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JULY 2 0 1989

L-89-260
10 CFR 50.59

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

Gentlemen:

Re: St. Lucie Unit 1
Docket No. 50-335
Report of 10 CFR 50.59 Plant Changes

Pursuant to 10 CFR 50.59(b)(2), the enclosed report contains a brief description of plant changes/modifications (PCM) which were made under the provisions of 10 CFR 50.59. Included with the brief description of each PCM is a summary of the safety evaluation. This report includes PCMs completed between January 23, 1988 and January 22, 1989 and correlates with the information included in Revision 8 of the Updated Final Safety Analysis Report.

Very truly yours,

C. O. Woody
Acting Senior Vice President - Nuclear

COW/EJW/gp

Enclosure

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

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St Lucie Plant Unit 1

Report of Changes Made

Under the Provisions of

10CFR 50.59

for the period ending January 22, 1989

PLANT CHANGE/MOD REVIEWED FOR PSL1 FSAR AMENDMENT 8

NUMBER	REVISION	TITLE
583-079	0-1	ESFAS POWER SUPPLY
075-082	0	DIESEL GENERATOR HIGH CAPACITY TURBOCHARGER INSTALLATION
297-177	0-3	REACTOR CAVITY FILTRATION SYSTEM
089-182	2	REACTOR VESSEL HEAD SHIELDING
114-182	0-5	TURBINE SUPERVISORY INSTRUMENTATION
355-183	0	THERMAL SHIELD REMOVAL-PHASE III
177-184	0-3	ROSEMOUNT TEMPERATURE TRANSMITTER REPLACEMENT
039-185	2	EDG SUBSYSTEM FLOW DIAGRAMS
090-185	0	REFUELING & FUEL XFR MACHINE
092-185	0	ESFAS POWER SUPPLY
[illegible]-185	2	REPLACEMENT OF VALVE SOLENOIDS
204-185	0	ICW BACKUP LUBEWATER BACKFLOW PREVENTER REPLACEMENT
028-186	0	CCW PUMP JOURNAL BEARING MATL CHG
050-186	0-1	INSTRUMENT AIR UPGRADE
052-186	0-1	UPPER GUIDE STRUCTURE LIFT RIG REPAIR
074-186	1	HEATER DRAIN PUMP DEMINERALIZED WATER SUPPLY
077-186	1	10CFR50.49 EQ LIST REVISION
099-186	0	UGS LIFT RIG LOAD TEST FIXTURE
114-186	0	CONDENSATE PUMPS EXPANSION JOINT REPLACEMENT
119-186	1	10 CFR 50.49 EQ LIST REVISION
128-186	0	S.U. TRANSF LOCKOUT DISC SWITCH
131-186	0	AUTO LEAK RATE TESTER FOR PERSONNEL AIR LOCKS
133-186	0-1	QSPDS SOFTWARE MODIFICATION
[illegible]8-186	0	BECKMAN WASTE GAS SYSTEM OXYGEN ANALYZER REPLACEMENT
140-186	1,2	ANNUNCIATOR NUISANCE ALARMS

PLANT CHANGE/MOD REVIEWED FOR PSL1 FSAR AMENDMENT 8.

NUMBER	REVISION	TITLE
147-186	0	ICW DISCH PIPE ZINC RIBBON
013-187	0	SIMULATOR TRAINING FACILITY GAI-TRONICS
016-187	0	CCW, TCW, OBCW VALVE ACTVATOR REPLACEMENT
018-187	0	DRAIN FOR PIPE LINE R-WM-040
034-187	1	CONDENSER OUTLET TUBE SHEET AND WATERBOX COATINGS
036-187	0	CONDENSER TUBING STRAIN GAUGE INSTALLATION
039-187	0	CONDENSER RECIRC TO COND PNEUMATIC SQRT EXTR REPLACEMENT
041-187	0	MAIN FEEDWATER REG VALVE POS IND REMOVAL
054-187	0	CONDENSATE POLISHER TIE-INS
075-187	0	FIRE DETECTOR MODIFICATIONS
[illegible]-187	0-2	ERDADS/SAS UPGRADE
078-187	0	REPL OF F & P CONTROLLERS
085-187	0	TURBINE GENERATOR SEAL OIL SYSTEM ENHANCEMENT
088-187	0-1	REMOTE REACTOR VESSEL LEVEL INDICATION
105-187	0	CHARGING PUMP BLOCK MATL CHG
116-187	0	REPLCMNT OF S.R. BATT 1A&1B
119-187	0	GROUTING OF MASONRY BLOCK WALLS
123-187	0	CEA MG SETS LOCK-OUT RELAY
128-187	0-1	SI TANK & CONT FAN COOLER INST UPGRADE
141-187	0	480V PCB TRANSFORMER REPLACEMENT
142-187	0	480V LOAD CNTR 1A3 & 1B3 TRANSFORMER REPLACEMENT
143-187	0	480V PCB TRANSFORMER REPLACEMENT
152-187	0	SIT SAMPLE VALVE AS BUILD MODIFICATION
[illegible]-187	0	CEDS COIL PWR PROG PART LENGTH REMOVAL
001-188	0	MOISTURE SEPARATOR REHEATER SHELL REPAIR

PLANT CHANGE/MOD REVIEWED FOR PSL1 FSAR AMENDMENT 0

NUMBER	REVISION	TITLE
003-188D	0	CONDENSER EXPANSION JOINT IMPINGEMENT PLATE MODIFICATION ASS
005-188	0-1	METRASCOPE REPLACEMENT ASST
006-188	0	RCP COOLER HEAT EXCHANGER TUBE LEAK DETECTION
007-188	0-2	RCP VIB MONIT EQUIP UPGRADE
009-188	0	EQ DOC PACK & DISCONN MOV SPACE HTRS
010-188	0	STATION AIR/INST AIR PRESS IND RPLCMNT
011-188	0	RAB/RCB WALKWAY
012-188	0	1B & 1D INSTRUMENT INVERTER DRAWING CHANGES
013-188	0-1	LIGHTING PANEL RELAY
015-188	0	ICW LUBE WATER PIPE RESTRAINT MODIFICATION
[illegible]-188	0	CONDENSATE PUMP DISCHARGE SAMPLING LINES
019-188	0	TURBINE LUBE OIL SYS/RESERVOIR PERMANENT FLUSH CONNECTIONS
020-188	0	LP 122 CKT EXCHANGE
021-188	0	DG BUILDING DELUGE VALVES CLAPPER LATCH ASSEMBLY REPLACEMENT
022-188	0	WIRE DELETION FROM SWITCH SS-2/380
026-188	0	FLOOD PROTECTION STOP LOG #19
029-188	0-1	B.A. CONC RED DOC PAC UPDATE
031-188	0-1	RCB PROTECTIVE COATINGS MAINTENANCE
033-188	0	INSTRUMENT CHANGES FOR HUMAN FACTORS CONCERNS
035-188	0	EXTRACTION STM PIPING MATL UPGRADE
038-188	0	FDWTR REG SYS CONTROLLER REPLACEMENT
043-188	0	EQ LIST REV- SPARE PARTS
046-188	0	EDG DWG & INSTR LIST CORRECTIONS
[illegible]9-188D	0	REACTOR HEAD O-RING RETAINING RING MODIFICATION
055-188D	0	ICW & CW PUMP PACKAGE REPLACEMENT

PLANT CHANGE/MOD REVIEWED FOR PSL1 FSAR AMENDMENT 8

NUMBER	REVISION	TITLE
059-188	0	CONDENSER INLET TUBE SHEET AND WATERBOX COATING
060-188	0	DIESEL GENERATOR GOVERNOR INSTABILITY
064-188	0	REACTOR CAVITY INFLATABLE SEAL
074-188	0	MAIN GEN LINKS MOD
075-188	0	PT INDICATION ENHANCEMENT
076-188D	0	REPLACEMENT OF PRESSURE IND PI-18-3
077-188D	0	REPLACEMENT OF FLOW TRANSMITTER FT-09-3B1
082-188	0	REPL BLDWN CONTROL VLV POSITIONERS
093-188D	0	MAIN GEN SURGE CAP REPLACEMENT
094-188	0	BORIC ACID CONCENTRATION REDUCT
[illegible]-188	0	S/U TRANSFMR 1A&1B DIFF RELAY REPLCMT
107-188	0	FUEL POOL PURIF SYS PUMPS MECH SEAL REPLACEMENT
109-188	0	CONDENSATE RECOVERY SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT
111-188	0	TURBINE GLAND SEAL SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT
113-188D	0	SECONDARY SIDE WET LAYUP SYS PUMPS MECH SEAL REPLACEMENT
114-188	0	DUAL CEA EXT SHAFT REPLACEMENT
115-188D	0	CONDENSER INLET WATER BOX DRAIN
117-188	0	REACTOR COOLANT PUMP CASE TO COVER GASKET REPLACEMENT
119-188	0	SAFETY INJ SYS BLANK FLANGE REMOVAL
122-188	0	ICW PRESSURE INDICATOR UPGRADE
125-188	0	CONTAINMENT FAN COOLER SHORT TERM RESTORATION
127-188	0	CONDENSATE POLISHER SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT
140-188	0	STEAM GEN TUBE PLUG DESIGN
[illegible]2-188	0	600V TAPING PROC
143-188	0	FIRE PUMP BKR O.L. TRIP DEV

PLANT CHANGE/MOD REVIEWED FOR PSL1 FSAR AMENDMENT 8

NUMBER	REVISION	TITLE
150-188	0	RCB EQUIP HATCH DOOR OPER DWG
159-188D	0	30 INCH STEAM GENERATOR NOZZLE DAM SEALS
161-188D	0	MAIN STEAM NOZZLE BLOCK DRAIN PIPE REPLACEMENT
162-188	0	TURBINE DRAIN VLV REPLACEMENT
165-188	0	ITT BARTON TRANSMITTER REPLACEMENT
167-188	0	MFWTR VLVS SPRING RETAINER
170-188	0	IN-CORE INSTRUMENT THIMBLE FLANGE REPLACEMENT
171-188	0	IN-CORE INSTRUMENT THIMBLE FLANGE REPLACEMENT
175-188D	0	MISC. SNUBBER MODIFICATIONS
177-188D	0	MISC. SNUBBER UPGRADE
[illegible]-188D	0	ICW LUBE WATER FLANGE REPLACEMENT
179-188D	0	AFW PIPE & RESTRAINT CORROSION
180-188D	0	FEEDWATER FLOW INST LIST RANGE CORRECTION
181-188D	0	FWRV TECH MANUAL UPDATE TO REFLECT SNUBBER INSTALL
183-188D	0	RCP SEAL CARTRIDGE O-RING PART NUMBER CHANGE
188-188D	0-1	SS/996 REPLACEMENT FOR 1B DIESEL GENERATOR
200-188D	0	THROTTLED VALVE CWD LS DEV
205-188D	0	ADD SETPOINT INFO OR RPS TO SETPOINT INDEX
210-188D	0	I&C VENDOR MANUAL UPDATE (1988)
217-188	0	600V CABLE SPLICES
220-188D	0	PACIFIC VLV 20" CHECK VLV HINGE MATL CHG
221-188D	0	ZURN STRAINER MODEL 595A PART NUMBER CHANGE
242-188D	0	VELAN VALVE PARTS MATERIAL CHANGES HCV3615
[illegible]5-188D	0	CHECK VALVE FEED FROM SHUTDOWN HTEXCH 1A 1B SEAL MAT CHNG
246-188D	0	PORV LOWER SEAL BUSHING GASKET V1402, V1404

PLANT CHANGE/MOD REVIEWED FOR PSL1 FSAR AMENDMENT 8

NUMBER	REVISION	TITLE
285-188D	0	BETA ANNUNCIATOR INST MANUAL
330-188D	0	FT-3321 CWD CHANGE
335-188D	0	SAS POWER FEED CWD CHANGE
004-985	0-2	UPGRADE OF NORTH WASTE WATER TREATMENT FACILITY
007-985	0	HYPOCHLORITE CELL FLUSH SYSTEM
171-985	0	HYPOCHLORITE SYSTEM INSTRUMENT ENHANCEMENT
199-985	1	WATER TRTMNT PLT REGENTN WSTE NEUTRLZTN TANK MOD-BOOSTER PMP
012-986	0	WPT GROUND EROSION REPAIRS
020-986	0	INTAKE CANAL DREDGING AND SLOPE RESTORATION
039-986	0	BLOWDOWN BUILDING RADIATION MONITORING SYSTEM
[illegible]-986	0-1	SIMULATOR TRAINING FACILITY PIPING TIE-INS
032-988	0	SECURITY BLDGS ENHANCEMENTS
095-988	0	SGBTF MONITOR STORAGE TANKS-VENT STACK REPLACEMENT
106-988	0	SG BLOWDOWN TREATMENT FACILITY SYSTEM PUMPS MECH SEAL RPLMT
137-988	0	GENERATOR RETAINING RINGS REPLACEMENT
164-988D	0	REPAIR OF FIRE PROT LINE 6"-FP-153
290-988	0	REMOVAL OF FORMS BLDG FROM SECURITY PERIMETER
N/A	0	REMOVAL OF GUIDE TUBE PLUGGING DEVICES
N/A	0	RELOAD SAFETY ANALYSIS - CYCLE

ESFAS POWER SUPPLY

CHANGE REQUEST SCOPE

This change request is to cancel the replacement and relocation of a new ATI power supply (Consolidated Controls Corporation (CCC)) Device No. PS105) proposed under the original PC/M.

