The units were therefore restored by the vendor to 6 cylinder units using his conversion kits, which give slightly shortened stroke and therefore the same capacity as the original 5 cylinder units.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Change was done by vendor using many original parts and replacement parts equal to or better than original parts.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Failure of all 3 units is already considered in the Final Safety Analysis Report. The present type of unit has a longer and better satisfactory operational record than the 5 cylinder units.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   There are still 3 units of same capacity as the original compressors (two are required by Technical Specifications).

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

"Thot INPUT TO REACTOR PROTECTION SYSTEM"

This PC/M reverses 2 pairs of leads for each channel of Thot input to the RPS. The former arrangement provided a positive input signal and the present arrangement gives the required negative input signal.

This change is not an unreviewed safety question because:

(1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

This change does not alter any functions of the RPS; it just corrects the as-built system to comply with the original design intent.

(2) The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

(3) The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 54-76

PSL Unit #1

"RESTRICT ORIFICE IN CONTAINMENT PURGE SUCTION LINE"

This change further restricts the installed orifice in the Containment Purge Suction ductwork. System flow was originally set at design flow under dirty filter conditions. With clean filters, system flow is higher and creates a greater than desired vacuum in containment. The design dirty filter condition is considerably higher than the actual ΔP's reached before normal filter replacement.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This system is not safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Further restriction of the original installed orifice enables this non safety related system to better meet original design intent. The filter will be changed well before the design (maximum ΔP) dirty filter condition is reached so there is no need for the excess flow.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This system is not discussed in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 55-76 PSL Unit #1

"CHANGE SET PRESSURE ON RELIEF VALVE (RV) 5124"

This relief valve is located on the pressurizer steam space sample line downstream of the flow throttle valve and sample cooler. The normal flow rate in this 3/8 inch line created sufficient back pressure to exceed the conservative valve set pressure. This change increased the set pressure 15 psi to prevent lifting the relief valve.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   A break of this line is already evaluated and design pressure of the line is higher than the new valve set pressure (90 psig).

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 57-76 PSL Unit #1

"PERSONNEL AIR LOCK SEAL LEAK TEST ALARM MODIFICATION"

This change modified the test unit and alarm circuitry so if the air lock door were opened during a leak test, the test would be terminated, pressure vented off and the alarm would not occur. Immediately venting the pressure increases seal life and eliminating spurious leak rate test failure alarms avoids unnecessary distraction of the operators.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No airlock related accidents are discussed in the Final Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments in (1) above. Also, the circuitry was tested for proper operation after the modification.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change meets present Technical Specification requirements.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 58-76

PSL Unit #1

"CHARGING AND LETDOWN FLOW TO DATA PROCESSOR"

This change corrects the wiring for the charging and letdown flow signals to the data processor (for calorimetric calculations). The data processor required the Δp signal directly from the flow transmitters but previously was receiving its signal from the indication portion of the circuit. This potential problem was identified in 1975 but could not be resolved until full information on the data processor was received from the vendor in early 1976.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The monitoring instruments involved are not required to safely shutdown the plant and are not discussed in the Final Safety Analysis Report in relation to accident conditions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The monitoring instruments involved are not required to safely shutdown the plant and are not discussed in the Facility Safety Analysis Report in relation to accident conditions.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 59-76

Unit #1

"WASTE CONCENTRATOR PRESSURE SWITCH REPLACEMENT"

This change installs new pressure switches with adjustable differential ranges on the waste concentrator. The new switches will automatically start and stop the concentrator feed pumps on low and high levels respectively. The old switches would not do that properly due to their lack of adjustable differential range.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The equipment involved is non-safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No functional changes are made; the system is modified to meet original design intent.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 60-76

PSL Unit #1

"C.E.A. DROPPED ROD CONTACTS FOR DATA PROCESSOR"

This change removes jumpers in the CEDS cabinets for dropped rod signals to the data processor for 4 CEA's. Previously the contacts for the 4 CEA's were in parallel with the data processor and it did not sense these 4 CEA's if they drop. Now the circuit is a series circuit and the data processor functions properly. This was discovered during pre-critical preoperational testing.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter any circuit functions; it just corrects the "as-built" system to confirm to original design intent.

2. The possiblity for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility described in the Final Safety Analysis Report.

Plant Change/Modification 63-76

Unit #1

"HEAT TRACING CIRCUITS IMPEDANCE CHANGE"

Four heat tracing circuits originally had less than 2 ohms impedance. The heat tracing circuits' controllers will not operate properly below 2 ohms. This change modified the circuits to have greater than 2 ohms impedance while still providing virtually identical heating capacity.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No function changes are made; the 4 circuits are modified to perform per the original design intent. Materials used are excess from the original supply used for heat tracing circuit. This change will improve reliability for the affected 4 circuits.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See 1 Above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 64-76 PSL Unit #1

"ADD ORIFICE UNIONS & ORIFICES TO CONTAINMENT INSTRUMENT AIR COMPRESSORS"

This change adds orifices (and replaces existing unions with special unions) to the compressor unloading lines. These rotary, water seal ring compressors spray water on each unloading (discharge back to suction) cycle and the orifice will prevent this. As part of this PC/M, the nipple supporting the suction air filter/muffler was increased in length from 2 inches to 24 inches to prevent accumulated moisture from dripping out.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This is not a safety related system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 65-76

PSL Unit #1

"CHANGE I.V.M. POWER SUPPLY AND ADD ANNUNCIATION ON RTGB.102"

This change supplies power to the reactor Internals Vibration Monitor panels from an uninterruptible supply and adds a constant voltage transformer to eliminate voltage fluctuations. Also, it adds a visible and audible alarm to alert the operators that the I.V.M. has detected an undesirable condition.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No I.V.M. related accident is discussed in the Final Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comment under (1) above. Also, this change will increase the reliability of the I.V.M. by eliminating unnecessary low voltage shutdowns.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 66-76 PSL Unit #1

"POST-LOCA PANEL RESISTANCE TEMPERATURE DETECTORS"

Due to availability, 2 uncompensated RTD's were installed as an interim solution. They would not calibrate to the desired degree of accuracy. New, compensated type RTD's were installed and properly calibrated by this PC/M, which is based on the Field Report which tracked/documented the original problem.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change removes the interim solution and restores the equipment to meet the original design accuracy.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 69-76

Unit #1

"PERSONNEL AIR LOCK INDICATOR LIGHTS"

This change added warning lights by the interior and exterior personnel air lock doors to indicate that an airlock door seal leak test is in progress or results were out of tolerance and airlock doors should not be opened until the test is completed (light goes out) or problem is resolved. Prior to this modification, personnel approaching the airlock doors had no indication of seal tester status or of a possible alarm condition.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Design function of system has not changed- relay and indicating lights were added for indication only.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comment under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 70-76

PSL Unit #1

"COOLING WATER CANAL LEVEL AND TEMPERATURE INDICATION"

This change adds discharge and intake canal level indication and adds a pre-alarm to installed discharge canal temperature instrumentation. The discharge level will give operators indication to help avoid overflowing the canal banks. The pre-alarm will give operators warning that condenser ΔT is approaching the limit given in the environmental Technical Specifications.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident of malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This circulating water canal level and temperature instrumentation is non safety related and is for indication/alarm only.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new instruments give indicaiton and alarms only.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The temperature pre-alarm will aid in meeting environmental technical specifications but has no effects on the margin of safety.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 72-76

PSL Unit #1

"WASTE GAS HEADER DRAIN CONNECTIONS"

This change installs drain lines and valves at various low points in the waste gas header and adds slope to part of the header. This eliminates pockets in the header and allows draining of accumulated moisture from the header.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new drain connections and valves are designed and fabricated to the same as or better standards as the original header, and this system is not safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

"ADD FLANGES IN CHARGING PUMP SEAL WATER VENT LINES"

This change adds flanges in the vent lines so the seal water pump can be removed for maintenance without cutting and rewelding the vent line.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Failure of this line/change would not prevent operation of the seal lube system and in addition, the charging pumps can be operated without the seal lube system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   There is no change in function of the seal lube water system or the vent lines.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 75-76 PSL Unit #1

"ADDITIONAL WATER SEAL ON CCP SEAL LUBE WATER PUMP SHAFTS"

This change installed an additional water seal on the shafts of the Coolant Charging Pumps' Seal Lube Water Pumps. These pumps are a one-piece pump/motor unit and the one existing seal allowed moisture to enter the motor area causing motor/motor bearing failure.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The seal lube water system is not safety related and, this change improves the reliability of the system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change does not affect any functions of the seal lube system and improves system reliability.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 77-76

Unit #1

"FAST DEAD BUS TRANSFER FROM AUXILIARY TO START-UP TRANSFORMERS"

This change adds time delay relays to the two (A and B) transfer circuits which will prevent transfer of plant auxiliary loads from the Auxiliary to the Startup Transformers if greater than .17 seconds (10 cycles) has elapsed since loss of power to the Auxiliary Transformers. This will prevent out of synchronization transfer of in-plant switchgear to the system grid.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The involved circuits are non-safety related, non-Class IE equipment. This change involves only power for non-vital auxiliary loads, and, this will improve overall reliability by preventing out of synchronization transfer which could result in equipment damage.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Total loss of all off-site power is already evaluated.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 78-76

PSL Unit #1

"1C AUXILIARY FEEDWATER PUMP CONTROLS"

This change modifies 2 contacts at one control station for 1C AFW pump (steam driven). Previously, when the 1B steam supply header was selected at the remote operating station, the pump shut down. The wiring was changed to agree with that at the main (Control Room) control station so the pump would shut down only when desired. This was discovered during hot functional testing before initial criticality.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter any functions of the system; it just corrects the "as built" system to agree with the original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 80-76

Unit #1

"ACCUMULATORS ON WASTE GAS COMPRESSOR DISCHARGE LINES"

This change installs accumulators on the discharge lines of each waste gas compressor between compressor and discharge check valve. Formerly the check valves chattered at the end of each discharge stroke. The accumulators will prevent this, thus saving time, money and radiation exposure by reducing valve maintenance.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The accumulators are designed, fabricated and installed to the same as or better specifications than the original equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification 81-76

PSL Unit #1

"MAIN STEAM ISOLATION VALVES AIR SUPPLY MODIFICATION"

This change installed additional air accumulators connected to the existing ones for air supply to the MSIV's. The existing accumulators were adequate to hold the valves open for 8 hours (minimum) without supply air as described in the Final Safety Analysis Report. The new accumulators were added to ensure ability to close the MSIV's within technical specifications limits without supply air.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   There are no changes of function and materials/installation criteria used were equal to existing design.

   The consequences of accidents evaluated in the Final Safety Analysis Report remain the same or are decreased. The closure time of an MSIV is improved with this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   There are no changes of function and materials/installation criteria used were equal to exisint design.

   The consequences of accidents evaluated in the Final Safety Analysis Report remain the same or are decreased. The closure time of an MSIV is improved with this change.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   There are no changes of function and materials/installation criteria used were equal to existing design.

   The consequences of accidents evaluated in the Final Safety Analysis Report remain the same or are decreased. The closure time of an MSIV is improved with this change.

Plant Change/Modification No. 82-76 PSL Unit #1

"STILLING TUBE MODIFICATION"

This stilling tube is a sensing line for two level switches which give a low level alarm for the intake cooling pumps suction source (intake well). This change shortens the tube and relocates a bracket to prevent interference with the intake well trash rake.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The tube still extends below the low level alarm setpoint and the instrument gives only an alarm - there are no control features associated with these instruments.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change does not change any function of the involved instruments.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The tube still extends below the required low level alarm point.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 84-76

PSL Unit #1

"COMPONENT COOLING WATER CHEMICAL ADDITION TANK DRAIN LINE"

This change reroutes the CCW Chemical Addition tank drain line to the chemical drain system. This will aid in waste treatment, avoid the health hazard exposed chromated water poses and prevent contamination or exposure problems should CCW become contaminated.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased. Neither the tank nor the drain line is safety related. The drain comes from the CCW chemical addition tank through an originally installed valve and <u>then</u> the drain line is rerouted from a sump to the chemical drain system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The CCW chemical addition system is non safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The chemical addition system is not discussed in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 85-76

PSL Unit #1

"MODIFICATION OF TUBING SUPPORTS ON HOT LEG 1A SAMPLE LINE"

This change modified the tubing supports for this sample line. The original supports were sliding collars which had an inside diameter slightly too small to allow the tubing to slide when it expanded due to plant heatup. The new supports allow free thermal expansion but still provide seismic restraint.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter the functions, numbers or locations of the tubing supports; it corrects installation to allow full thermal growth.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 88-76

PSL Unit #1

"MAIN STEAM ISOLATION VALVES BYPASS VALVES CONTROL CIRCUITS"

This change bypasses the seal-in feature and the limit switch lockout feature. Previously, if the valves were open about 5% or less (to warm up steam lines), they could not be (electrically) closed without first going to greater than 5% open.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change enhances the reliability of bypass valve closure under all conditions including Main Steam Isolation Signal.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 89-76

PSL Unit #1

"COLLAR ON PIPE PENETRATING THE CONTAINMENT SUMP SCREEN"

One pipe experienced sufficient thermal growth to cause greater than the maximum 1/2 inch gap where it penetrated the screen entering and exiting the sump area. This change added a stiff wire mesh collar outside the present screen to cover this gap.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The decrease in screen flow area is negligible (less than 1/2%) and the collars are located outside the present screen.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The collars are added to ensure we meet the requirements of the Final Safety Analysis Report (no gap in containment sump screen greater than 1/2 inch).

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 97-76

PSL Unit #1

"SMOKE AND HEAT (FIRE) DETECTION SYSTEM
AUDIBLE ALARM"

The original alarm for the fire detection system was not loud enough to be heard easily over normal control room noise levels. This change added a louder alarm on the system console to replace the original fire detection console alarm.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This system is not nuclear safety related and the change does not alter any functions; it replaces the original alarm with a louder one. This ensures the system meets the original design intent of notifying the operator of any smoke/heat detector alarms.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 101-76

PSL Unit #1

"POWER OPERATED RELIEF VALVE RELAY COILS"

The original relay coils were rated at 115-125 Volts DC which is unit battery voltage. However, normally the battery chargers are in operation to ensure the batteries remain fully charged and system voltage is about 135V D.C. The new relay coils are rated for continuous operation in the higher voltage.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new relays were purchased to the same as or better specifications than the original relays and will enhance system reliability by preventing coil burnup.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change does not alter any design intents or functions.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 102-76 PSL Unit #1

"INSTALLATION OF HYDRAULIC SNUBBER TESTER"

Technical Specifications require periodic performance testing of hydraulic snubbers. This change allows installation of the testing machine in the Reactor Auxiliary Building.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The machine is not itself safety related and it is not located in the vicinity of any safety related equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   Neither the machine nor the area in which it is located are discussed in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 103-76

PSL Unit #1

"REMOVAL OF CYCLE TIME WIRE IN CEDM CPP'S"

This change removed a wire from the CEDM Coil Power Programmers which had been acting as an antenna, picking up electronic noise and causing/contributing to inadvertent rod drops. The wire involved ran from a connector terminal to another terminal and was not connected to any other part of the circuits. It originally had been for monitoring the cycle time signal for test purposes but this function previously was deleted from plant design.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The wire does not have any functional purpose and the design intent of the circuit and system are not changed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change reduces the probability for dropped rods and cannot create any new accidents.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 104-76

Unit #1

"REPLACE RELAYS IN CONTROL ROOM OUTSIDE AIR INTAKE RADIATION MONITORS"

The original relays were 120 VAC relays modified for use in a DC circuit. The vendor has informed us this is not suitable for long term use. New, DC relays have been installed per this PC/M.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change alters no functions; it corrects the as built non-safety related system to meet original design intent/function.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comment under 1. above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 109-76

PSL Unit #1

"MODIFY FILTER OUTLET PIPES"

This change shortens the outlet standpipes of the CVCS, fuel pool and waste management filters by 3/4 inch. This is done to accommodate the "throwaway" filter/cage assemblies previously approved (before licensing) as a means of significantly reducing personnel radiation exposure.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter any functions of the affected systems.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This is not a functional change and does not affect any filter performance monitoring instrumentation.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 112-76

PSL Unit #1

"CONTROL ROOM AIR CONDITIONING THERMAL EXPANSION VALVE CAPACITY REDUCTION"

This change installs new internals of lower capacity in the (freon) thermal expansion valves in the control room air conditioning system. The original larger valves cycled excessively and due to low freon velocity appeared to allow compressor oil "hideout" in the system. The smaller valves will cycle less, promoting greater stability and create higher velocity freon flow to avoid oil "hideout".