The original ATI power supply (CCC Part No. KDD1907 (Lambda Part No. LXs-C-15) will be retained as in the MA ESFAS cabinet as a result of this change request. The proposed replacement ATI power supply (CCC Part No. LXS-D-15-R)) will not be utilized, as a result of this change request. The location of the new ATI power supply as proposed under the original PC/M will be cancelled. Therefore, the use of the mounting hardware for the new ATI proposed power supply will be cancelled under this change request.

The original ATI power supply will be relocated under approved PC/M 92-185.

PC/M 583-79
CHANGE REQUEST 1

SAFETY EVALUATION

This change request to the Engineered Safeguards Features Actuation System Cabinets (ESFAS) has no effect on nuclear safety and does not alter the intent of the original safety analysis for PC/M 583-79.

The existing ATI power supply (Lambda Part No. LXS-C-15) will be maintained and its location will not change as a result of this change request. Performance of the existing ATI power supply has been reliable for at least ten years of operation.

The power supply is used for test circuit and does not perform a safety related function. The power supply is located in a mild environment and therefore the requirements of 10CFR50.49 do not apply.

The ATI power supply will be relocated under approved PC/M 92-185 to improve ventilation which will result in lower operating temperatures and provide improved performance.

The probability of occurrence of an accident previously evaluated in the FSAR has not been affected since the ESFAS is not utilized in determining the probabilities of accidents.

The consequences of an accident previously evaluated in the FSAR have not been changed since this change request does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

This change request does not affect any other safety related equipment.

The consequences of the malfunction in the FSAR have not been affected. Redundancy, function, or failure mode capacity have not been changed.

The possibility of an accident of a different type than analyzed in the FSAR has not been created since this change request does not affect any systems vital to safety or whose failure could directly result in a non-controlled release of radioactive material.

The possibility of equipment malfunction of a different type than analyzed in the FSAR has not been increased.

The margin of safety as defined in the basis for any Technical Specification has not been changed since this change request does not change the performance, capabilities, or operating characteristics of the ESFAS.
In conclusion this change request to PC/M 583-79 does not involve an unreviewed safety question.

DIESEL GENERATOR HIGH CAPACITY TURBOCHARGER INSTALLATION

SYSTEM DESCRIPTION

1.0 Design Description

The diesel turbochargers supply compressed air to boost the performance of the diesel engines. When the engines operating below 50% load, the turbochargers are gear driven from the crankshaft. The St. Lucie diesel generators have been experiencing numerous turbocharger failures due to the gear train. Electro-motive diesel has produced a new high capacity turbocharger with a new, stronger gear train that extends time between scheduled overhauls up to 1500%. The new high capacity turbocharger is a commercial grade item manufactured by EMD of General Motors to the same high quality standards as the original assembly.

2.0 Function

The diesel turbocharger supplies compressed air to the diesel engine to provide more power output per cubic inch of piston displacement.

3.0 Operation

This modification will not affect the diesel generators' load capabilities or operating characteristics and thus do not change the operating procedures.

SAFETY ANALYSIS

This change does not involve an unreviewed safety question because:

1. a) The probability of occurrence of an accident previously evaluated in the FSAR has not been affected since the diesel generators are not utilized in determining the probabilities of accidents.

   b) The consequences of an accident previously evaluated in the FSAR have not been changed since this modification does not affect the operability of any equipment required to mitigate the effects of an accident.

   c) The probability of malfunction of equipment important to safety previously evaluated in the FSAR has not been adversely affected since EMD states,

   > "High capacity turbochargers have not exhibited any adverse effects from the heavier components even when used in locomotives which experience relatively high "g" forces during hard couplings and high speed crossovers. In our opinion, it is highly improbable that the slight turbo weight increase would sufficiently change resonant frequencies to create seismic sensitivity where it does not now exist. From a functional standpoint, we expect identical response from the 17.9:1 gear ratio high capacity turbo as we received from the 18:1 ratio standard turbo."

   In addition, this modification does not affect any other safety related equipment.

   d) The consequences of the malfunction of equipment important to safety previously evaluated in the FSAR have not been affected for the same reasons give in 1 (c).

2. a) The possibility of an accident of a different type than analyzed in the FSAR has not been created since this modification does not affect any systems vital to safety or whose failure could result in an uncontrolled release of radioactive material.

   b) The possibility of equipment malfunction of a different type than analyzed in the FSAR has not been increased for the same reasons given in 1 (c).

3. The margin of safety as defined in the basis for any Technical Specification has not been changed since this modification does not change the performance, load capabilities, or operating characteristics of the diesel generators.

REACTOR CAVITY FILTRATION SYSTEM

INTRODUCTION

During refueling operations, the refueling cavity is filled with water. The water provides protection from radiation given off by individual fuel bundles in addition to cooling them off. When the reactor head is first removed, radioactive impurities (crud) may be released into the water. As a result, refueling cavity water becomes turbid, making it difficult to observe removal and replacement of fuel assemblies below the water level.

In order to ensure water clarity in the refueling cavity during refueling operations, a reactor cavity filtration system is needed. At present, the purification portion of the fuel pool system performs this function during refueling; however, it may not have sufficient capacity for ensuring the water clarity that is needed.

The fuel pool purification system, located in the FHB maintains clarity and purity of water in the fuel pool, refueling water tank and the refueling cavity. The purification loop consists of the purification pump (150 gpm capacity), ion exchanger, filter, strainers and surface skimmers. Fuel pool water is circulated by the pump through a filter which removes particulates larger than 5 micron size and through an ion exchanger to remove ionic material.

During refueling operations, this same system is used for purification of the refueling cavity. The 3 inch suction and discharge piping are routed from the FHB, through the penetration room in the RAB and into the refueling cavity inside the RCB. This same system is also used for filling and draining the refueling cavity. The existing suction and discharge nozzles inside the refueling cavity are 9 inches apart and are located at the far end of the refueling cavity away from the reactor vessel. In order to provide better filtration, the suction and discharge locations would require more separation in order that filtered water has a chance to disperse before being sucked back through the filtration system. Also, the suction line would need to be extended to the vicinity of the reactor vessel where it can do the most good. It is at the reactor vessel head that water first becomes contaminated requiring removal. It is also where we require visually clear water. This contaminated water needs to be drawn out before it has a chance to drift into the farther reaches of the refueling cavity.

Since we must be able to drain the refueling cavity using the fuel pool purification system, the existing suction location at the bottom of the refueling cavity must be maintained. However, we also need to have suction at the vicinity of the reactor vessel head which exists at a higher elevation. Therefore, additional suction line must tee off the existing suction pipe and be routed to the reactor vessel. Valves would need to be provided so that this line could be valved out during draining operations.

In addition to routing additional suction piping to the vicinity of the reactor head, the water clarity can be improved by installing a separate reactor cavity filtration system. This system would handle purification of the refueling cavity alone, relieving the fuel pool purification system in the FHB from this additional duty. With a separate reactor cavity filtration system, we would have full time purification of the refueling cavity as opposed to intermittant purification under the old system. Also, a separate system could be installed having greater capacity than the present fuel pool purification system.

Since the fuel pool purification pump is required for filling and draining, the reactor cavity filtration system would tee off the existing suction and discharge lines and we would thus maintain the capability of filtering the refueling cavity with the purification portion of the fuel pool system.

This PC/M does not constitute an unreviewed safety questions or involve a tech. spec. change. The reactor cavity filtration system will operate only during plant shutdown while refueling. The addition of this system should reduce the time required for removal and replacement of the fuel assemblies.

REACTOR VESSEL HEAD SHIELDING

ABSTRACT

THIS REPORT DEMONSTRATES THE ACCEPTABILITY OF THE TEMPORARY R.V. HEAD SHIELD AND ITS PERMANENTLY ATTACHED SUPPORT STRUCTURE, AS PROVIDED BY NUCLEAR POWER OUTFITTERS OF CRYSTAL LAKE, IL., FOR USE IN ST. LUCIE UNIT 1. IT ALSO FULFILLS THE REQUIREMENT FOR SUBMITTAL OF A "DESIGN PACKAGE" AS REQUIRED BY FLORIDA POWER AND LIGHT SPECIFICATION FOR ENGINEERING PROCURMENT, AND INSTALLATION FOR PC/M 89-82, DATED OCTOBER, 1982.

IV. SAFETY ANALYSIS

The reactor vessel head shielding consists of lead wool blankets hung around the reactor head during plant outages. The blankets are hung from a support system which is permanently attached to the reactor vessel head lift rig.

The shielding system does not perform a nuclear safety related function. It has been designed to withstand all applicable loads specified in the original plant design. A NUREG 0612 analysis has been performed to include the additional loading of the shielding system on the reactor head lift rig. The analysis shows that all NUREG 0612 requirements are satisfied.

Apropriate QA and QC requirements have been identified as well as procedures to assure the installation and use will conform to the design criteria.

The probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to the safety of the plant, previously evaluated in the FSAR, has not been increased. There is no possibility of accident or malfunction different than those previously evaluated. Also, there are no changes to the technical specification of the plant. Therefore, it can be concluded that the addition of the reactor vessel head shielding system does not pose an unreviewed safety question pursuant to 10CFR 50.59.

ST LUCIE PLANT - UNIT 1
TURBINE SUPERVISORY INSTRUMENTATION

INTRODUCTION

The turbine is a Westinghouse tandem compound four-flow exhaust 1800 RPM unit with one high pressure and two low pressure elements. The AC generator and brushless-type exciter are directly connected to the turbine generator shaft. The unit is provided with throttle valve steam chest assemblies located on each side of the high pressure turbine casing. The structural shapes of the casings and their methods of support are carefully designed to obtain free but symmetrical movements resulting from thermal changes.

Turbine Supervisory Instrumentation (TSI) is designed to provide optimum insight into the mechanical integrity of the turbine generator. This system utilizes a combination of monitoring, recording and logging to collect data on the operation of the turbine. The TSI system is used to sense subtle changes in the operation of the turbine generator. The items listed below are considered very important in the control of safe starting, loading and monitoring of the turbine:

A. Radial Vibration and Vibration Phase Angle
B. Rotor Eccentricity
C. Differential Expansion
D. Thrust Bearing Monitor
E. Case Expansion
F. Turbine Speed and Acceleration
G. Instrumentation Racks and Cable Terminations
H. Mimic Display and Annunciation Lights
I. PRobes, Cables and Conduit Installation

The Bently Nevada TSI system will ultimately replace the existing Westinghouse Turbine Supervisory equipment. However, the replacement will be accomplished in two (2) stages. The first stage is designed to install the Bently Nevada system without removing the exisitng Westinghouse equipment. While the second stage will be to disconnect the Westinghouse equipment and to complete the connection of the new system.

Included in the first stage of implementation are the installation of the brackets, the probes, the conduits, the electrical boxes, the TSI cabinet, and the annunciator mimic display for the Bently Nevada system.

The existing Westinghouse thrust bearing probe located at the coupling spacer between the jackshaft and the low pressure (LP) turbine 1B is to be replaced with new Bently Nevada probes. The mounting bracket for the Westinghouse probe will be modified to accept the new Bently Nevada probes. The existing Bently Nevada thrust bearing probes at the balance ring, being longer, could not be relocated and therefore will be removed.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59 a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The turbine supervisory instrumentation is non-safety related and non-seismic. The modification is being implemented for added protection of the turbine generator and to reduce potential unscheduled downtime. It improves plant reliability but does not otherwise affect the existing turbine design.

An evaluation for the impact of the added masses on the RTGB-101 and the change in the dynamic characteristics of the RTGB will be made by Civil Dept. It was found that the required modification has no significant impact on the dynamic characteristics of the board. The TSI cabinet is non-safety related but is located on the RAB elevation 43.0 level in the vicinity of safety related equipment. The cabinet was seismically analyzed and it was determined that it will maintain its structural integrity during a seismic event. Furthermore the displacement are sufficiently small to preclude interaction with adjacent equipment. The implementation of this PCM does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

THERMAL SHIELD REMOVAL - PHASE 3

SYSTEM DESCRIPTION

1.0 Function

Inspection of the core support barrel (CSB) and thermal shield (TS) revealed damage at the CSB to TS connection. Phase III of the Thermal Shield Removal allows the thermal shield segments to be removed from the Unit #1 refueling canal for eventual offsite shipment.

2.0 Design Description

Removal of the TS from the Unit #1 refueling pool will involve the following steps:

1. Loading 24" wide x 6' long TS segments into the transfer shield in the Unit #1 refueling pool.
2. Transport of the transfer shield with contents out of the Unit #1 containment and onto a truck which will transport the shield to Unit #2 via a prescribed route.
3. The transfer shield will be lifted from the truck and placed in the Unit #2 cask handling area where the transfer shield contents will be put into a shipping cask for offsite transport.

3.0 Operation

The operation of all equipment utilized in this effort shall be performed by a qualified operator.

SAFETY ANALYSIS

The transport of thermal shield segments does not involve an unreviewed safety question because:

1. (a) The probability of occurrence of an accident previously evaluated in the FSAR has not been affected since Unit #1 will not be operating while the transfer is taking place and no work will be performed in or over the Unit #1 spent fuel pool. Work being performed in Unit #2 is in the cask handling area and will not affect operation of Unit #2 as there is no fuel in the Unit #2 spent fuel pool. The haul route has already been evaluated for heavier loads than a fully loaded shield.

   (b) The consequences of an accident previously evaluated in the FSAR have not been affected since the shield drop accident will not result in offsite doses in excess of those accidents previously evaluated.

   (c) The probability of a malfunction of equipment important to safety has not been affected since the failure modes assumed as a result of the transfer process will not impact safety related equipment operation.

   (d) The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR remain unchanged based on item 1a, 1b, and 1c above.