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new internals are a modification kit designed and fabricated by the vendor of the original valves. And, failure of one (of 3) air conditioning units is evaluated. The greater stability of operation will improve system reliability.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   Complete failure of one air conditioning unit is already considered in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 113-76

Unit #1

"ADD VIBRATION RESTRAINT TO CHARGING PUMP SUCTION LINE"

This change added a vibration restraint with snubber to the charging pump suction line. This avoids the possibility of long term operation with slight vibration causing damage to the pipe.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The fluid boundary of the system is not changed. The restraint is designed, built and installed to Seismic Class I standards as good as or better than original specifications.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change reduces the possibility of a piping failure.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The fluid boundary of the system and the piping configurations are not changed.

This change does not represent a change to the facility as described in Final Safety Analysis Report.

Plant Change/Modification 114-76

PSL Unit #1

"EXCORE NUCLEAR INSTRUMENTATION LINEAR AMPLIFIER GAIN CHANGE"

This change increased the gain of the subchannel B (upper detectors) linear amplifiers for Channels B, C and #10 (Control Channel #2) of the Power Range Linear Channels. These amplifiers were designed and built with the flexibility to change the gain if needed to accomplish subchannel calibration and it was done by moving two wires to different terminals on the amplifier cards. The need for this change was discovered during power ascension (at 20% power) while testing and calibration were in progress.

This change is not an unreviewed safety question because:

1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change takes advantage of the designed flexibility of the system and should be considered a normal calibration adjustment.

2) The possibility for an accident of malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change does not change the function of any part of the system but simply changes the amplifier gain much as does the installed fine adjustment/calibration potentiometer.

3) The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 116-76

PSL Unit #1

"REACTOR PROTECTION SYSTEM AT POWER NOISE REDUCTION"

This change moved the cable shield ground from the instrument ground to the loop transmitter common (-). This gave a significant reduction in 60 cycle electronic noise in the circuits.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change did not alter any functions; it just rerouted the ground connection.

2. The probability of occurrence or the consequences of an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Instrument failure is already addressed.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 117-76

Unit #1

"INSTALL SNUBBERS ON ΔP INSTRUMENTS FOR INTAKE COOLING WATER STRAINERS"

This change installed snubbers on the sensing lines to the ΔP instruments. Previously pressure surges in the lines caused spurious flow alarms even though flow was normal.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The snubbers (pulsation dampers) were purchased to the same as or better specifications than the original components.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No design intents or functions were altered. Addition of these passive components will improve overall system reliability by eliminating spurious alarms and reducing effects of pressure pulsations on the instruments.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 118-76

PSL Unit #1

"MODIFICATION TO SEISMIC RESTRAINTS FOR 1"SAFETY INJECTION LINES"

In post core load hot functional testing, it was discovered that restraints for 2 lines (1" SI-120 and 1" SI-237) did not allow for the full thermal growth experienced by the lines. This change modified 3 restraints to allow full thermal growth while still providing Seismic Class I support.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change reduces the probability of an (already analyzed) piping failure and does not increase the consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Components used were designed and built to the same as or better specifications than the original restraints.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comments under (1) and (2) above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 119-76

PSL UNIT #1

"MODIFICATION TO SEISMIC RESTRAINT FOR BLOWDOWN VALVE FCV-23-4"

In post core load hot functional testing it was discovered that the restraint for steam generator blowdown valve FCV-23-4 did not allow for the full thermal growth experienced by the valve/line. This change modified the restraint to allow full thermal growth while still providing Seismic Class I support.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change reduces the probability of a piping/valve failure already analyzed in the Final Safety Analysis Report and does not increase the consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Components used were designed and built to the same as or better standards than the original restraint.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comments under (1) and (2) above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 120-76

PSL Unit #1

"DIGITAL DATA PROCESSOR-MOVEABLE INCORE DETECTOR SYSTEMS INTERFACE"

This change makes wiring modifications and adds a depth encoder driver (amplifier) so the incore detector system can properly feed signals to the DDPS. The DDPS controls the incore system and provides the information from that system to the operators. The two systems are from different vendors and were not entirely compatible.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This does not alter any functions of either system; it simply corrects an interface problem so these two non safety related systems will function per original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 121-76 PSL Unit #1

"EMERGENCY CORE COOLING SYSTEM AREA LOW VACUUM ALARM"

The low vacuum alarm was wired so that it was armed at all times. Since the fans which maintain the required vacuum are required to run only under accident conditions (auto-start by Safety Injection Actuation Signal) this resulted in many spurious alarms. The alarm was rewired to be armed only when there is a Safety Injection Actuation Signal which is the only time this alarm is meaningful.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Reduction in spurious alarms will improve operator response to an actual alarm thus improving overall system operation. This alarm circuit does not affect fan operation or actual vacuum in the ECCS area.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new relay used meets at least the same qualifications as the original components.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This alarm is not discussed in the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 122-76

PSL Unit #1

"LIMITORQUE CONTROL CIRCUIT MODIFICATION"

This change removed limit switch contacts in the control circuits which were intended to prevent valve chatter on closure. However the contacts also prevented the valves from closing if they were open 5% or less (without first opening them to >5%). The vendor stated that the operators involved would not chatter. The valves involved were the High Pressure Safety Injection pump discharge valves, High Pressure Safety Injection and Low Pressure Safety Injection header isolation valves and auxiliary feed pump discharge valves.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change corrects the system to meet original design intent and improves the reliability of the involved valves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No functions were changed and valve reliability was improved.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 123-76

PSL Unit #1

"COOLANT CHARGING PUMP PACKING AND SEAL MODIFICATION"

This change reduces the number of primary packing rings, installs a spring loaded bushing to retain packing instead of only a spring, replaces a metal packing adapter with a closer tolerance non-metallic adaptor and changes the secondary packing design (cross-section). These changes (already done at another plant with similar CCP's) will solve the problem of extremely short packing life presently being experienced.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This will improve operation/reliability of the CCP's. The new design is equal to or better than the original design and has been approved by both the pump vendor and the Nuclear Steam Supply System vendor.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No pump functions are affected by this change.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 124-76

.. PSL Unit #1

"REVERSE POLARITY - POWER DEPENDENT INSERTION LIMIT"

This change reversed two leads in the Reactor Protective System Channel "D" so the PDIL would increase as power was increased. As previously reported in Licensee Event Report 335-76-20, the PDIL was decreasing as power was increased.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not affect any functions of the circuit; it just corrects the "as-built" input polarity to meet original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comment under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comment under 1 above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 126-76

Unit #1

"CHANGE CABLE FOR WIDE RANGE NI CHANNEL C"

This change corrects an erroneous cable pulling card and installs the specified type of cable from the electrical penetration room to the Control Room.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Installation of the specified cable changes the channel to meet original design intent and reduces the probability of (already analyzed) cable/channel failure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comments under (1) above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 129-76

PSL UNIT #1

"CHARGING PUMP HIGH LEVEL CUTOUT BYPASS"

This change installs a key-operated bypass switch to allow running more than one charging pump with high pressurizer level. This will allow better control of pressurizer level during cooldown and more expeditious filling of pressurizer when taking the plant solid and filling the drained pressurizer after maintenance.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Strict administrative controls will prevent use other than when specified above and when pressurizer level goes above 100% (top of indicating range).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Per original design we could run at least one charging pump at all times.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 130-76 PSL Unit #1

"EXCORE NUCLEAR INSTRUMENTATION AUDIO CIRCUIT NOISE"

The audio count rate circuit was the source of electronic noise in the Reactor Protective System circuits when switching ranges. This change installed toggle switches to eliminate the noise and prevent more spurious plant trips.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change provides additional separation between Nuclear Instrumentation and Réactor Protective System channels.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The functions of the affected systems remain exactly the same and no new functions have been added.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 134-76 .

PSL Unit #1

"INTAKE COOLING WATER PUMP INSTRUMENT CABLE REROUTING"

This change reroutes one cable to another conduit so the original conduit is available to support electrical services to the Steam Generator Blowdown Treatment Building.

This change is not an unreviewed safety question because:

1. The probability of occrrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The rerouted cable is a pump pressure indication instrument cable and is not evaluated in the safety analysis. It is rerouted in a proper fashion through safety class conduit and does not reduce redundancy.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 135-76

Unit #1

"RESLOPE STEAM GENERATOR LEVEL REFERENCE LEG PIPING"

This change reslopes the reference leg piping for the level transmitters so they will slope downward toward the steam generators when the plant is at hot operating conditions. This removes a low point which could retard circulation and prevent proper operation of the condensate pots and thereby affect indicated steam generator level.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change increases overall reliability of the Steam Generator level indication system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No design functions or intents are changed; this change improves the implementation of the original design intent.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 137-76

PSL Unit #1

"EXCORE NUCLEAR INSTRUMENTATION LINEAR POWER RANGE DRAWER MODIFICATION"

This change replaces a resistor with one of different rating and replaces a potentiometer with a more sensitive vernier drive potentiometer to obtain greater resolution and control of voltage input to the amplifiers of the drawers. It also adds a switch as part of the potentiometer which completely removes the trip test pot from the circuit when the test pot is "off". This removes a residual signal which formerly was applied to the circuit.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No functions have been added or deleted and the design intent is not altered. This change improves the implementation of the original design intent by separating the test signal from the actual signal and improving the resolution of the amplifier gain adjust.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 146-76 PSL UNIT #1

"EXTRA REACTOR HEAD CABLE TRAYS"

This change adds four (4) temporary (refueling use only) cable trays to be mounted above the existing trays during refueling outages. The reactor head cables (instrument and CEDM power and position indication cables) can be folded up into the new trays after being disconnected from the head. This storage area will prevent tangling of cables (formerly folded back into the original trays) and damage to the cable connectors.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The cables are not in use during refueling when the new trays are in use. The new trays will be removed when the head cables are reconnected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The added trays simply hold the cables as did the original trays during refueling when the cables are not in use. Same trays and supports are used as in the original design.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 147-76 PSL UNIT #1

"MODIFICATION OF FUEL POOL PURIFICATION LOOP SIPHON BREAKER"

This change temporarily removed the plug installed in the siphon breaker per PC/M 43-76 to allow storage of irradiated fuel in the fuel pool for replacement of poison pins in the assemblies (see PCM 176-76). After the irradiated fuel was removed from the fuel pool, the plug was reinstalled.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Removing the plug returns the fuel pool to original design conditions for storage of irradiated fuel. When the fuel is removed, the siphon breaker or the fuel pool are not required. The plug was reinstalled after fuel was removed (See 3. below).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The removal of the siphon breaker is administratively controlled per the plant backfit list. Also see PC/M 43-76.

3. The margin of safety as defined in the basis for technical specifications has not been decreased. The siphon breaker is required by the Technical Specifications when spent fuel is stored in the fuel pool. The plug is required during plant operation to provide an interim source of tornado protected makeup water per the PSL Unit 1 operating license. The siphon breaker would prevent pumping the water out if it were not plugged.

Plant Change/Modification 157-76

PSL Unit #1

"FUEL TRANSFER TUBE SHIELDING"

This change added concrete radiation shielding to the north side of the fuel transfer tube between the Reactor Containment and Fuel Handling Buildings. The need for this shielding was identified before licensing and added to the backfit list. It was advanced in schedule to allow defueling of the reactor due to the power distribution anomaly. (See PC/M's 176 and 192.)

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equpment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   . This Seismic Class I shielding (passive component) will reduce radiation exposure during spent fuel transfers and does not alter the design or configuration of the related structures.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comment under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 158-76 PSL UNIT #1

"PIPE HANGER MODIFICATION"

This change modified five (5) hangers on two (2) non-seismic but safety related (Category 2 & 3) lines. This was done to provide clearance for installation of steam generator blowdown lines to the Steam Generator Blowdown Treatment Facility. The affected lines were Reactor Coolant Pump Seal Bleedoff to the Volume Control Tank and a Fuel Pool Purification line.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The modified pipe support hangers are made of the same materials, to the same specifications as were the originals. The routing of the lines is not changed and no functional changes are involved. The original hanger attachment to the building is used. The hangers were not and are not seismic.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1, above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 159-76

PSL Unit #1

"ADD CHECK VALVE IN BORIC ACID HOLDING SYSTEM"

The change added a check valve in a line from the Boric Acid holding tank downstream of a relief valve which discharges to the holding tank. During certain evolutions, the line was pressurized and leakage past the relief diluted the holdup tank. The check valve will prevent the line and the relief from being pressurized.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The line is non-safety related and complete failure of the line would result only in partial loss of feed flow to the non-safety related boric acid concentrators.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See Comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   The line/related system is not discussed in the technical specifications.

PLANT CHANGE/MODIFICATION NO. 160-76 PSL UNIT #1

"HIGH PRESSURE AND LOW PRESSURE SAFETY INJECTION HEADER ISOLATION VALVES POSITION INDICATION"

This change moves a resistor from one leg to another in the position indication circuitry. This was done as the original circuit would not calibrate properly to meet the desired accuracy for the % open indicator. This indicator is one of three indications in the control room for valve position and valve position can be determined at the valve itself.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No functions are changed. The % open meters are for operator information only. The design intent and function remain the same as the original design.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No new components or functions are added. This change simply ensures the indicators will calibrate to the desired accuracy.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 161-76

PSL UNIT #1

"ADD FUEL OIL SOUNDING TAPES AT EMERGENCY DIESEL FUEL OIL TANKS"

This change adds sounding tapes to the diesel fuel oil tanks. This is primarily a company policy regarding inventory control of fuel oil but also provides a back-up level indication to help ensure proper reserves of diesel fuel oil are maintained per the technical specifications.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The tapes are mounted on the manhole cover plates on the top of the tanks and do not affect tank integrity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This provides a back-up indication to ensure technical specifications on diesel fuel oil reserves are met.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 163-76

Unit #1

"ADDITION OF BARRIER IN TWO CABLE PULL BOXES"

This change added a barrier in 2 cable pull boxes to maintain minimum separation requirements between safety related and non-safety related cable.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change alters no design functions or intents; it corrects 2 cable pull boxes to meet the original design intent.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1. above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 164-76

PSL Unit #1

"INSTALLATION OF DRIP PANS OVER TSP BASKETS"

This change adds drip pans over the trisodium phosphate dissolving baskets located in containment for post-Loca coolant pH control. The pans will prevent condensing moisture from process lines from reaching and partially dissolving the TSP in the baskets. This will aid in ensuring proper amounts of TSP are maintained.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The implementation of this PC/M deals with an item which in itself is not required for safe shutdown of the plant, thus it will not increase the probability of occurrence of any accident. The drip pans and all supports are Seismic Class I and will not come loose in the event of an earthquake or a LOCA and thus cannot cause blockage of the containment pump screens.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change improves the margin of safety by ensuring the proper inventory of TSP can be maintained.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 166-76

PSL Unit #1

"INSTRUMENT AIR TIE-IN FOR STEAM GENERATOR BLOWDOWN TREATMENT FACILITY"

This change adds the line and valve to allow supplying instrument air from the turbine building to the blowdown treatment facility.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Instrument air is non-safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   Instrument air is non-safety related.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 168-76

Unit #1

"PRESSURIZER SPRAY VALVE SEAT MODIFICATION"

This change provides new, interference fit seats and seat retainers for the spray valves. This will be aligned by the interference fit and seal welded in place. This is recommended by the valve vendor as the original screwed-in retainer can loosen and allow leakage past the seat.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No functional or design changes are made and this modification will improve valve performance by eliminating the possibility of leakage bypassing the seat.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new seat and retainer ring are provided by the original vendor to the same as or better specifications as the original parts.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification 167-76

Unit #1

"MODIFY REACTOR HEAD CABLE JUNCTION BOXES"
(REFUELING DISCONNECT BOXES)