2. (a) The possibility of an accident of a different type than any analyzed in the FSAR has not been created. The results of the analysis for a shield drop in excess of 11 feet showed that corrective action could be taken while still maintaining offsite doses within a fraction of 10 CFR 100 limits, thus remaining within the bounds of previous analyses.

   (b) The possibility of a malfunction of equipment important to safety of a different type than any analyzed in the FSAR remains unchanged for the reasons outlined above.

3. The transport of the thermal shield has no effect on the margin of safety as defined in the basis for any technical specification.

ROSEMOUNT TEMPERATURE TRANSMITTER REPLACEMENT

INTRODUCTION

This PC/M is for the installation of nineteen (19) new Rochester model temperature transmitters to replace the existing Rosemount models. The existing Rosemount model 442 temperature transmitters are no longer available from Rosemount. A comparable model by Rochester Instrument Systems will satisfy the operational conditions as well as meet the safety/seismic requirements. This modification will also incorporate the implementation of two (2) signal transmitters to increase the load capability of two (2) transmitter loops mentioned in the system description.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be incrased; or (ii) if a possibility for an accident or malfunction of a differenty type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specifications is reduced.

These new Rochester model temperature transmitters are located in a mild environment and are seismically qualified to IEEE-344-1975. In addition, these units are manufactured to the Rochester Quality Assurance Program for nuclear devices; therefore meeting traceability and reportability requirements. This PC/M is for replacement of nineteen (19) transmitters, thus providing a more accurate and reliable model. Therefore, this modification will not increase the probability of the occurence of any accident, whether previously evaluated or of a different type than previously evaluated and will not reduce the safety of the plant.

This PC/M does not reduce the margin of safety as defined in the basis of any technicaly specification, nor does it require a revision of a technical specification.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question, therefore prior Commission approval is not required for implementation of this PC/M.

Modification Description

This PC/M releases the new Diesel Generator Subsystem Flow Diagrams to the site. The following activities must be completed before the new flow diagrams can be issued as permanent plant drawings:

1.) All valves and instruments must be tagged in the field as per the new flow diagrams.

2.) Affected operating procedures must be reviewed to determine if revision is required to reflect the new tag numbers or flow diagram numbers.

Safety Analysis

1a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR:

Flow diagrams are not considered in evaluating FSAR accidents.

1b. With respect to the consequences of an accident previously evaluated in the FSAR:

Flow diagrams are not considered in evaluating FSAR accidents.

1c. With respect to the probability of malfunction of equipment important to safety previously evaluated in FSAR:

Flow diagrams are not considered in determining the probabilities of safety related equipment malfunctions.

1d. With respect to the consequences of malfunction of equipment important to nuclear safety previously evaluated in the FSAR:

Flow diagrams are not considered in determining the probabilities of safety related equipment malfunctions.

2a. With respect to the possibility of an accident of a different type than analyzed in the FSAR:

Flow diagrams are not considered in evaluating FSAR accidents.

2b. With respect to the possibility of a malfunction of a different type than analyzed in the FSAR:

Flow diagrams are not considered in determining the probabilities of safety related equipment malfunctions.

3. With respect to the margin of safety as defined in the basis for any technical specification:

Flow diagrams do not impact technical specification safety margins.

Based on the above, the new flow diagrams and the tagging/retagging of diesel generator valves and instruments are determined not to involve an unreviewed safety question. There are no system modifications involved.

ST LUCIE UNIT 1
REPLACEMENT OF LOAD WEIGHING SYSTEM FOR REFUELING AND FUEL TRANSFER MACHINES

ABSTRACT

This engineering design package covers the replacement of the load-weighing system for the refueling and fuel transfer machines. The existing system is manufactured by W C Dillon which no longer have spare parts available. The original equipment manfacturer, PAR System Corporation, will design, supply and install the new load weighing system. As discussed in UFSAR Chapter 9, this system is designed for safe handling and storage of fuel to and from the reactor. The equipment is normally used at 18 month intervals for a period of approximately three (3) weeks during which time it must operate continuously without maintenance or service. Also this system must be able to withstand loadings induced by the design base earthquake. Therefore, this PCM is classified as "Quality Related". This item does not require revision to the plant technical specifications, nor does it meet the criteria for an unreviewed safety question. Therefore, pursuant to 10CFR50.59 this modification can be made without prior commission approval.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possiblility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modification described in this PC/M replaces existing components associated with the refueling machine. The refueling machine is only required to operate for approximately three (3) weeks at eighteen (18) month intervals as per UFSAR Section 9.1.4.2. This system is not required for normal operation of the plant. It is only required for fueling and removal of fuel from the reactor, therefore this equipment is not required for safe shutdown of the plant. With regard to spent fuel handling accidents as described in UFSAR Section 15.4.3, the results of the fuel handling accident are not affected by this equipment change out. The digital readout will ensure that when removing a fuel assembly it is not damaged by excessive lifting forces. The digital scale should in fact provide more accurate information to the operator to better preclude this event.

The new equipment has been seismically analyzed to preclude its disalignment during a seismic event. Therefore, this will not impact the "light loads" accident analysis.

The failure of this component, not to function, would preclude further fuel movement until its repair. Redundancy of this system is not required. In as so much that this system is not required to shutdown the reactor, cool the core or cool another safety system or the reactor containment (after an accident), nor is it part of any system that reduces radioactivity released in an accident. Note, only these portions of a system that are designed primarily to accomplish one of the above functions, or the failure of which could prevent accomplishing one of the above functions, is designated safety related. The system is required to withstand loadings induced by the design bases earthquake. Therefore, this PCM is classified "Quality Related".

The modifications to the load cell supports shall be designed and analyzed by PAR Systems as to maintain the seismic integrity of the equipment. The results of their analysis will be reviewed by Ebasco.

The implementation of this PC/M does not require a change of the plant specifications.

"The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required."

ST. LUCIE UNIT #1
ESFAS POWER SUPPLY

ABSTRACT

This modification provides for the replacement of ESFAS measurement cabinet's instrument loop power supplies which are no longer available from the original equipment manufacturer. These supplies furnish the sensor current for containment pressure, and refueling water tank level.

This change performs a nuclear safety related function and is powered from Class 1E safety sources. This PC/M does not involve as unreviewed safety question.

SAFETY EVALUATION

a. The probability of occurrence of an accident previously evaluated in the FSAR has not been affected since the ESFAS is not utilized in determining the probabilities of accidents.

b. The consequences of an accident previously evaluated in the FSAR have not been changed since this modification does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

c. This modification does not affect any other safety related equipment.

d. The consequences of the malfunction in the FSAR have not been affected. Redundancy, function, or failure mode capacity have not been changed.

e. The possibility of an accident of a different type than analyzed in the FSAR has not been created since this modification does not affect any systems vital to safety or whose failure could directly result in an uncontrolled release of radioactive material.

f. The possibility of equipment malfunction of a different type than analyzed in the FSAR has not been increased.

The margin of safety as defined in the basis for any Technical Specification has not been changed since this modification does not change the performance, capabilities, or operating characteristics of the ESFAS.

In conclusion this change does not involve an unreviewed safety question.

REPLACEMENT OF VALVE SOLENOIDS

INTRODUCTION

The purpose of this PC/M is to replace twelve (12) existing ASCO and two (2) AVCO valve solenoids, no longer manufactured, with "Qualified" ASCO solenoid NP-8316 Series on various valves. The replacement solenoids are environmentally qualified IEEE-323-1974, IEEE-344-1975, and IEEE-382-1972. Also four (4) of the existing ASCO solenoid valve seats are to be rebuilt, using ASCO spare parts kit components (Elastomers)

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59 A proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurence or or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion.

The replacement of ASCO valve solenoids have been qualified to IEEE-323-1974, and IEEE-383-1972 requirements. The qualification documentation package has been provided by ASCO via their report AQS-21678/TR Rev A. The qualification test program simulated the effects of long-term operation under normal operating conditions and the effects of Design Basis Accident. (DBA). the effects included exposure to the environmental extremes of temperature, pressure, humidity, radiation, vibration, and chemical spray. With adequate margin, the qualification program demonstrated that the equipment can perform its specified function under the anticipated normal operating and DBA conditions.

The replacement of ASCO valve solenoids are one-for-one replacement of the existing ASCO solenoids. For the AVCO solenoids, valves FCV-25-2 and FCV-25-5, the addition of tubing fittings for the air supply system is required. However, there is no effect on the seismic qualification of those valves due to the additional weight of the required fittings.

The replacement of ASCO valve solenoids for the CCW isolation valves is also a one-for-one replacement of the existing ASCO solenoids. The replacement of these solenoids enhance the existing valves by installing environmentally qualified solenoids.

The implemetation of this PC/M does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question, therefore, prior Commission approval is not required for implementation of the PC/M.

ST. LUCIE UNITS 1 & 2

ICW BACKUP LUBEWATER BACKFLOW PREVENTOR (REA-SLN-85-41)

ABSTRACT

This Engineering Package covers replacement of the existing ICW service water backup lubewater supply backflow preventor with one that is currently manufactured. The existing backflow preventor vendor no longer manufactures these thus spare parts are difficult to obtain. This package is classified as non-seismic, non-nuclear safety related.

SAFETY EVALUATION

With respect to Title 10 of the code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (1) if the probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specifications is reduced.

The subject modification provides for replacement of the existing backflow preventor. This backflow preventor ensures that seawater does not backflow into the service water system thus contaminating the domestic water supply. With respect to 10CFR 50.59, failure of this backflow preventor: (1) does not increase the probability of an accident or malfunction of equipment important to safety since the supply is separated from all safety related equipment by a double check valve class break and is located remotely with respect to such safety related equipment and cannot fall on or hit such equipment; or (2) does not create possible accident scenarios not previously addressed by the Safety Analysis Report since it functions only as a system enhancement and does not have to function in any postulated accident conditions; or (3) does not affect or require changes to the Technical Specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question. Therefore prior commission approval is not required for implementation of this PC/M.

Additionally per the FSAR Section 9, this backup lubewater supply is not required to perform any safety related functions nor is the modification within any safety related or seismic boundaries. Therefore the modification is considered to be non-safety related, Quality Group D.

ST. LUCIE UNIT 1

Component Cooling Water Pump Journal Bearing Shell Material Change (REA SLN-421-86-2)

ABSTRACT

This engineering package covers replacement of the existing cast iron journal bearing shells on the component cooling water pumps 1A, 1B & 1C with shells made of carbon steel. The existing cast iron shells are no longer available and the manufacturer's replacement part is the carbon steel shell. As addressed in the Safety Evaluation, this modification is considered nuclear safety related. Based on the 10 CFR 50.59 review, it has been demonstrated that this change does not involve an unreviewed safety question, and the change will not affect plant safety. Additionally, no change is required to the Technical Specifications. Accordingly, prior NRC approval is not required for implementation of this design.

SAFETY EVALUATION

The Unit 1 Component Cooling Water pumps are nuclear safety related and are classified as ASME Section III, Class 3 Quality Group C components. They are required to provide a heat sink for safety related components associated with reactor decay heat removal for safe shutdown or LOCA conditions. The journal bearing shell material change affects both journal bearings in the 1A, 1B and 1C pumps.

Failure of the bearing shell (regardless of material utilized) and respective journal bearing will result in failure of the component cooling water pump. However, failure of a single pump has been previously evaluated and has been accounted for in the Component Cooling Water System design bases as identified in the FSAR. Measures exist to ensure adequate decay heat removal for safe shutdown or LOCA conditions should a single pump fail. Since the new shell parts are internal to the bearing housing, failure of an additional component cooling water pump simultaneous to the first pump failure is not possible based on single failure criteria. In addition, since the new shell material is functionally equal or better than the existing cast iron material, the probability of pump failure remains unchanged.

Based on the above evaluation and information provided in the Design Analysis, it can be demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 is not created. Since no other accident beyond what has been previously addressed in the FSAR has been identified and no other safety related equipment or components are affected as addressed in the failure modes analysis, the probability of occurence of analyzed accidents has not been increased. The replacement is equal or better to the equipment replaced. No new accidents or malfunctions are introduced as a result of this design change. Additionally, the margin of safety as defined in the Technical Specifications has not been reduced and no Technical Specification changes are required. Therefore an unreviewed safety question does not exist.

Since this modification does not involve an unreviewed safety question and does not change or alter the Technical Specifications, this change is acceptable with respect to 10 CFR 50.59 and does not require NRC approval prior to implementation.

PCM 050-186

ST LUCIE PLANT - UNIT NO 1
INSTRUMENT AIR UPGRADE
REA-SLN-481

ABSTRACT

This Engineering Package (EP) is for the installation of 2 new air compressors, 2 new desiccant air dryers and removal of the existing desiccant air dryer, afterfilter package and refrigerant air dryer which do not have sufficient capacity to accomodate the new compressors. One of the two new air compressors and one new air dryer will operate and the other will serve as a standby. The existing compressors will remain as backup, especially for loss of offsite power, since only these compressors can be loaded on the diesel generator. In addition, the backup air supply to the MSIVs and FCV9011 and FCV9021 will be removed since the new compressors will be able to supply adequate air flow at the required pressure.

This EP is classified as Non-Safety Related since the instrument air (IA) system compressors and associated equipment performs no safety function. The safety evaluation has determined that this EP does not constitute an unreviewed safety question and implementation of the EP does not require a change to the Plant Technical Specification. Therefore, prior NRC notification for implementation of this EP is not required.

This EP has no impact on plant safety and operation.

Supplement 1

The Supplement 1 provides revised design bases/analysis, safety evaluation, operation ~~and maintenance~~ guidelines and FSAR change package.