This change enlarged the opening into the head cable junction boxes. The original openings were undesirably small. They were nearly filled, thus restricting access to hook up the connectors and their size/location required some cables to make two sharp curves to mate the connectors. This resulted in excessive time for connecting/disconnecting the cables and subjected the cables to undesirable risk of damage.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change does not alter any function or design intent of these cables.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   This change makes the systems supplied by these cables more reliable in that it improves access and makes it possible to better insure proper connection without damage or undue stress to the cables or their connectors.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 172-76

PSL Unit #1

"ADD NEW GAIN RANGE TO LINEAR POWER RANGE NUCLEAR INSTRUMENTATION DRAWERS"

As reported for PC/M 114-76, 3 drawers required gain increases (option was designed into the amplifiers) for proper output. This change installs a new gain range between the two options supplied by the manufacturer. This allows all drawers to be wired the same and provides proper gain span for all the drawers.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   There are no new functions or design intents. This change adds and uses another gain span between (and partially overlapping) the two options (ranges) provided by the manufacturer. This will still provide proper gain for all drawers (as now exists) and allows all drawers to be wired the same which eliminates a possible source of confusion to the personnel calibrating and maintaining the drawers. The new resistors and capacitors used were purchased to the same as or better specifications as the original ones.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 173-76

PSL Unit #1

"CORRECTION OF HYDRAULIC SNUBBERS OVERFILL HOLE LOCATION"

This change returned 7 snubbers to the vendor for relocation of the overfill (weep) holes. These 7 (of many at PSL #1) had the holes mislocated so that hydraulic fluid inventory was reduced which would require frequent inspection and/or refill thus affecting plant availability.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change affects no function or design intent; it just corrects the snubbers to meet their original design intent. The mislocation deprived the snubbers of fluid margin which could affect plant availability due to the Technical Specification requirements.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change improves plant ability to meet the Technical Specifications without affecting plant availability.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 176-76

PSL Unit #1

"REPLACE PSL CYCLE 1 FUEL POISON PINS"

This change replaced (reconstituted) the boron carbide poison pins in 108 fuel assemblies for the St. Lucie Unit 1, cycle 1 core. The original pins had excessive internal moisture content, resulting in hydride corrosion and perforation of the cladding. The flow plate was cut to allow replacement with new poison pins essentially identical to the originals. A retention grid was placed over the holes and mechanically fastened (crimped) in place. The work was done by vendor technicians, in the PSL spent fuel pool, under vendor and Florida Power & Light supervision, with final QC inspection by both vendor and Florida Power & Light personnel.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The temporary equipment used was designed/selected and load tested to ensure that the probability of a fuel handling accident due to equipment failure was not increased. Although the number of fuel transfers over the life of the plant has not been quantified, it should be noted that the fuel transfers required for repair are the same as could be required by fuel inspections which are implicit in the original design of the spent fuel storage facilities.

   Also, analysis was performed to verify that transfers/rework could not result in inadvertent criticality even if juxtaposition of up to 4 assemblies in the borated spent fuel pool should occur. For this evolution surveillance was established to verify boron concentration was maintained. Analysis of the results of severe fuel assembly damage was performed and proven to be within the bounds of the FSAR analysis of a fuel handling accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The FSAR analysis of damage to fuel assembly fuel pin cladding does not address the mechanism of damage but only addresses the results of such damage. Analysis of the results of severe fuel assembly damage was performed and proven to be within the bounds of the FSAR analysis of a fuel handling accident.

Plant Change/Modification No. 176-76 (cont.)

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   A review of the basis for technical specification indicated that no reduction in margin of safety would result due to the minimal fuel exposure (irradiation) time; the maximum power level of 80% obtained prior to shutdown; and the fuel decay time of 76 days prior to commencement of reconstitution.

   NOTE: This PC/M covered the actual work of replacement. See PC/M 192-76 for analysis of the return to critical operation using the modified fuel. Also, see CEN-38 Rev. 0 and Rev. 1 previously submitted to the NRC.

Plant Change/Modification No. 177-76

PSL Unit #1

"ANNUNCIATE 15% PER HOUR POWER CHANGE"

This change installed connections and modified the Digital Data Processor System to annunciate on the RTGB any power change of 15% per hour or more.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Change adds an indication (alarm) function only and does not affect any DDPS functions or accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change provides indication to help alert the operators to take technical specification required action on a power change of 15% or greater per hour, thus increasing the margin of safety.

This change does not represent a change in the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 181-76

PSL Unit #1

"REVISE CONTROL SCHEME FOR VOLUME CONTROL TANK LEVEL CONTROL VALVES"

This change provides a new controller with adjustable deadband for controlling valves V-2501 and V-2504. The original controller functioned but caused the valves to cycle open and shut continuously, resulting in eventual damage to the valve operator motors.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   There is no change in design intent or function. System reliability is improved as the valves will not stroke continuously and be damaged. This portion of the Chemical and Volume Control System is non-safety related.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The new controller was purchased to the same as or better specifications than the original.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 182-76

PSL Unit #1

"SHUTDOWN COOLING RELIEF VALVE MODIFICATIONS"

Due to problems experienced (lifting below setpoint and excess blowdown) these two valves were modified as follows:

1. Repositioned so the valve stem/disc assemblies are in the vertical plane

2. Revamp the inlet piping to provide a direct flowpath of larger pipe size

3. Install expansion loops in the outlet piping to reduce stresses if one header is operating (hot) and the other is shutdown (cold).

4. Reset the blowdown from 25% to 10%.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No design functions or intents are changed and the valves/system will be more reliable (Refer to LER 335-76-40 dated August 18, 1976). The revised piping was designed and fabricated to the same specifications as the original.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 183-76

PSL Unit #1

"LETDOWN BACKPRESSURE CONTROL MODIFICATION"

This change adds a lead/lag unit to the Chemical and Volume Control System backpressure control valves and an adder/subtractor unit to the letdown control valves. The adder/subtractor unit will slightly delay opening of the letdown valves to allow more time for the backpressure valves to respond. The lead/lag unit will cause the backpressure control valves to anticipate changes in flow from the letdown valves based on the pressurizer level error signal. This will prevent the pressure surges which have lifted the relief valve downstream of the backpressure control valves.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This does not change any design intent or functions but merely coordinates the controls of the two valves to provide smoother, more integrated functioning of the system and improve reliability.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 185-76

PSL Unit #1

"EXTEND LEAK TEST CONNECTIONS FOR FUEL TRANSFER TUBE PAST THE SHIELDING"

This change adds tubing to extend the local leak rate test connections to the outside of the radiation shielding. This allows testing the fuel transfer tube nozzle seals without removing the radiation shielding for access.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No design functions or intents are changed and the two extensions are designed and fabricated to the same specifications as the original connections. The connections remain within containment and the shield building annulus which are protected, filtered release areas.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   This change significantly reduces the work and radiation exposure necessary for the leak rate testing required by the Technical Specifications.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 190-76

PSL UNIT #1

"LIQUID WASTE DEMINERALIZER SYSTEM"

This change installed additional liquid waste ion exchangers to supplement the installed waste ion exchanger. A small demineralizer was added in parallel to the original one to allow resin replacement or maintenance without stopping waste processing. Two "polishing" demineralizers (in parallel) were added to the system to increase the decontamination factor. This allows use of the system at higher inlet activities than was previously possible.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new components are equivalent to the original equipment. This system is non-safety related. Liquid waste release to the environment is a manually controlled and monitored function independent of the means used for in-plant processing and this change will not affect the FSAR or Technical Specification Environmental Sections.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1 above.

3. The margin of safety as defined in the basis for Technical Specifications has not been decreased.

   See comments under 1 above.

Plant Change/Modification 192-76 Unit #1

"RETURN TO POWER USING RECONSTITUTED FUEL"

This change presents the safety analysis for using, at power, the fuel modified by PC/M 176-76. Briefly, that modification consisted of: cutting webs/drilling holes in the fuel assembly flow plate; removing 12 poison pins and replacing with new ones; and installing a retention grid over the holes.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The only changes were to the poison pins and the upper flow plate. The fuel pins were not affected and the fuel assembly alignment was checked (gaged) to verify it was not affected.