Although the safety evaluation has been revised, the original results of evaluation as stated above remain unchanged.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulation, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This EP is for the addition of two 100% capacity new compressors, two new desiccant air dryers and removal of the existing low capacity desiccant dryer, afterfilter package, refrigerant air dryer and supplemental air bottle racks and associated piping.

Safety Evaluation (Continued)

Failure of the instrument air compressors and components resulting in loss of IA and consequent affects as stated in the FSAR Subsection 9.3.1.3 have been reviewed. This modification does not add any new failure modes for the safety related air operated valves. However, existing solenoid valves, without regulators, whose maximum operating pressure differential capacity is less than 115 psi will be replaced via Design Equivalent Engineering Package (DEEP) No 154-188D. Malfunction, if any, of these solenoid valves will lead to the Fail Safe mode of the process valves. The IA system design pressure and temperature downstream of the IA aftercooler remain unchanged, therefore there is no concern for the valve actuators. This modification is therefore classified as non-nuclear Safety Quality Group D and non-class 1E.

The increase in IA requirements from 155 SCRM to 400 SCFM and pressure from 90-100 psig to 105-115 psig is based on FPL studies for the requirement of the IA.

Removal of the supplemental air bottle racks which are tied into the accummulators to maintain the MSIVs and feedwater FCVs air system pressure between 100-105 psig is considered acceptable because the new compressors (1C and 1D) will be able to provide adequate instrument air flow at the required pressure.

Based on the above description, the modification included in this EP is considered to be non-safety related. This EP does not involve an unreviewed safety question, and following are the bases for this justification:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The instrument air system compressors and associated equipment are not used directly in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and will have no effect on equipment vital to plant safety.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The components involved in this modification have no safety related function and no changes have been made to the normal operational design of the system with the compressors 1C and 1D in operation. In this mode the IA compressors 1A and 1B discharge valves V18109 and V18119 are closed to prevent IA leakage via these compressors. Similarly, whenever the IA compressors 1A and 1B are required to operate, valve V18586 is closed to prevent IA leakage via compressors 1C and 1D.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not included in the bases of any Technical Specification.

ST. LUCIE UNIT 1 PCM 052-186

Upper Guide Structure (UGS) Lift Rig Repair
REA-SLN-86-010

ABSTRACT

This engineering package covers the repairs and modification to the damaged UGS Lift Rig. The lift rig was damaged on November 6, 1985 when one of its attachment points failed during a UGS lift operation. Although the UGS Lift Rig is non safety-related, its failure could result in damage to nearby safety-related equipment. Therefore, quality-related design requirements have been imposed to assure QC inspection of the repairs and modification. The implementation of this PCM does not pose any unreviewed safety questions nor does it affect any safety-related equipment.

Supplement 1

This supplement documents the "as-built" configuration of the repaired UGS lift rig. It includes the actual column chord dimensions and incorporates the minor design changes that were made in the field during the repair effort. The original safety evaluation has been reviewed for the impact of the changes addressed by this supplement and it has been determined that it remains valid.

Safety Evaluation

This engineering package provides for the repair and modification of the UGS lift rig to comply with its original functional requirements. The UGS lift rig is not a safety-related piece of equipment and is only used during refueling operations when the plant is in the cold shutdown mode. Since failure of the lift rig while lifting the UGS could result in a load drop onto the reactor and irradiated fuel assemblies, this component is considered important to safety. For this reason, quality-related design requirements have been imposed to assure QC inspection of the repairs and modification.

The modified lift rig has been structurally reanalyzed for dead and seismic loads subject to the requirements of NUREG 0612, ANSI N14.6, and the applicable ASME and ASTM codes. The results of this analysis demonstrates that the new and existing components are all within allowable stress levels.

The containment heat sink, hydrogen generating source, and free volume analyses described in FSAR Section 6.2 are not affected by this modification, since the lift rig replacement parts are the same as or similar to those installed originally.

This modification does not change any assumptions made or conclusions drawn in the St. Lucie FSAR, and there is no new failure mode introduced that has not been previously evaluated in the FSAR. However, FSAR Figure 9.1-8 must be updated to reflect the repairs and modification to the lift rig.

For the above reasons, the repairs and modifications to the UGS lift rig will not increase the probability of occurrence nor the consequences of a design basis accident or malfunction of equipment important to the safety of the plant. Additionally, there will continue to be no possibility of an accident or malfunction different than those already evaluated in the FSAR. Finally, the margin of safety as defined in the Plant Technical Specifications has not been reduced. It is therefore concluded that this modification does not pose an unreviewed safety questions pursuant to 10 CFR 50.59 and does not affect any Technical Specifications.

HEATER DRAIN PUMP DEMINERALIZED WATER SUPPLY

ABSTRACT

This design package provides the required engineering for adding permanent piping from the demineralized water system to the Unit 1 heater drain pumps' mechanical seals. The piping will make available to the seals the necessary back up flushing water meeting the appropriate chemistry requirements. The back up water source is required during initial plant startup whenever the pumps sit idle.

Based on the failure modes analysis and 10 CFR 50.59 review, this modification does not impact any safety related equipment and is not relied upon for any accident prevention or mitigation. Thus it does not constitute an unreviewed safety question and is correctly classified as Non-Nuclear Safety Related. Implementation of this modification, therefore, does not require prior NRC approval.

Supplement 1

This package revision provides valve drawings for valves added by this PC/M and modifies the expiration date to reflect the correct format. The scope of work specified by this Engineering Package has not been affected by this revision. The safety classification and the safety evaluation as stated is correct and is not impacted.

SAFETY EVALUATION

The Unit 1 Heater Drain Pumps are located in a Non-Nuclear Safety Related system and as such are not required to function during any existing analyzed accident scenario. Therefore, modifications to these pumps affect only Non-Nuclear Safety Related, Quality Group D equipment.

Based on the failure mode analysis, failure of the demineralized water supply piping could result only in failure of the heater drain pumps. Since the piping and components are located remote from any safety related equipment or components, failure of this equipment will not inhibit operation of any safety related equipment or components.

Based on the above evaluation and information supplied in the design analysis it can be demonstrated that an unreviewed safety question as defined in 10CFR50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Since this design change does not alter or affect equipment used to mitigate accidents, the probability of occurrence of analyzed accidents remains unchanged.

- The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

  There is no new failure mode introduced by this change that has not been evaluated previously in the FSAR.

- The margin of safety as defined in the basis for any Technical Specifications has not been reduced.

  This change has no affect on any existing Technical Specifications.

10 CFR 50.59 EQ LIST REVISION

ABSTRACT

This engineering design package provides the vehicle for updating several areas of equipment qualification. This package includes corrections to the 10CFR50.49 list, changes in maintenance requirements, and various documentation corrections.

This design package is considered nuclear safety related because it affects equipment falling under the scope of 10CFR50.49. This package does not represent an unreviewed safety question since it deals strictly with enhancing the present documentation used to qualify equipment at St. Lucie.

Safety Evaluation

This engineering design package provides for several documentation changes to the present St. Lucie Unit No. 1's equipment qualification documentation. This documentation will affect the future procurement of various safety related components and assist in validating the components' ability to function during a design basis accident. Therefore, this design package is considered safety related.

The documentation changes addressed in this package range from corrections of typographical errors on the 10CFR50.49 list to reviews of a vendor's equipment qualification test report. None of the changes require physical modification to any plant system. They do however, affect the future maintenance of various instruments.

Based on the above and the information supplied in the design analysis it can be deomonstrated that an unreviewed safety question as defined by 10CFR50.59 does not exist. Since this change does not alter any equipment used to mitigate accidents, the probability of occurrence of an analyzed accident remains unaffected. This design package only enhances the environmental documentation of various instrumentation and in no way affects the plant design, therefore the possibility of an unanalyzed accident or malfunction has not been created.

The surveillance requirements of the Technical Specifications were reviewed against the equipment qualification maintenance requirements addressed in this package in the design analysis. No Technical Specification changes are required by this design package.

In conclusion, the changes proposed by this design package are acceptable from the standpoint of nuclear safety because they do not involve an unreviewed safety question and no Technical Specification changes are required.

SUPPLEMENT #1

The revisions incorporated by supplement #1 do not affect the original safety evaluation.

ST LUCIE PLANT - UNIT NO. 1
UPPER GUIDE STRUCTURE LIFT RIG LOAD TEST FIXTURE
(REA SLN-86-010)

ABSTRACT

This Plant Change/Modification (PC/M) consists of the fabrication and installation of a temporary structure which will be used to load-test the Upper Guide Structure (UGS) Lift Rig after it is repaired and modified. The structure will be attached to the reactor missile shields in their laydown area. The static load test will be performed by the reactor polar crane using the missile shields as test loads. After the load test, this temporary structure will be removed from the containment.

This PC/M is not classified as safety-related since the load test structure will not perform or affect any safety-related function. Although failure of the test fixture will not result in any interaction with safety-related equipment or functions, Quality Related requirements will be applied to the design because of the importance of thé UGS Lift Rig to plant operations. The Quality Related design requirements assure Q. C. inspection of the installation and independent verification of the design of the load test structure.

This PC/M does not constitute an unreviewed safety question. The only effect on plant operations will occur during the refueling outage UGS Lift Rig load test. The implementation of this PC/M does not affect any safety-related equipment.

SAFETY EVALUATION

With respect to title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification is not classified as safety related since the load test structure does not perform any safety function. Failure of the load test structure before, during, or after the load test could not affect any safety-related equipment or function since the structure will be installed in the containment only during the refueling outage and is located away from any safety-related components. Failure of the load test structure during the load test could damage the UGS Lift Rig. The UGS Lift Rig is not a safety-related component, but it is important since its failure during lifting of the UGS could result in a load drop onto the reactor and irradiated fuel assemblies. For this reason, the load test structure has been designed using Quality Related requirements.

The containment heat sink analysis inventory of hydrogen generating items and free volume assumptions described in FUSAR Section 6.2, are not affected by this modification, since the load test structure is temporary and will be removed from the containment prior to plant operation.

The modifications included in this PC/M do not involve any unreviewed safety questions because:

i The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification will have no effect on equipment performing a safety function.

ii There is no possibility for an accident or malfunction of a different type than any previously evaluated since the load test structure performs no safety function and no changes have been made to any operational design. Failure of the load test structure will have no effect on any safety-related equipment or function.

This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PC/M does not require a change to the plant technical specifications.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question; therefore, prior Commission approval is not required for implementation of this PC/M.

ST LUCIE PLANT - UNIT NO 1
CONDENSATE PUMPS EXPANSION JOINT REPLACEMENT
REA-SLN-85-153

ABSTRACT

The existing expansion joints in the Condensate Pumps suction are made of elastomeric material which has deteriorated due to aging. The deterioration is so severe that the Condensate System is susceptible to air permeation. To correct this problem the existing expansion joints will be replaced with new stainless steel expansion joints.

The Condensate System considered in this Engineering Package is non-safety related. Accordingly, this Engineering Package is classified as non-safety related. The safety evaluation has shown that this EP does not impact plant safety and operation and does not constitute an unreviewed safety question or require a Technical Specification change. Therefore, prior NRC approval is not required for implementation.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

The Condensate Pump expansion joints are located between the Condenser and Condensate Pumps and are utilized to absorb differential thermal expansion between these components. They do not perform any safety related function and therefore are classified as non-safety class, Quality Group D. The failure analysis has shown that failure of these components will not affect any safety related equipment.

This non-safety related modification does not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The St Lucie Unit No 1 FSAR, Section 10.4.6 describes the Condensate System. This Section indicates that the portion of Condensate System being modified and the expansion joint are not designed to seismic Class I standards and they are not used in any safety analysis for accidents or malfunction of equipment. This system is non-safety related and will have no effect on equipment vital to plant safety.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created. The components involved in this modification do not perform any safety related function. No changes have been made to the operational design of the Condensate System.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not directly included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specification.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

PCM NO. 119-186
REV NO 1

ST LUCIE PLANT - UNIT NO 1

10CFR50.49 ENVIRONMENTAL QUALIFICATION LIST REVISION

REA SLN-85-58

ABSTRACT

This Engineering Package provides the vehicle for updating several areas of equipment qualification. This package includes corrections to the 10CFR50.49 list, changes in maintenance requirements, and various documentation package corrections.

This Engineering Package (EP) is considered Nuclear Safety Related because it affects equipment falling under the scope of 10CFR50.49. This package does not represent an unreviewed safety question since it deals strictly with enhancing the present documentation used to qualify equipment at St Lucie Unit No 1 and no physical plant modifications are required by the EP. The safety evaluation of this package indicates that a change to the Plant Technical Specifications is not required. Removal of equipment from the 10CFR50.49 list does not affect plant safety and operation.

Supplement 1

This EP revision adds terminal blocks to the 10CFR50.49 list and their associated Equipment Qualification Documentation Package 8770-A-451-17.0 "Amerace Terminal Blocks". The equipment and EQ Documentation Package does not affect the original safety evaluation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This Engineering Package provides for several changes to the present St Lucie Unit No. 1's 10CFR50.49 list. This documentation will affect the future procurement of various safety related components and assist in validating the components' ability to function before, during and after a design basis accident. Therefore, this EP is considered Nuclear Safety Related.

The documentation changes addressed in this package range from corrections of typographical errors on the 10CFR50.49 list to additions and deletions of equipment as a result of EQ documentation packages reviews. None of the changes require physical modification to any plant system. They do, however, affect the future maintenance of various equipment.

The possibility of new Design Basis Events (DBEs) not considered in the UFSAR is not created since this change does not alter any equipment used to mitigate accidents. This modification is an enhancement of the environmental qualification documentation of various equipment and in no way affects the plant design.

Due to the fact that this EP does not affect or modify any cables essential to safe reactor shutdown or systems associated with achieving and maintaining shutdowns, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore the proposed design of this package is in compliance with the applicable codes and UFSAR requirements for fire protection equipment.

Since this modification involves no physical modifications to safety related equipment and changes in the maintenance schedules will not result in failure of equipment, the degree of protection provided to Nuclear Safety Related equipment is unchanged. Removal of equipment from the 10CFR50.49 list does not affect the plant's safety. The probability of malfunction of equipment is unchanged. The probability of malfunction of equipment important to safety previously evaluated in the UFSAR remains unchanged. The consequences of malfunction of equipment important to safety previously evaluated in the UFSAR are unchanged. The possibility of malfunctions of a different type than those analyzed in the UFSAR is not created.

Based on the above, the modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created by this modification. Function, mounting and the ability to withstand harsh environmental conditions have not been altered and this modification does not affect any other safety related equipment.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since this modification does not change the requirements of the Technical Specifications.

The implementation of this PCM does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST LUCIE PLANT - UNIT NO 1
STARTUP TRANSFORMER LOCKOUT DISCONNECT SWITCHES
REA-SLN-677-10

ABSTRACT

This Engineering Package (EP) provides for the installation of disconnect switches in the plant startup transformers lockout relay circuits. The purpose of this change is to facilitate lockout relay maintenance testing while eliminating the possibility of inadvertent plant trip by propagation of a lockout relay trip during lockout relay maintenance test.

This EP is classified as Quality Related since lockout circuit actuation will trip the startup transformer and would result in plant operation under Limiting Conditions for Operation as defined in the Plant Technical Specification. Subsequent loss of offsite power to the station buses could affect plant trip, starting and loading Emergency Diesel Generators. A review of the changes to be implemented by this PCM was performed in accordance with the requirements of 10CFR50.59. As indicated in the Safety Evaluation (Section 3.0), this PCM does not involve an unreviewed safety question, nor does it require a revision to the plant Technical Specifications. This modification will have no effect on plant safety or operation. Prior Commission approval is not required for the implementation of this PCM.

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated, in FSAR Section 8.3.1.1, is not increased since the startup transformers and their lockout trip circuits are not Nuclear Safety Related equipment. Failure of the test switches will not affect the availability of the Emergency Diesel Generators in the event of loss of offsite power (LOOP).

ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the startup transformers are used for plant startup and shutdown. In the event of test switch failure which may result in unavailability of the preferred offsite power source (start-up transformer), the emergency diesel generators can provide the power required for safe shutdown as previously evaluated in FSAR Section 8.3.

(iii) This modification does not change the margin of safety as defined in the basis for any Technical Specification. This has been determined based on the fact that this modification does not exceed the limitations of Plant Technical Specification and does not affect safe reactor shutdown, the mitigation of the consequences of a design basis event (DBE), or the control of radioactive releases to the environment.

This EP affects equipment that is Non-Nuclear Safety Related. However, since startup transformer failure, and startup transformer trip signal actuation will result in plant operation under Technical Specification limitations, this EP is classified as Quality Related.

This EP has no effect on cables essential to safe reactor shutdown or components listed on the Essential Equipment List. There are no changes to equipment involving 10CFR50 Appendix "R" Fire Protection requirements (see attachment 7.1). Thus, the proposed design of this package is in compliance with the applicable codes and FSAR requirements for fire protection equipment.

Implementation of this PCM does not require a change to the Plant Technical Specifications and may be implemented without prior Commission approval.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question nor does it require a revision to the plant Technical Specifications,and prior Commission approval for the implementation of this PCM is not required.

ST LUCIE UNIT NO 1
AUTO LEAK RATE TESTER FOR PERSONNEL AIR LOCKS
REA-SLN-86-005

ABSTRACT

This engineering package allows for the replacement of the existing Volumetrics Automatic Leak Test System model 14324 (obsolete) with the currently available Volumetrics model 14330-2. This system will provide both local and remote (main control room) alarm on failure of leak rate test. Since there are no essential cables associated with this EP, this package has no impact on 10CFR50 Appendix "R" requirements.

The leak rate test system is not required for safe reactor shutdown and does not serve to mitigate the consequences of a design bases event (DBE) and is therefore not safety related equipment. However, since this package includes modifications to control board annunciators, and is required to maintain the limits of St Lucie - Unit 1 Technical Specification Section 3/4.6, "Containment Air Locks," it is considered Quality Related.

This engineering package will restore automatic test capabilities to the personnel air locks and reduce manpower requirements to manually operate the existing leak rate testers. The existing interior door tester, currently inside the containment vessel, is relocated outside the exterior door so as to minimize personnel contact with the RCA.

The implementation of this PCM does not require any change to the St Lucie - Unit 1 Technical Specifications. The modifications do not involve an unreviewed safety question and prior Commission approval for the implementation of this package is not required.

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The proposed modification affects personnel air hatch leak rate testing which provides local and remote (main control room) alarm on test failure.

The probability of occurence of a DBE previously addressed in the FUSAR is not affected by this modification. This system will in fact decrease the probability of a breach of containment by assuring containment integrity. Failure of this system to operate properly will be annunciated thereby preventing the performance of inaccurate testing. The possibility of new DBEs not considered in the FUSAR is not created since the design philosophy has been previously discussed in the FUSAR. This modification is an enhancement to a pre-existing system as is being performed to provide the highest caliber equipment possible.

Due to the fact that this EP does not involve any cables essential to safe reactor shutdown or systems associated with achieving and maintaining safe shutdown conditions, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore the proposed design of this package is in compliance with the applicable codes and FUSAR requirements for fire protection equipment.

The leak detection system is not necessary for safe reactor shutdown nor does it serve to mitigate the consequences of a design bases event (DBE). Hence, this package is not safety related. However, since a remote alarm in the Main Control Board is provided by the Containment Personnel Air Lock Automatic Test System and modifications to the annunciator panels are included in this design, this package is Quality Related. Quality Control Engineer shall witness the installation of the new annunciator tile in the Control Room as well as the auto leak rate tester in the personnel access area.

As the evaluation of failure mode (Section 2.2.7) indicates, the failure mode of this system has no effect on safety related systems or equipment. Hence the degree of protection provided to nuclear safety related equipment is unchanged. The probability of malfunction of equipment important to safety, previously evaluated in the FUSAR remains unchanged. The consequences of malfunction of equipment important to safety previously evaluated in the FUSAR are unchanged. The possibility of malfunctions of a different type than those analyzed in the FUSAR is not created.

The implementation of Quality Related PC/M 131-186 does not require a change to the plant technical specifications, nor does it create an unreviewed safety questions.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST. LUCIE UNIT 1
QSPDS SOFTWARE MODS

ABSTRACT

This Engineering Package covers the modifications to the previously certified software of the Qualified Safety Parameter Display System (QSPDS). The modifications consist of additions to assist the plant operator in accident monitoring. There is no major QSPDS hardware modification as a result of this PC/M. However, the exchange of identical Erasable Programmable Read Only Memory (EPROM) chips were required as a results of software modifications.

This Engineering Package is safety related because it involves modifications to a safety graded system QSPDS. The QSPDS is a safety grade class 1E processing and display system used for post-accident monitoring. The hardware and software changes of this PC/M were evaluated against 10CFR 50.59. The results of the evaluation are that there is no unreviewed safety question.

SAFETY EVALUATION

This engineering package is safety related because it involves a modification to a safety graded system. We have evaluated the effects of this PC/M with respect to regulation 10CFR50.59. The two applicable items for the QSPDS are:

a) Unreviewed Safety Questions

There are no major hardware changes due to this PC/M, since the exchanged hardware (EPROM's) are identical to original. The software changes consist of the addition of one display page which is consistent with the requirements of format, content and visibility of the original design. Therefore, there is no increase in the probability of occurrence or consequence of an accident, or malfunction of equipment because of this modification to the QSPDS. The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR has not been created. In addition the margin of safety is not decreased by this PC/M. Instead, the safety margin is considered to be increased due to the increased visibility of the safety parameters by the operator as a result of this PC/M.

b) Technical Specifications

The requirements established in the Technical Specification for the QSPDS are unaffected by this PC/M. The changes of this PC/M did not affect design, nor previous function, it merely improved Human Factors Engineering considerations.

BECKMAN WASTE GAS SYSTEM OXYGEN ANALYZER REPLACEMENT
ST LUCIE PLANT - UNIT NO 1
REA-SLN-86-030

ABSTRACT

In order to increase the availability of the oxygen analyzers for frequent monitoring of the oxygen levels in the waste gas decay tanks, the existing oxygen analyzers will be replaced with updated oxygen analyzers having an analytical element designed for sampling services in either liquid or gaseous sample streams.

The inherent design features of the replacement analyzers will include the design and operational criteria for sample monitoring and installation in potentially hazardous locations, therefore, this design shall be considered as Quality Related.

The implementation of this Engineering Package will have no impact on plant safety or plant operation.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this Engineering Package (EP), this PCM does not involve an unreviewed safety question, nor does it require a revision to the technical specification; therefore, prior Commission approval is not required for implementation of this PCM.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased since the Oxygen Analyzers are used for frequent monitoring of oxygen concentrations in the waste decay tanks and as described in PSL-1 FSAR Subsection 11.3.2.1 this system's function is not essential for the safety of the plant. The replacement of Oxygen Analyzers will provide control improvements to maintain the Waste Gas Analysis System functional with significant reduction in system maintenance and component replacements.

ii) The possibility of an accident or malfunction of a different type other than any evaluated previously in the safety analysis report is not created since:

a) This installation is in accordance with the Code of Federal Regulation 10 CFR 50.49 and no impact is incurred by this installation.

b) The new equipment mountings and added components have been analyzed in accordance with the specification for the Design Fabrication and Erection of Structural Steel for Building, and it has been determined that the stresses with the new equipment are less than the panel stresses with the original equipment.

c) This installation is in accordance with the Code of Federal Regulation 10 CFR 50.49 and has been determined to have no impact on the Environmental Qualification criteria since the equipment does not monitor or mitigate the event causing the harsh environment.

d) The Waste Gas System Oxygen Analyzers are neither required for safe shutdown nor for mitigating the consequences of an accident.

iii) The margin of safety as defined in the bases for any Technical Specifications is not affected by this EP since the components involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PC/M does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10 CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required.

ST LUCIE PLANT - UNIT 1
ANNUNCIATOR NUISANCE ALARMS
REA-SLN-86-052

ABSTRACT

This Engineering Package (EP) covers the modifications of five annunciator circuits in the Main Control Room. Existing logic, circuit configuration and components will be changed in the Reactor Turbine Generator Boards (RTGBs) so as to eliminate existing nuisance conditions caused by erroneous alarm indication of these five annunciator circuits. By implementing this EP, these circuits will be consistent with the "Dark Annunciator" concept which allows for lighted annunciators during off-normal conditions only.

This EP is classified as Nuclear Safety Related since it involves the interposing of a control relay in a safety related circuit (hydrogen analyzer) and the extension of safety related power supply cables (10482E, 10482L, and 10485H). The safety evaluation has determined that this EP does not involve an unreviewed safety question and does not require a change in the plant technical specifications. This PCM may be implemented without prior NRC approval.

SUPPLEMENT 1

This Engineering Package Revision covers modification of the six annunciator circuits associated with annunciated windows P-30, P-35, P-36, P-42, Q-40 and X-5 in the Control Room. These modifications, which include relocation of local reset switches, installation of reflashers and logic modifications, will eliminate the nuisance alarm status of the six annunciators. By implementing this PCM Supplement, these six annunciators will be brought into compliance with the "Dark Annunciator" concept of NUREG 0700 "Guidelines for Control Room Design Review".

The original Safety Evaluation has been revised. The Safety Evaluation still concludes, however, that this EP does not involve an unreviewed safety question, or a change to the technical specifications. Therefore, prior NRC approval is not required for implementation of the PCM.
The intent of the original Safety Evaluation is not affected by this supplement.

SUPPLEMENT 2

This Engineering Package Revision covers modification of the three annunciator circuits associated with annunciator windows N-45, R-50, and S-24 in the Control Room. These modifications, which include the installation of four (4) relays, evaluation to support setpoint modifications and drawing corrections, will eliminate the nuisance alarm status of the annunciators.

The Safety Evaluation of Supplement 1 to this PCM has been revised. The Safety Evaluation still concludes that this EP does not involve an unreviewed safety question or a change to the Technical Specifications. Therefore, prior Commission approval is not required for implementation of the PCM. The intent of the original Safety Evaluation is not affected by this supplement.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed changed shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the annunciators serve no function in the control of plant operations or safe shutdown. Electrical separation is provided between redundant safety related wiring and components and annunciator logic which is separated to protect control functions from being affected by annunciation circuit failure.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no changes have been made to the operational design of any control circuits or associated systems.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification.

Since this EP affects equipment that is identified as Nuclear Safety Related (Hydrogen Analyzer and SI Tank Isolation Valves 3614, 3624, 3634 & 3644) and requires the extension of Nuclear Safety Related power supply cables (10482E, 10482L, and 10485H), it is considered Nuclear Safety Related.

Due to the fact that the EP does not involve any cables essential to safe reactor shutdown or systems associated with achieving and maintaining safe shutdown conditions, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore, the proposed design of this package is in compliance with the applicable codes and St Lucie - Unit 1 FSAR requirements for fire protection equipment.

Implementation of Nuclear Safety Related PCM 140-186 and Supplements 1 & 2 to the same PCM do not require a change to the plant technical specifications and may be implemented without prior NRC approval.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PCM, as well as Supplements 1 & 2 to the same, is not required.