   The poison pins were changed only as follows:

   A. Internal moisture content was reduced-the cause of the perforations in the cladding was hydride corrosion due to excessive moisture content. Reduction in moisture will improve reliability by removing the prerequisite for this corrosion mechanism, thus minimizing probabilities of recurrence of the boron loss/redistribution and resulting flux anomalies.

   B. The nominal OD of the pins was increased .0045 inches - this has been evaluated as having no significant effect on flow characteristics or temperatures within the pins.

   C. To facilitate handling the upper end cap of the new poison pins was modified - this has no effect on in-core performance.

   D. The lower end caps of the new poison pins were modified to ensure proper retention in the retention grid - this was done to ensure proper hold down against up-lift forces such as would be experienced in a postulated LOCA.

   E. The upper end plenum spring has been changed to the vendor's current design for similar (14 x 14) fuel assemblies - this has been evaluated as having no adverse effects on in-core performance.

1. (Cont'd)

The same materials were used as in the original pins and poison loading was the same as the original "as built" poison pins. Due to the low depletion of the fuel this loading will have no significant effects on core performance.

The upper flow plates were changed only as follows:

A. Holes (.56 in dia.) were drilled in each corner. These were partially blocked by the hold down pins of the new retention grid and the combination will have no significant adverse effects on flow or strength characteristics of the flow plates.

B. One web was removed from each side of the flow plate and the opening partially filled by the new retention grid.

The hold down pins (A above) and retention grid provide poison pin holdown, against postulated LOCA and normal conditions, equivalent to the original design. The material removed from the flow plates does not significantly affect their rigidity or strength. The modified flow plate/ new retention grid does not significantly affect total fuel assembly/core flow characteristics.

To summarize, the modifications to the flow plate and the new poison pins have no significant effects on any performance characteristics of the fuel assemblies and the modification should prevent recurrence of the flux anomaly previously found.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

See Comments under (1) above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change restores the fuel assemblies to essentially "as built" conditions regarding reactor physics parameters while having no significant effects on any mechanical properties. Restoration of the physics performance will eliminate the flux anomaly which had the potential of exceeding Technical Specification limits on power distribution.

NOTE: See PC/M 176-76 and attached summary also. For further details, see CEN-38 (F)-P Rev. 1, submitted with letter L-76-368 dated 25 October 1976.

Power Distribution Anomaly and Fuel Reconstitution

On June 30, 1976, with the reactor at 80% power and all control rods out, a routine power distribution map gave the first indication of a small azimuthal power tilt. This was attributed at that time to detector errors or failure. It should be noted that at this time Technical Specifications for tilt and total radial peaking factor (F_r^T) were suspended for physics testing in accordance with the special test exceptions of the Technical Specifications.

Within the next week a few incore alarms were received. During evaluation of these, it was found that the calculated alarm setpoints were in error (LER 335-76-34, August 6, 1976) and it was also determined that the previously indicated tilt was still present. The alarms were corrected. On July 6, 1976, plant power was reduced to 50% for routine cleaning of a condensate pump strainer. While at 50% power, it was determined conclusively from the power distribution that an azimuthal tilt of approximately 4% was present along with an axial peaking value of 1.5, as compared to an expected value of <1.35. This tilt was verified using the moveable incore detector system. Technical Specifications for tilt and total radial peaking factor (F_r^T) were reinstated. It should be noted here that at no time was the plant in violation of any Technical Specification regarding azimuthal tilt or peaking. (LER 335-76-35, July 23, 1976)

On July 13, reactor power was reduced to about 10^{-2}% and a low power physics test program commenced. This program was a repeat of selected protions of the LPPT performed after initial startup. At this time two theories were offered as possible explanations: 1) a selective deposition of crud on the fuel leading to local flow maldistributions; and 2) early burnout of the burnable poison pins in the fuel assemblies. The results of these tests (available July 18) verified that the tilt was present, and that the core was more reactive (about .45%) than predicted. This second finding tended to support the early poison pin burnup theory. It was decided to open the reactor vessel for inspections and a shutdown/cooldown was commenced. Over the next week, many discussions were held, data was reduced and evaluated and theories postulated. None of this information could conclusively explain the existing phenomenon; therefore, on July 27, actual disassembly of the reactor began.

Representative fuel assemblies were removed from various areas of the vessel and inspected. The crud buildup theory was quickly dispensed with, as blisters and perforations were found on the poison pin cladding. More fuel assemblies were removed and inspected. Sufficient flaws were found to statistically demonstrate that there was a core wide problem with the cladding of the burnable poison pins. It should be noted here that no evidence was noted of any fuel pin anomalies. Due to the core-wide poison pin problem, the plant was defueled.

After the discovery of these poison pin cladding failures, a new theory was postulated. This was that the failure allowed the boron within the rods to wash out and be lost or to migrate and redistribute within the poison pins. This boron loss/redistribution theory correlated much better than any other theories previously considered.

It was then necessary to resolve two major concerns: 1) what caused the cladding failure and 2) what must be done to return the plant to power operation. To aid in resolving the first concern, poison pins were removed from selected fuel assemblies. These were submitted to on-site visual and eddy current testing. Then they were sent to research laboratories to determine the cause of the cladding failures, the mechanisms of boron loss and redistribution, and verification that loss of boron had occured in some pins and that it could cause the observed results.

As a result of these laboratory/test reactor inspections, the cause of the failure was confirmed to be hydriding of the zircalloy cladding of the pins. This was caused by excessive moisture content within the pins. Under incore conditions of high temperature and neutron flux the moisture produced free hydrogen which attacked the cladding. It was proven that the perforations did result in loss/redistribution of boron from the affected poison pins under incore conditions. And, it was confirmed that this loss/redistribution of boron could create the conditions observed at the St. Lucie Plant.

Then, regarding resolution of the second concern, it was determined that on site replacement of the poison pins with new ones of much lower moisture content was the appropriate solution. At the time this decision was made, some of the pin removal equipment had already been proven in removal of the pins for testing. So, there was reasonable assurance the job could be done even though it had to be done under water in the spent fuel pool. The vendor's specifications and controls on moisture content were significantly tightened to avoid repetition of the original problem. Replacement of the poison pins resulted in fuel assemblies virtually identical to the original ones except for minor fuel depletion (burnup).

Actual reconstitution (removal of old pins and installation of new ones) commenced on October 5, 1976. The basic procedure consisted of: drilling or cutting the flow plate webs above the poison pins; cleaning and deburring the newly machined surfaces; removing old pins using a template to ensure fuel pins were not removed; installing new pins; installing a retention assembly over the flow plate; and final QC and FP&L acceptance inspection. This process is described in greater detail in the FP&L submittals leading up to Ammendment 10 to St. Lucie License DPR-67, dated 3 December 1976, which allows resumption of power operations.

By November 3, 1976 this process was close to complete and core reload was commenced. By November 7, all but 2 assemblies were completed and on November 10, the last of the 108 assemblies had been reconstituted and core reload was continuing (supplementary LER 335-76-35, December 17, 1976).

We have now resumed power ascention testing and thus far have seen no evidence of any anomalies except those directly related to the uneven fuel depletion (burnup) which resulted from the power tilt/peaking. These have been minor in magnitude and should be self-correcting as plant operation (and fuel depletion) continue. The activities after fuel reconsititution (fuel reload, initial criticality etc) will be described in our supplementary Startup Report(s).

Plant Change/Modification No. 194-76

PSL Unit #1

"STEAM GENERATOR BLOWDOWN TIE-IN FROM UNIT #1 TO TREATMENT FACILITY"

This change tied in the blowdown lines to the Unit #1 interface in the penetration room. The blowdown treatment facility is a requirement of our license.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Reyort has not been increased.