ST. LUCIE UNIT 1

ICW DISCHARGE PIPE ZINC RIBBON
(REA-SLN-85-137-12)

ABSTRACT

This engineering package covers the installation of zinc ribbon sacrificial anodes in the Intake Cooling Water (ICW) discharge piping. The anodes will be installed in the pipe beginning at the Component Cooling Water (CCW) wall and extending to the discharge canal. The zinc anodes will provide cathodic protection for the internal surface of the epoxy coated carbon steel pipe. The PC/M is classified as Quality Related because the sacrificial anodes are to be installed inside the Safety Related ICW pipe. The anodes perform no safety function, do not affect Plant safety or operation and the installation does not constitute an unreviewed safety question or require a change to the plant Technical Specifications.

SAFETY EVALUATION

The sacrificial anodes to be installed in the ICW discharge pipe as described in this design package do not have a safety function. As demonstrated by the failure modes evaluation in the design analysis, the principle effect on safety is the potential for internal pipe coating damage in the event that a zinc ribbon pipe attachment fails. For this reason, Quality Related design requirements have been applied and the modification is classified as Quality Related.

Based upon the above and information supplied in the design analysis, it can be demonstrated that an unreviewed safety question as defined by 10CFR50.59 does not exist.

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety evaluated in the safety analysis report has not been increased because the zinc anodes are installed downstream of all active ICW System components evaluated in the FSAR. Therefore, there is no interaction with the evaluated system components.

ii) The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created because, as demonstrated in the design analysis, the worst case failure of the zinc anode pipe attachments would have no impact on the ICW System capability to perform its design functions as specified in FSAR Section 9.2.1. In addition, the pipe attachments (thermit welds) have been evaluated and the determination has been made that the bonding process will not cause detrimental metallurgical conditions or impact the pipe coating systems.

iii) The margin of safety as defined in the basis for any Technical Specification has not been reduced. The installation of the zinc anodes will have no impact on the structural integrity of the ICW system piping or the design flow requirements of the system. For this reason, it is concluded that the margin of safety has not been decreased.

10CFR50.59 allows changes to a facility as described in the FSAR if an unreviewed safety question does not exist and if a change to the Technical Specifications is not required. As shown in the preceding sections, the change proposed by this design package does not involve an unreviewed safety question because each concern posed by 10CFR50.59 that pertains to an unreviewed safety question can be positively answered.

In conclusion, the change proposed in this design package is acceptable from the standpoint of nuclear safety, does not involve an unreviewed safety question, does not require a change to the Technical Specifications and does not require prior NRC approval.

FLORIDA POWER & LIGHT COMPANY
ST LUCIE PLANT - UNIT NO 1
SIMULATOR TRAINING FACILITY GAI-TRONICS
REA-NONE

ABSTRACT

This Engineering Package (EP) includes modifications to provide Gai-Tronics communication capability, including emergency alarms and instructions, for the Simulator Training Facility. The new Gai-Tronics equipment will be tied into the existing St Lucie Unit 1 Gai-Tronics System at the Service Building.

The modifications presented by this Engineering Package impact only non-safety related equipment. However, two conduit supports are being added to a safety related block wall. The additional loading has been reviewed and determined to have no effect on the structural integrity of the wall. The Gai-Tronics modifications are required in order to assure compliance with the St Lucie Plant Emergency Plan. Also, the power supply for the Gai-Tronics System is supplied from a vital AC source. Therefore, this package is classified as Quality Related.

A review of the changes to be implemented by this PC/M was performed against the requirements of 10CFR50.59 as indicated in Section 3.0 of this EP. As a result, the expansion of the Gai-Tronics System to include the Simulator Training Facility does not constitute an unreviewed safety question and will not affect plant safety and its operation. The implementation of this PCM does not require a change to the Plant Technical Specifications. Therefore, prior commission approval is not required for implementation of this EP.

Safety Evaluation

PCM 013-187

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This Engineering Package provides the engineering and design details required to expand the St Lucie Unit 1 Gai-Tronics Communication System to include the Simulator Training Facility. Gai-Tronics speakers will be located throughout the Simulator Training Facility in order to assure complete coverage for the emergency alarm signals (e.g., site evacuation alarm). One handset/speaker amplifier will be located in the Simulator Facility to provide two-way Gai-Tronics communication capability.

Based upon the expansion of the Gai-Tronics System presented by this EP, the breaker for circuit 33 of non-safety related Vital AC Bus No 1 (the power feed for the Gai-Tronics System) will be increased from a 20 amp to a 30 amp breaker. Also, the feeder cable from Vital AC Bus No 1 to the Gai-Tronics Power Distribution Cabinet (Cable 11201F) will be changed from 1-2/C #10 to 2-1/C #4. All other supplemental equipment (i.e., the 70 amp fuse, isolation transformer and feeder cables 11201Y and Z associated with the power supply) remain unchanged. The additional load presented by this modification (3.6 amps maximum at 120V AC) is considered insignificant and will have no impact on loading. Also, the increase in breaker size will not affect circuit breaker coordination. A fault on Vital AC Bus No 1, circuit 33 will not result in the loss of the entire vital bus (i.e., the circuit breaker and/or fuse for circuit 33 will clear the fault before any upstream breaker opens).

Based upon the feeder cable changeout, a new non-safety related conduit is required in the RAB from Vital AC Bus No 1 to cable tray C3. Two new conduit supports are added to seismically designed block wall #167. The additional loading has been reviewed and it has been concluded that neither the stress levels in the wall nor its fundamental natural frequency are significantly affected by this modification.

The Gai-Tronics System and the 120V Vital AC System are not safety related systems. The expansion of the Gai-Tronics System to include the Simulator Training Facility has no impact on any other plant systems or operations.

The Gai-Tronics System is not required to mitigate or monitor any result of an accident.

Failure of this system has no impact on previously generated safety analysis reports. The margin of safety as defined in the bases for any Technical Specification is not reduced.

The implementation of this PC/M does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve any unreviewed safety questions, and prior Commission approval for the implementation of this PCM is not required.

CCW, TCW AND OBCW VALVE ACTUATOR REPLACEMENT
ST LUCIE PLANT - UNIT NO 1
REA-SLN-86-031

ABSTRACT

This Engineering Package (EP) is to replace two (2) existing Bettis actuators, including all accessories, on the shutdown cooling heat exchanger isolation valves, I-HCV-14-3A and 3B. These actuators open the valves on the safety injection actuation signal (SIAS) and supply cooling water to the heat exchanger. During the outage of 1985, the actuators were inspected and found to have cylinder wear. The existing actuators, model 746-X-2SR-42, are no longer manufactured and spare parts are not available, therefore they are being replaced with Bettis actuators, model NT312-SR4-M3. The new actuators operate in the same manner and will perform the same function as the existing actuators.

The modification considered in this EP is on the Component Cooling Water System. The valves and actuators are Class 3 Seismic Category I, therefore this EP is classified as Safety Related.

Design details are provided for the installation of the new actuators and all accessories on the existing valves.

The safety evaluation has shown that this EP does not constitute an unreviewed safety question and implementation of this EP does not require a change to the Technical Specification. Therefore, prior NRC approval is not required for implementation.

The implementation of the EP will have no impact on plant safety or operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modification considered in this EP is classified as safety related because the shutdown cooling heat exchanger isolation valves, I-HCV-14-3A and 3B, and actuators are Safety Class 3, Seismic Category I. In the modification the two (2) existing Bettis actuators, model 746A-X-2SR-42 will be replaced with new actuators, model NT312-SR4-M3, because the existing actuators are no longer manufactured and spare parts are not available. The actuators open the isolation valves in the event of an SIAS. The new actuators will perform the same function in the same manner as the existing actuators. No new failure modes are created. On loss of power or loss of air the valves "fail open".

The modifications included in this Engineering Package do not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated are not increased because the new actuators will perform the same function in the same manner as the existing actuators.

ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated. This EP does not modify the intended operation or test requirements of the system because the new actuators will perform the same function in the same manner as the existing actuators.

iii) This modification does not change the margin of safety as defined in the basis for any Technical Specification because it neither changes the design parameters of the CCW system nor does it change the CCW design flow or functional requirements.

The implementation of this change does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question therefore, prior NRC approval for the implementation of this PCM is not required.

ST. LUCIE UNIT 1

Drain for Pipe Line 2-WM-D40
(REA-SLN-86-86)

ABSTRACT

This Engineering Package (EP) is for preparation and issuance of necessary changes in documentation to reflect permanent installation of a drain connection/tap installed in the spool piece downstream of FIT 6648 in line 2-WM-D40 by Circuit Alteration Tag No. 2025. FIT 6648 is located in a vertical run of piping in the Waste Management System off gas header in the RAB. The off gas header must be periodically drained due to condensation in the piping. FIT 6648 acts like a check valve and prevents water from going to a low point drain. The drain connection/tap located in the spool piece allows water above (downstream) FIT 6648 to be drained out of the header.

The existing piping at the drain connection/tap location is non-seismic, Quality Group D, performs no safety related function, has no affect on safety related equipment, has no affect on plant safety and operation, and the gas flowing through the pipe is acceptable for discharge to the environment. But, since the gas and condensation in the piping has the potential to be radioactive, the EP is classified as quality related. Based on a failure mode analysis and 10 CFR 50.59 review, the change proposed by this EP is acceptable from the standpoint of nuclear safety, it does not involve an unreviewed safety question, and does not require any changes to Technical Specifications. Therefore, prior NRC approval is not required for implementation of the modification.

SAFETY EVALUATION

This EP is for preparation and issuance of necessary changes in documentation to reflect permanent installation of a drain connection/tap installed in the spool piece downstream of FIT 6648 in line 2-WM-D40 by Circuit Alteration Tag No. 2025. FIT 6648 is located in a vertical run of piping in the Waste Management System off gas header. The off gas header must be periodically drained due to condensation in the piping. FIT 6648 acts like a check valve and prevents water from going to a low point drain. The drain connection/tap located in the spoolpiece allows water above (downstream) FIT 6648 to be drained out of the header.

The existing piping at the drain connection/tap location is non-seismic, Quality Group D, performs no safety related function, has no affect on safety related equipment or functions, and the gas flowing through the pipe is acceptable for discharge to the environment. But, since the gas and condensation in the piping has the potential to be radioactive, the EP is classified as quality related.

Based on the above and the information supplied in the design analysis, it can be demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  The drain connection/tap and associated piping are not used in any safety analysis for accidents or malfunction of equipment. This modification is non-nuclear safety related and will have no effect on equipment vital to plant safety. Based on this, the probability of occurrence or the consequences of all analyzed accidents remain unchanged.

- The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

  This modification is non-nuclear safety related and based on the failure modes analysis will have no effect on safety related equipment and functions.

- The margin of safety as defined in the basis for any Technical Specification has not been reduced.

  No function of the subject drain and associated piping is controlled by or in the basis for, any Technical Specification. Thus, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

10 CFR 50.59 allows changes to a facility as described in the FSAR if an unreviewed safety question does not exist and if a change to the Technical Specification is not required. As shown in the preceding sections, the change proposed by this design package does not involve an unreviewed safety question because each concern posed by 10 CFR 50.59 that pertains to an unreviewed safety question can be positively answered. Also, no change to the Technical Specifications is required based on the above evaluation.

In conclusion, the change proposed in this design package is acceptable from the standpoint of nuclear safety, does not involve an unreviewed safety question, and does not require any change to Technical Specifications. Therefore, prior NRC approval is not required for implementation of the modifications.

ST. LUCIE UNIT 1

CONDENSER OUTLET TUBE SHEET AND WATERBOX COATINGS

ABSTRACT

This engineering package addresses the addition of an epoxy coating to the condenser outlet tube sheets and waterboxes. This modification will enhance the corrosion resistance of the tube sheets and waterboxes and allow reduction of the cathodic protection system potentials and current densities.

The condensers and the plant circulating water system are classified as non-nuclear safety related and therefore, the modification addressed in this engineering package does not consistute an unreviewed safety question. Furthermore, the addition of a protective coating to the condenser outlet tube sheets and waterboxes does not require a change to the plant Technical Specifications.

Supplement 1

This supplement consists of the correction of the drawing number listed under Section 11.2 of this Engineering Package and the correction of a typographical error in the abstract. These changes do not affect the original design bases and do not alter the conclusions of the original design analysis or safety evaluation.

SAFETY EVALUATION

As noted in FSAR Sections 9.2.3 and 10.4.5, the condensers and circulating water system perform no nuclear safety related function. A failure mode evaluation of the proposed condenser outlet tube sheet and waterbox coatings has determined there is no potential for interaction with equipment or functions important to nuclear safety. Accordingly, the modification addressed by this engineering package is classified as non nuclear safety related.

Based on the above evaluation and information supplied in the design analysis, it has been demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Since there is no potential for interaction between the modification addressed by this engineering package and equipment of functions important to safety, previous safety analysis report evaluations related to safety remain unaffected.

- The possibility of an accident or malfunction different than those previously evaluated in the safety analysis report has not been created.

  No new accidents or malfunctions associated with the failure of the condenser outlet tube sheet and waterbox coatings have been created.

- The margin of safety as defined in the basis for any Technical Specification has not been reduced.

  Since there is no potential for interaction between the modification addressed by this engineering package and equipment or functions important to safety, the margin of safety as defined in any Technical Specification remains unaffected.

In conclusion, the modification proposed in this engineering package is acceptable from the standpoint of nuclear safety, does not involve an unreviewed safety question and does not require a change to any Technical Specifications. Accordingly, NRC approval prior to implementation is not required.

ST. LUCIE UNIT 1

Condenser Tubing Strain Gage Installation

ABSTRACT

As part of the investigation into titanium condenser tube hydriding (hydrogen embrittlement) which has been discovered at the St. Lucie Nuclear Plant, strain gage instrumentation will be used to measure actual tube strain following the unit's return to power operation after the present refueling outage. Data on actual tube strain levels during full power operation is required in order to develop "realistic" criteria for future tube plugging which may be required due to hydriding.

This design package provides the engineering necessary to install a 1 1/2 " diameter penetration into the condenser steam space to allow for routing of strain gage wiring. Also provided are guidelines for installing the strain gages and the lead wiring in the condenser and through the new penetration. Following testing, the strain gage lead wiring is to be cut, and the new penetration is to be capped and all joints are to be seal welded. During the next refueling outage, the wiring and "piping" conduit are to be removed from inside the condenser.

This design package is classified as "Non-Nuclear Safety Related" since it affects only nonseismic, Quality Group D piping and structures in Non-Nuclear Safety Related systems.

Based on the failure modes analysis and 10 CFR 50.59 review, this modification does not impact any safety related equipment and is not relied upon for any accident prevention or mitigation. Thus it does not constitute an unreviewed safety question. Since there are no unreviewed safety questions, and since no changes to technical specifications are involved, this PC/M may be implemented without prior NRC approval.

SAFETY EVALUAITON

The Unit 1 Condenser is a Non-Nuclear Safety Related component and as such is not required to function during any existing analyzed accident scenarios. Therefore, modifications to the condenser affects only Non-Nuclear Safety Related, Quality Group D equipment.

The added penetration will meet all design criteria of existing penetrations to insure that the condenser pressure boundary is maintained.

Postulated failures of the materials would have no impact on safe shutdown of the plant, or safety related systems. Any materials involved in this modification which could be postulated to become dislodged would be caught in the condenser hotwell pump screen. None of the materials are large enough to impact pump suction. Additionally, postulated failures of the condenser would have no impact on safe shutdown of the plant, or safety related systems. The condenser is not used to prevent postulated accidents, mitigate the consequences of such accidents, maintain safe shutdown conditions, or adequately store spent fuel.

The following statements demonstrate that an unreviewed safety question, as defined by 10 CFR 50.59, does not exist:.

* The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Failure of the condenser is not considered as an accident initiating event or considered in determining the probability of an accident. Also, since this design change does not alter or affect equipment used to mitigate accidents, the probability of malfunction of equipment important to safety remains unchanged.

* The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis has not been created.

  There is no new failure mode introduced by this change that has not been evaluated previously in the FSAR. Additionally, no failure modes analyzed by the FSAR are affected by this design.

* The margin of safety as defined in the basis for any Technical Specifications has not been reduced.

  This change has no effect on any existing Technical Specifications and does not require any changes to the Technical Specifications.

Since no unreviewed safety questions have been determined to exist, and since no revisions to the Technical Specifications are required, NRC approval is not required prior to implementation.

CONDENSATE RECIRCULATION TO CONDENSER SQUARE ROOT EXTRACTOR REPLACEMENT ST. LUCIE PLANT - UNIT NO. 1 REA-SLN-86-011

ABSTRACT

This Engineering Package covers the replacement of one (1) square root extractor. The presently installed square root extractor is no longer being manufactured and a suitable replacement is being provided for maintenance reasons. This Engineering Design Package is considered quality related since the replacement device is an integral part of the condensate recirculation system and a direct replacement for previously approved instrument. The instrumentation loop, of which this device is part of, is not used to mitigate incidents and accidents and, therefore, this PC/M is not considered to be safety related.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR 50.59. As indicated in Section 3.0 of this PCM, this PCM does not involve an unreviewed safety question, nor does it require a revision to the technical specification. Therefore, prior commission approval is not required for the implementation of this PCM.

SAFETY EVALUATION

The changing out of the Square Root Extractor in this PC/M does not involve an unreviewed safety question because:

This EP reflects no interference with the safety equipment in that they are not required for a safe reactor shut-down and could not be used to mitigate an accident. The square root extractors are non-safety related. This modification will have no effect on equipment performing any safety function. There is no possibility for the creation of an accident or malfunction. In the event of a total failure of this square-root extractor, it will have no effect upon any safety related equipement.

The probability of occurrence of the consequences of an accident or malfunction of equipment important to safety previously evaluated is neither increased nor occurs since this system is non-safety related. This modification will have no effect on equipment performing any safety function.

This system and/or component parts are not used in any accident scenario and there is no possibility for creating an accident or malfunction of a different type than any evaluated previously in the safety report. Its failure will have no impact on the plant safe shut-down.

It has no effect upon the margin of safety as defined in the basis for any technical specification since the replacement of the square root extractor

does not change the original design or operation and the proposed new extractor's are functionally identical to existing units. There are no changes to the plant technical specifications.

The foregoing constitutes, per 10CFR 50.59, the written safety evluation which provides the basis that this change does not involve an unreviewed safety question. Therefore, prior commission approval is not required for implementation of this PC/M.

ST. LUCIE UNIT NO 1

MFRV POSITION INDICATORS REMOVAL

REA-SLN-85-043-10

ABSTRACT

This engineering package covers the removal of two Main Feedwater Regulating Valve position indicators (ZI-9011,9021) from RTG Board 102 along with associated wiring, cable, and conduit. A steel plate will be fastened to the control board to cover the exposed area.

Since these indicators are operationally unreliable, the potential exists for incorrect interpretation of regulating valve position. Removal of the indicators will accomplish the resolution of a Human Factors Discrepancy (HED). No modifications to the valve control circuitry will be performed. Hence, routine valve operations will continue to be controlled from signals received automatically via the Feedwater Regulating System. Therefore, this modification will not have any adverse effect upon plant safety or operation.

There are neither any Technical Specification nor Regulatory Guide 1.97 requirements for these devices.

Since this design requires a modification to the RTG board, Quality Related requirements shall be imposed.

These changes were reviewed against the requirements of 10CFR50.59. As verified in the Safety Evaluation, this change neither requires a Technical Specification revision nor is it an unreviewed safety question. Therefore, prior NRC approval is not required.

SAFETY EVALUATION

This EP is classified as Quality Related because the components being removed, while performing a Non-Nuclear Safety Related function, are installed in the RTG Board where the potential exists for impacting Safety Related equipment through modification of the wiring in the RTG Board, the removal of equipment, and the installation of cover plates that could potentially have an effect on the seismic integrity of the RTG Board.

This design proposes to remove the Main Feedwater Regulating Valve (MFRV) position indicators currently installed in the RTG Board 102.

The indicators are unreliable and could provide misleading valve position indication. Removal of the indicators will not affect the operator's ability to determine feedwater flow or steam generator level. Ample instrumentation is available to monitor these parameters from the control room. In addition, indicating lights in the control room will remain to determine whether the subject flow control valves are fully open or fully closed.

The indicators being removed do not perform a Nuclear Safety Related function and are not included under any Technical Specification or Regulatory Guide 1.97 requirement.

Internal wiring changes are being performed in the RTG Board to disconnect the subject indicators and to remove (SIS) wiring. When required, only qualified wire jumpers will be installed inside the RTG Board. No conduit is being removed adjacent to, or in the vicinity of the RTG Board or control room.

The restoration of the RTG Board through appropriate cover plates to replace the removed indicators has been evaluated within this package. This evaluation concluded both that the seismic integrity of the RTG Board will be retained and that no missiles could be generated during a seismic event which could adversely impact Safety Related equipment.

ST LUCIE PLANT - UNIT NO 1
CONDENSATE POLISHER TIE-INS
REA-SLN-85-14

ABSTRACT

This Engineering Package (EP) is for the installation of the 24 inch Unit 1 tie-in piping and manual isolation valves required for the future connection of the Condensate Polisher System (CPS) to the St Lucie Unit 2 Condensate System. It also includes the installation of the 8 inch tie-in piping that will connect the CPS backwash pump suction to an existing Unit 2 Condensate Storage Tank non-safety class connection. This is for providing the capability of using the Unit 2 condensate for backwashing the condensate polisher.

This EP is classified quality related since it also involves modifications to the RTGB-102 which is seismically qualified and located in the Unit 1 Control Room. The modifications to the RTGB-102 involve the addition of the tie-in isolation valve position indicating lights and alarm for valve misalignment. The safety evaluation has determined that the EP does not constitute an unreviewed safety question and implementation of the EP does not require a change to the Plant Technical Specification. Therefore, prior NRC notification for implementing this EP is not required.

This EP has no impact on plant safety and operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulation, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This Engineering Package (EP) is for the installation of the 24 inch condensate Unit 1 tie-in piping required for the future connection of the Condensate Polisher System (CPS) to the Unit 2 Condensate System. It also includes the installation of an 8 inch connection to the condensate polisher backwash pump suction from an existing non-safety class connection to the Unit 2 condensate storage tank. This is for providing the capability of using, in the future, Unit 2 condensate for backwashing the condensate polishers. The portions of the Condensate System, Condensate Storage Tank piping and the CPS that this modification will be implementing do not perform any safety function or interact with safety related equipment; however, since this EP also involves modifications to the RTGB, which is seismically qualified, for the addition of an annunciator and indicating lights for the condensate polisher isolation valves, it is classified quality related.

SAFETY EVALUATION (CONTINUED)

The new annunciator and indicating lights that will be added to the RTGB have been designed and will be installed to the same requirements as existing annunciators and indicating lights in the RTGB. This addition of components to the RTGB has been reviewed and considered acceptable and in compliance with the seismic requirements applicable to the RTGB.

Based on the above description, the modification included in this Engineering Package (EP) is considered to be quality related. This EP does not involve an unreviewed safety question, and the following are bases for this justification:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The portions of the Condensate System, where this modification will be implemented, and the CPS are not used in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and will have no effect on equipment vital to plant safety. The addition of the new annunciator and indicating lights to the RTGB has been reviewed and considered to be acceptable and in compliance with the seismic requirements applicable to the RTGB.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provided the bases that this change does not involve an unreviewed safety question and prior Commisision approval for the implementation of this PCM is not required.

ST LUCIE UNIT NO 1
FIRE DETECTOR MODIFICATIONS
REA-SLN-86-63-10

ABSTRACT

This Engineering Package covers the modifications to the fire detection system which is part of the Fire Protection System.

This Engineering Package will provide the engineering and design details required to implement the replacement of the existing ionization smoke detectors. The existing detectors are divided into two (2) groups: The originals (installed eleven (11) years ago) which are obsolete; and their replacements (installed as the originals failed) which are no longer manufactured. To ensure the reliability of the fire detection system, new state of the art ionization smoke detectors will be installed.

The Fire Detection System is non-safety related, but is provided in areas that contain or present a fire exposure to equipment essential to safe plant shutdown. Therefore, this Engineering Package is classified as Quality Related.

The safety evaluation has determined that this modification does not involve an unreviewed safety question and implementation of this PCM does not require a change to Plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PCM is not required.

This EP has no impact on the plant safety and operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This Engineering Package provides the engineering and design details required to implement the replacement of the existing ionization smoke detectors with new detectors and new wiring bases. The existing detectors are either obsolete or no longer manufactured.

The implementation of this Engineering Package ensures the availability of the individual detectors to detect a fire.

SAFETY EVALUATION (CONTINUED)

Fire detection systems are provided in areas that contain or present a fire exposure to equipment essential to safe plant shutdown. Therefore, this Engineering Package has been classified as Quality Related.

Based on the preceding, this EP does not involve an unreviewed safety question and the following are the bases for this justification:

(i) The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased, since the replacement of the ionization smoke detectors enhances the operability of the equipment. The replacement of the obsolete detectors with new detectors is on a one to one basis, with the new detectors having the same characteristics as the existing detectors. The possible failure of the detectors will not prevent safety related equipment from performing their intended functions. The detectors are not required during an accident condition.

(ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type than any previously evaluated. The detectors are not required during an accident condition nor will they prevent safety related equipment from performing their functions. The existing detectors are being replaced on a one to one basis. This modification does not affect any safety related equipment.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The functions of the Fire Detection System that are controlled by the applicable Technical Specifications are maintained by this change.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST LUCIE PLANT - UNIT 1
ERDADS/SAS UPGRADE
REA-NONE

ABSTRACT

This Engineering Package provides for modifications to the computer room in preparation for implementing an upgrade to the Emergency Response Data Acquisition and Display System, which is also known as the Safety Assessment SYstem (ERDADS/SAS), under PCM 076-187 Supplement 1 and PCM 077-287. Included in this work are the connection of the computer room to the adjoining office, relocation of computer room and office doors, installation of a false floor in the office, upgrade of lighting and convenience oulets, and installation of conduit and cables for the computer control terminals, the data loading terminal, CRT #12 console, and disk drives.

This Engineering Package is classified as quality related due to the cable and conduit which are being installed to support SAS quality related components. Implementation of this PCM does not involve an unreviewed safety question or change to the Plant Technical Specifications. Therefore, it can be implemented without prior Commission approval.

Implementation of this EP will not affect the safety of operation of the plant.

SUPPLEMENT 1

In addition to modifying the computer room, this Engineering Package provides for an upgrade to the ERDADS/SAS hardware and software including the Safety Parameter Display System (SPDS), in the St Lucie - Unit 1 control room, computer room and technical support center. It will improve the performance and display capabilities of the existing system and will include new display CRTs and keyboards, new color hardcopiers, additional printers, a data loading terminal, additional memory and new internal computer switching and communications components.