   This system is part of the original design per the F.S.A.R. The tie-in and piping added in this PC/M is only to implement the system as described.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   The piping involved is designed as Seismic Class I where applicable and whip restraints are installed to prevent damage to nearby safety related equipment.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

   See comments under 1 and 2 above.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

Plant Change/Modification No. 196-76

PSL Unit #1

"STEAM GENERATOR FEED RING MODIFICATION"

In order to prevent draining the feed ring upon low S/G water level each feed ring had 74 nozzles penetrating the bottom. From these nozzles, standpipes (1" diameter) extended up into the feedring to near the top. An inspection revealed that the nozzles and standpipes were susceptable to vibration and fatigue failure. The standpipes were removed and the nozzles plugged. To allow feed flow and prevent draining of the feed ring 36 four inch 90^{o} elbows were welded to the top of each feed ring.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The elbows will have lower velocity thus resulting in less vibration and potential for erosion so they are less likely to fail than the previous design. Feedwater instability (water hammer) is already evaluated in the FSAR and this change will not increase the probability of occurrence.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No design function or intent is changed. This modification provides a better method of implementing the original design intent.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 197-76

PSL Unit #1

"RELOCATE CRAFT ACCESS GATE AND GUARD STATION"

When the license was issued, contractor craft access was controlled by a guard station at the south end of the turbine building adjacent to the contractor's support facilities. As work started under the Limited Work Authorization for Unit #2, it was found desirable to separate the Unit #1 backfit craft personnel and their access entirely from the Unit #2 area and provide separate (limited) support facilities. The access gate and guard station were relocated to the north side of the site. This location is further away from the safety related systems area than the original gate and is in view of the main guard station which was not true for the old location.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   Failure of the site security system is not analyzed in the FSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   No change in design function or intent of the security system or its equipment was made. The location of a gate and its guard station was changed and the old location sealed in the same manner as the rest of the site security perimeter.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

Plant Change/Modification No. 200-76

Unit #1

"FUEL HANDLING EQUIPMENT MODIFICATIONS DUE TO FUEL RECONSTITUTION" *

This change added an alignment plate to the Spent Fuel Handling Machine grapple to ensure only the center guide tube could be grappled and that the new retention grid * would not be damaged by the grapple. Also, it removed part of the grapple shoe lugs on the Refueling Machine to ensure the new retention grid would not be damaged by that grapple.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The change will not adversely affect handling of unmodified assemblies and will ensure proper handling of the modified assemblies. No design intents or functions are changed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

*See PC/M 176-76

Plant Change/Modification No. 202-76

Unit #1

"GOVERNOR MODIFICATION FOR STEAM DRIVEN AUXILIARY FEED WATER PUMP"

This change added a small oil reservoir directly to the governor system. This provides a source of oil very close to the governor oil pump and gives faster governor response upon a quick start. This slows the initial rate of acceleration and prevents overspeed trips during turbine startup.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   This change was recommended by the turbine vendor and is now included on their new units. The parts are supplied by the original vendor to the original specifications. This change will improve system reliability by preventing overspeed trips and possible equipment damage. Auxiliary Feedwater pump failure is analyzed. This passive component will not change/increase probability of that failure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See Comments under 1. above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 203-76 PSL UNIT #1

"DUAL CEA EXTENSION SHAFT REPLACEMENT"

This change replaced the original dual CEA extension shafts with ones modified to prevent the uncoupling problems experienced during core deload for fuel reconstitution. The major changes were replacing the tubular operating shaft with a solid one to minimize the possibility of stretching and the addition of extensions on the coupling expanders (plungers) to provide an alternate uncoupling technique if needed.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   No design intents or functions were changed and the new shafts were supplied by the original vendor to specifications as good as, or better than, original.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

   See comments under 1. above.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PLANT CHANGE/MODIFICATION NO. 204-76

PSL UNIT #1

"REVISE RESTRAINTS ON LINE 2" SI-141"

A flanged spoolpiece was added to this line to allow installation of a hose for use of the acoustic emission technique during plant hydrostatic testing. This change adjusts the original restraints and adds 2 new ones.

This change is not an unreviewed safety question because:

1. The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report has not been increased.

   The new restraints are designed and fabricated to the same specifications as the original ones.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report has not been created.

3. The margin of safety as defined in the basis for technical specifications has not been decreased.

This change does not represent a change to the facility as described in the Final Safety Analysis Report.

PROCEDURE CHANGES

MARCH 1, 1976 - DECEMBER 31, 1975

The following list summarizes those procedure changes which are changes to procedures as listed in the FSAR in accordance with the provisions of Title 10, Code of Federal Regulations, Section 50.59. A summary of the safety evaluation is provided for each change.

Procedure	Change
Operating Procedure No. 1600021 Unit #1 Initial Core Loading	This change allowed operation of the Spent Fuel Handling Machine for handling new fuel for the initial core loading without the overload interlock in operation and allowing only one fuel assembly to be handled at a time with no water in the spent fuel pool. This was to comply with Amendment #1 to the St. Lucie Operating License allowing such operation for the period March 5 through March 19, 1976. This was not an unreviewed safety question as the interlock is for protection against a (spent) fuel handling accident and the fuel was new and unirradiated. The prohibition on water in the spent fuel pool was an additional precaution against inadvertent criticality.
Emergency and Off-Normal Procedure No. 0120042, Loss of Reactor Coolant	This procedure was revised June 15, 1976 to delete reference to use of the containment hydrogen sampling system due to problems with the environmental qualification of the system valves which are located inside containment. (See LER 335-76-28 dated June 18, 1976). At the time when sampling was to commence, the hydrogen recombiners will be placed into operation to control containment hydrogen concentration. This is an interim solution; the permanent solution (as described in Followup LER 335-76-28) is to replace the valves. This does not involve an unreviewed safety question as the recombiners will be placed in operation at least as soon as originally specified.
Operating Procedure No. 0120051, RCS Flow Determination by Calorimetric Procedure	This procedure was approved December 28 to replace the original procedure for RCS flow determination which used Reactor Cooling System ΔP readings to determine flow. This method of determining flow is independent of any geometric variations in the Reactor Cooling Pumps or ΔP instrument taps and an error analysis has shown it to be more accurate than the ΔP technique. A request for an amendment to our license has been submitted (letters L-76-424 of December 14, 1976 and L-77-4 of January 5, 1977) and contains full details of this technique and its safety analysis.

TESTS

The following list summarizes those tests, other than startup tests, performed under the provisions of Title 10, Code of Federal Regulations, Section 50.59. A summary of the safety analysis is included.

Back Pressure Regulating/Letdown Valve Pressure Testing - This test was conducted to determine the as installed transfer functions of the letdown and back pressure regulating valves and to determine the system response to known pressurizer level disturbances. The test was conducted by instrumenting the system response to ramp and step signal changes to the letdown valves. The purpose of the test was to accumulate data in order to improve the letdown flow operation in conjunction with the back pressure regulating valve to achieve smoother system operation. The test was not an unreviewed safety question because the system was operated within its design limits at all times.

CEA Guide Tube Material Irradiation Test Program - This test installed 3 zircaloy material test specimens in the St. Lucie Unit #1 core. See PC/M-1 for details and the safety evaluation summary.

FAILED FUEL INDICATIONS

On June 30, 1976, after operation at 78% power for nearly 5 days, Iodine levels had increased by a factor of about 25 over previous 50% power levels. The increase was from 2.2×10^{-4} uci/ml to 5.8×10^{-3} uci/ml. Back-up samples and calculation of the I-131/I-133 ratio confirmed that a small fuel failure had occurred. Iodine 131 peaked at 1.35×10^{-1} uci/ml which is .17 uci/ml Dose Equivalent Iodine. Our limit is 1.0 uci/ml Dose Equivalent Iodine. Within 72 hours Iodine 131 levels had started to stabilize at about 2×10^{-3} uci/ml which is equivalent to about 0.002% failed fuel.

On July 6, power was reduced to 50% for cleaning condensate pump strainers. Due to the power distribution anomaly * power was not increased after cleaning the strainers. On July 9, the unit was taken off the line for low power physics testing. After this testing, the fuel reconstitution * shutdown commenced and no further data could be obtained until startup after core reload.

After power operation commenced on December 10, Iodine levels were monitored closely and as of December 31, 1976, at 50% power, Iodine levels were only slightly higher than those at the previous 50% plateau in June. This follows the previous trend in that the failure was not apparent then until 80% power was reached. It is felt that the failure may be power dependent and may return when higher power levels (80% and above) are reached.

* See PC/M's 176-76 and 192-76. The discussion attached to PC/M 192-76 also covers inspection of the fuel, including eddy current inspections of poison pins. It should be noted that no fuel pin failures were detected during these inspections.

CORE BARREL MOVEMENT

Section 4.4.11.3 of the PSL #1 Technical Specifications requires the results of all periodic Amplitude Probability Distribution (APD) and Spectral Analysis (SA) monitoring to be included in this report.

However, Section 4.4.11.1 requires baseline measurements at various power levels up through (nominal) 100% power operations and a special report on the results to the NRC within 31 days after reaching 100% power. PSL Unit 1 has not reached 100% power and has not completed this baseline study. Therefore, we have not yet performed any meaningful periodic APD and SA monitoring and have not completed the baseline monitoring which will provide the data for evaluation of later results.

The report on the baseline monitoring will be submitted as required by Section 4.4.11.1 and results of periodic APD and SA monitoring will be included in the Annual Operating Report for 1977.

STEAM GENERATOR TUBE INSPECTIONS

Section 4.4.5.5.b of the PSL #1 Technical Specifications requires reporting all Steam Generator Tube Inspections in this report. For the period March 1, 1976 through December 31, 1976 no tube inspections were performed.

It is expected that tube inspections, as specified by Section 4.4.5.3.a of the Technical Specifications, will be performed during our first refueling in 1978 and reported in the Annual Operating Report for that year.

	Number of Personnel (>100 mrem)			Total Man-Rem		
Work & Job Function	Station Employees	Utility Employees	Contract Workers & Others	Station Employees	Utility Employees	Contract Workers & Others
Reactor Operations & Surveillance:						
Maintenance Personnel	0	0	0	0	0	0
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	4	0	0	1.20	0	0
Supervisory Personnel	0	0	4	0	0	.46
Engineering Personnel	0	0	0	0	0	0
Routine Maintenance:						
Maintenance Personnel	18	0	2	3.63	0	.23
Operating Personnel	1	0	0	1.31	0	0
Health Physics Personnel	3	1	0	1.60	.61	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Inservice Inspection: Not Applicable for 1976						
Special Maintenance: (Fuel Reconstitution)						
Maintenance Personnel	0	0	25	0	0	4.66
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	2	0	0	.36	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Waste Processing:						
Maintenance Personnel	0	0	0	0	0	0
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	0	0	0	0	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Refueling:						
Maintenance Personnel	44	5	0	10.00	1.08	0
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	6	0	0	.82	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
TOTAL:						
Maintenance Personnel	62	5	27	13.63	1.08	4.89
Operating Personnel	1	0	0	1.31	0	0
Health Physics Personnel	15	1	0	3.98	.61	0
Supervisory Personnel	0	0	4	0	0	.46
Engineering Personnel	0	0	0	0	0	0
GRAND TOTAL	78	6	31	18.92	1.69	5.35

CONTAINMENT PENETRATION LEAK RATE TESTS

The following routine local leak tests were performed during the reporting period to comply with Technical Specification 4.6.1.3.

Penetration Tested	Test Date
1. Personnel Air Lock	11/29/76
2. Emergency Escape Lock	11/4/76

The above tests were conducted in accordance with Operating Procedure No. 1300052, Rev. 3, "Airlock Periodic.Leak Testing".

All detected leaks were within their acceptance criteria. A summary analysis of the tests will be provided in accordance with 10CFR50 Appendix J, following the next integrated leak rate test on St. Lucie Unit #1.

Due to the length of the fuel anomaly outage, the first refueling of St. Lucie Unit #1 will extend beyond the time interval allowed for local leak rate tests. Therefore these tests were performed during this outage from 10-5-76 to 12-2-76 to avoid a shutdown prior to the next refueling outage to comply with Technical Specification 4.6.1.2d.

All tests were performed in accordance with Operating Procedure No. 1300051, Rev. 0.

The total as-found bypass leak rate was 7.0% of the total allowable. The total as found leak rate (bypass + all other) was 18.0% of the total allowable.

Repairs were made to five penetration boundary valves as seemed appropriate with regards to: probability of further degradation causing failure of next test, leak magnitude versus valve size, ease of repair, and scheduling. The total as-left leakage valves were as follows:

By-Pass Leakage	As-Left 4.5%	of total allowable
Total	As-Left 1.4%	of total allowable

The following table lists the valves and penetrations tested with the as-found and as-left leakage valves.

Penetration Number	Valves Tested	As Found *SCCM	As Left *SCCM
1	Main Steam Expansion Bellows	0	0
2	Main Steam Expansion Bellows	0	0
3	Main Feedwater Expansion Bellows	0	0
4	Main Feedwater Expansion Bellows	0	0
25	Fuel Transfer Tube Expansion Bellows	0	0
7	V-15328, I-MV-15-1	1634.2	14.2
8	I-V-18796, I-V-18794	209.0	209.0
9	I-V-18195, I-MV-18-1	42.7	42.7
10	I-FCV-25-4, I-FCV-25-5	19.8	19.8
11	I-FCV-25-3, I-FCV-25-2	98,724.0	820.0
14	V-6779, V-6741	12.8	12.8
23	I-HCV-14-7, I-HCV-14-1	3.4	3.4
24	I-HCV-14-6, I-HCV-14-2	8.9	8.9
26	V-2515, V-2516	1.5	1.5
28	V-5200, V-5203	0	0
29	V-5201, V5204	7.2	7.2
29	V-5202, V-5205	79.6	79.6
31	V-6554, V-6555	0	0
41	I-V-03-1307, V-3463	4.8	4.8
42	I-LCV-07-11B, I-LCV-07-11A	204.2	204.2
43	V-6301, V-6302	185.0	185.0
44	I-SE-01-1, V-2505	54.6	54.6

*SCCM = Standard Cubic Centimeters per minute

Penetration Number	Valves Tested	As Found *SCCM	As Left *SCCM
46	I-V07189, I-V07206	50.0	50.0
47	I-V07188, I-V-07-170	4.0	4.0
48	I-FSE-27-01, 02, 03, 04, 08	0	0
48	I-FSE-27-1341, I-FSE-27-10	0.2	80.2
51	I-FSE-27-05, 06, 07, 09	0	0
51	I-FSE-27-1342, I-FSE-27-11	118.0	42.2
52A	I-FCV-26, 01, 02	1591.2	747.2
52B	I-FCV-26-03, 04	572.1	354.1
52C	I-FCV-26-05, 06	1209.8	1209.8
52D	I-V00140, I-V-00143	6.2	6.2
52E	I-V00139, I-V00144	3.2	3.2
54	Blind Flange each end	0	0
56	I-V-25-11, I-V-25-12	250	250
57	I-V-25-13, I-V-25-14	301	301
58	I-V-25-15, I-V-25-16	1105	1105
67	I-FCV-25-8, I-V-25-20	1583	1583
68	I-FCV-25-7, I-V-25-21	30.0	30.0
Fuel Transfer Tube Flange		1.7	1.7
Maintenance Hatch		420	420
Personnel Airlock		4.2	4.2
Emergency Escape Lock		0	0
Electrical Penetrations		0	0

*SCCM = Standard Cubic Centimeters per minute

ABBREVIATIONS USED

A/C	Air Conditioner
AOV	Air Operated Valve
B.A.	Boric Acid
CCP	Coolant Charging Pump
CCW	Component Cooling Water (for Rx plant components)
Ch	Channel (i.e. one of four channels of the RPS)
CVCS	Coolant and Volume Control System (Charging and letdown)
CWD	Control Wiring Diagram
Disch	Discharge
FCV	Flow Control Valve
FW	Feedwater
FWP	Feedwater Pump
Hdr	Header
HPSI	High Pressure Safety Injection
Hx	Heat exchanger
ICW	Intake Cooling Water (sea water cooling for CCW, Turbine Cooling Water)
ISO or ISOL	Isolation (valve)
Ix	Ion exchanger (demineralizer)
LCV	Level Control Valve
LPSI	Low Pressure Safety Injection
MOV or MV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NI	Nuclear Instrumentation
PCV	Pressure Control.Valve

ABBREVIATIONS (cont)

PRZR or PZR	Pressurizer
RCP	Reactor Cooling Pump
RV	Relief Valve
Rx	Reactor
SDC	Shutdown Cooling (decay heat removal system)
S/G or S.G.	Steam Generator
SIT or SI Tank	Safety Injection Tank (Accumulator)
TM/LP	Thermal Margin-Low Pressure
TX	Transmitter
VCT	Volume Control Tank
V/I	Voltage to Current (signal) converter