This EP remains classified as quality related since the function of the ERDADS/SAS system, which is to assist the operators in evaluating the safety status of the plant, has not changed. The original safety evaluation has not been affected. Therefore, implementation of this EP does not involve an unreviewed safety question or a change to the Plant Technical Specifications. It may be implemented without prior Commission approval.

Implementation of this EP will not affect the safety or operation of the plant.

SUPPLEMENT 2

Supplement 2 to this Engineering Package modifies the design to replace the CRTs, video generators, and supporting components which were originally specified in Supplement 1 due to hardware compatibility problems. The overall design remains the same.

This EP remains classified as quality related since the function of the ERDADS/SAS system, which is to assist the operators in evaluating the safety status of the plant, has not changed. The original safety evaluation has not been affected. Therefore, implementation of this EP does not involve an unreviewed safety question or a change to the Plant Technical Specifications. It may be implemented without prior Commission approval.

Implementation of this EP will not affect the safety or operation of the plant.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

SAFETY EVALUATION (continued)

The modifications included in this Engineering Package do not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated are not increased since the existing input isolation of the ERDADS/SAS equipment will not be modified and will maintain the same level of protection for safety-related equipment.

ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no new safety-related functions or interfaces with safety-related systems are created by this EP.

iii) This modification does not change the margin of safety as defined in the basis for any Technical Specification, since no equipment installed or modified by this EP affects any parameter referenced in the Technical Specifications.

This EP modifies equipment which is not nuclear safety-related. However, since the ERDADS/SAS system assists control room personnel in evaluating the safety status of the plant, this EP is classified as quality related.

The Human Factors Engineering evaluation of the SPDS portion of the ERDADS system found seventy-four (74) HEDs. All four (4) Priority 1 HEDs have been corrected. Therefore, the HEDs found through this Human Factors Engineering review do not affect plant safety.

This EP has no effect on cables or components necessary for safe shutdown of the plant. Changes to equipment and structures involving 10CFR50 Appendix "R" fire protection requirements and changes to equipment on the Essential Equipment List have been addressed. (See Attachment 7.1). Thus, the proposed design is in compliance with applicable requirements for fire protection.

The implementation of this change does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

This EP has no effect on cables or components necessary for safe shutdown of the plant. Changes to equipment and structures involving 10CFR50 Appendix "R" fire protection requirements and changes to equipment on the Essential Equipment List have been addressed. (See Attachment 7.1). Thus, the proposed design is in compliance with applicable requirements for fire protection.

The implementation of this change does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST LUCIE PLANT - UNIT 1
REPLACEMENT OF FISCHER AND PORTER CONTROLLERS
REA-SLN-86-91-10

ABSTRACT

This Engineering Package (EP) covers the replacement of the now obsolete Fischer & Porter controllers with the currently manufactured and functionally equivalent Fischer & Porter controllers. The controllers are used to maintain the level and pressure parameters in the pressurizer within the required limits during the normal plant operation.

These controllers perform Non-Nuclear Safety Related functions. However, being located on the main control board, they are expected to maintain their structural integrity during the design basis seismic event. The controllers are classified Quality Related.

The safety evaluation (Section 3.0) indicates that this Engineering Package does not involve an unreviewed safety question, and does not require a change in the Plant Technical Specifications. Therefore, NRC approval for these modifications, prior to their implementation, is not required.

This EP has no impact on plant safety or operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfuntion of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report are not increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created by this modification because the function of the controllers has not been altered by this modification.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since the new controllers perform non-nuclear safety related functions and are not included in the bases of any technical specification.

The new controllers replace the obsolete controllers on Class 1E main control board, therefore, this EP is classified Quality Related.

The implementation of this EP does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question. Therefore, the PCM may be implemented without prior Commission approval.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST. LUCIE UNIT 1
TURBINE GENERATOR SEAL OIL SYSTEM ENHANCEMENTS
(REA: SLN-86-092-10)

ABSTRACT

This Engineering Package covers modifications to the Turbine Generator Seal Oil System as recommended in Westinghouse Operations and Maintenance Memo #051 (Reference 6.3). This modification provides for the installation of a "drip leg" in the air side seal oil pump suction line and an additional vent line between the existing vent line and the hydrogen side drain regulator tank. These system enhancements should minimize oil intrusion into the generator housing, and decrease the amount of dirt and contamination that would lead to damage/wear to system components.

The Turbine Generator Seal Oil System performs no safety related functions nor does it interact with safety related equipment. Therefore, this modification is classified as non-nuclear safety related.

Based on a failure mode evaluation and a 10 CFR 50.59 review, this modification does not involve an unreviewed safety question nor require changes to the Technical Specifications. Therefore, prior NRC approval is not required for implementation of this modification. This modification has no adverse effect on plant safety or operability.

SAFETY EVALUATION

This Engineering Package covers the modifications to the Turbine Generator Seal Oil System. A "drip leg" will be installed in the air side seal oil pump suction line. Also, an additional vent line will be installed between the existing vent line and the hydrogen side drain regulator tank. This modification is classified as non-nuclear safety related, since the Seal Oil System performs no safety related function and does not interact with safety related equipment, components, or functions.

Based on the above and information supplied in the design analysis, it can be demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Due to the location of the "drip leg" and the vent line, their failure would not cause interaction with any safety related equipment. Also, the turbine generator seal oil system is not considered by the FSAR in determining the probability of accidents, possible types of accidents, or in the evaluation of consequences of accidents. Therefore, it can be concluded that the probability of occurrence of accidents previously addressed in the FSAR remains unchanged.

- The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

  The components involved in this modification do not perform safety related functions. The operability of the turbine generator seal oil system has not been adversely affected by the modification. Also, the location of the "drip leg" and vent line eliminates the possibility of interaction with safety related equipment. Therefore, the possibility of an accident of a different type has not been created.

- The margin of safety as defined in the basis for any technical specification has not been reduced.

Since the components involved in this modification are not directly included in the bases of any technical specification, the margin of safety has not been reduced.

10 CFR 50.59 allows changes to a facility as described in the FSAR if an unreviewed safety question does not exist and if a change to the technical specifications is not required. As shown in the preceeding sections, the change proposed by this design package does not involve an unreviewed safety question because each concern posed by 10 CFR 50.59 that pertains to an unreviewed safety question can be positively answered. Also, no change to the Technical Specifications is required based on the above evaluation. Therefore, prior NRC approval is not required for implementation of this modification.

REMOTE REACTOR VESSEL LEVEL INDICATOR
ST LUCIE PLANT - UNIT NO 1
REA-SLN-87-006

ABSTRACT

This Engineering Package (EP) is to install a remote level indicator for the reactor vessel. This indicator will provide reliable level indication during refueling.

The modifications considered in this EP are on the Reactor Coolant System. The connections are designated as Nuclear Safety Related and seismically qualified since they are within the Reactor Coolant Pressure Boundary, and therefore, this modification is classified as Safety Related. The instrument side of the system downstream of the piping isolation valve is designated as non-safety, seismic design. Two transmitters (one wide range, one narrow range) and associated cables will be installed. Indication will be added to the Control Room to allow monitoring of refueling water level. The safety evaluation has shown that this EP does not constitute an unreviewed safety question and prior NRC approval is not required for implementation. The implementation of this EP does not require a change to the Technical Specification and does not reduce the margin of safety for any Technical Specification.

The implementation of the EP will have no impact on plant safety or operation.

Supplement No 1

The purpose of this supplement is to remove all hold points associated with this EP. The reactor coolant piping supports and the conduit supports within the containment area have been evaluated, so the hold points are no longer necessary.

The implementation of this supplement will have no impact on plant safety or operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulation, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

The modifications included in this Engineering Package are for the Reactor Vessel water level indicator installation involving piping, tubing, valves and orifices and differential pressure transmitters, all connected between the RCS and the Pressurizer.

Based on the above description, the modification included in this Engineering Package (EP) is considered to be safety related. This EP does not involve an unreviewed safety question, and the following are bases for this justification:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased since this modification provides a means whereby an accurate Reactor Vessel water level can be readily determined during refueling. During power operation this system is isolated from the RCS. The portions of this modification within the normal RCS pressure boundary have been designed to the original requirements of the RCS pressure boundary.

ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type than any previously evaluated because the modification provides double isolation valving which will isolate the system from the RCS during power operation.

iii) This modification does not reduce the margin of safety as defined in the basis for any Technical Specification because it neither changes the design parameter of the RCS nor does it change the RCS design flow or functional requirements.

The implementation of this PCM does not require a change to the plant Technical Specification.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question or a change to the Technical Specifications, and prior commission approval for the implementation of this PCM is not required.

ST. LUCIE UNIT 1
CHARGING PUMP BLOCK MATERIAL CHANGE
REA SLN-85-11-9

ABSTRACT

This design package covers the replacement of the current charging pump block material of 316 stainless steel (ASTM-A-182 F316) with 17-4 PH stainless steel (ASME-SA-705 Gr. 630 1150 HT). The 17-4 PH material has a tested fatigue strength approximately twice that of 304 or 316 stainless steel. Field testing of charging pump systems indicate that strong pressure pulsations exist at times in the system. These pulsations are in part responsible for the fatigue failures of the charging pump blocks. Reduction of pressure pulsations is a current concern. The increase in fatigue strength of the new material should result in a substantial improvement in block life. Based on a failure mode analysis and 10CFR50.59 review, the changes proposed by this engineering package are acceptable from the standpoint of Nuclear Safety. This modification does not involve an unreviewed safety question and a Technical Specification change is not required, therefore, prior NRC approval is not required for implementation of this modification. The function of the charging pumps is not altered by this modification. This engineering package is classified as Nuclear Safety Related.

SAFETY EVALUATION

This modification consists of replacement of the existing 316 stainless steel charging pump blocks with 17-4 PH 1150 HT stainless steel. This modification does not affect the design function of the charging pumps and does not introduce any new active components to the system.

The new material is stronger than the existing material and should provide a substantially longer service life for the block. Since the system and components modified by this engineering package are ASME Section III, Class II, this package is classified as Nuclear Safety Related.

The following constitutes an evaluation to determine if the implementation of this engineering package will result in an unreviewed safety question as defined by 10CFR50.59:

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis is not increased since no new active components are being added, and the failure modes of existing components are not being altered. Accident probabilities and consequences are not affected by this modification.

- The probability of an accident or malfunction of a different type than previously evaluated in the FSAR has not been created. Since the system design bases as described in FSAR Sections 9.3.4.3.2 (f) are not affected by this modification, no new accidents are made possible.

- The margin of safety as defined in the basis for any technical specification has not been reduced since no system design parameters are being altered. The technical specifications have been reviewed and it has been determined that no changes are required.

In conclusion, the change proposed in this design package is acceptable from the standpoint of nuclear safety, does not involve an unreviewed safety question, and does not require any change to technical specifications. Therefore, prior NRC approval is not required for implementation of the modifications.

ST LUCIE UNIT NO 1
REPLACEMENT OF SAFETY RELATED BATTERIES 1A and 1B
(REA-SLN-87-008-11)

ABSTRACT

This Engineering Package covers the modifications to the Safety Related Station Batteries 1A and 1B which are part of the 125V DC Distribution System.

This Engineering Package will provide the engineering and design details required to implement the replacement of the existing batteries with new batteries. The existing batteries are showing signs of degradation (the battery acid is contacting the copper posts). The new batteries will also have an increased spare design margin (capacity) of 3% over the existing batteries, which were installed in the early 80s, for future load growth capability.

The station batteries, which are part of the 125V DC system, are classified as Class 1E, are seismically qualified and perform a safety related function. This EP will be classified as Safety Related.

This EP does not constitute an unreviewed safety question since the modifications described above were reviewed in accordance with 10CFR50.59. and were determined to have no adverse impact on plant operations or safety related equipment.

The implementation of this PCM does not require a change to the Plant Technical Specifications.

This change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This Engineering Package provides the engineering and design details required to implement the replacement of the existing batteries with new batteries. The existing batteries are showing signs of degradation which could reduce the capacity of the battery cells.

The implementation of this Engineering Package increases the availability of the batteries, upon loss of the AC power system, to provide power sufficient to supply the DC loads until the battery chargers are loaded onto the diesel generators. The 125V DC systems, which include the station batteries, are safety related and complete separation and independence are maintained between equipment and circuits, including raceway. A single failure at any point in either system will not disable both systems.

The station batteries which are being replaced perform a safety related function within the 125V DC distribution system and are designed for operation under conditions that could be imposed by a Design Basis Accident (DBA). This Engineering Package has been classified as Safety Related.

Based on the preceding, the following conclusions can be made.

(i) The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased, since the replacement of the station batteries enhances the operability of the equipment. The addition of new batteries ensures that the batteries will supply the minimum DC power requirements to safely shutdown the plant and/or mitigate the consequences of a DBA.

(ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type than any previously evaluated. This modification affects accident mitigating equipment to enhance their operation. The DC system voltage remains the same but the new batteries provide an increased spare design margin (capacity) for future load growth. There is no introduction of any new failure mode for the equipment.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The safety function that is controlled by the various applicable Technical Specifications is maintained by this change. The proposed design ensures that the batteries will function as assumed during an accident. Thus the margin of safety provided by the Technical Specifications is preserved.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ABSTRACT

In the course of preparing the Fire Protection Appendix of the Unit 1 FSAR, a concern was raised as to whether certain masonry block walls assumed to be 3 hour fire barriers are actually grout filled. A safety evaluation was performed (Reference 6.5) which established that, if these walls are in fact not filled with grout and therefore not providing the full 3 hours of fire protection, the plant still maintains its ability to achieve safe shutdown. This safety evaluation recommended that an inspection of these walls be performed to establish their as-built condition. Such an inspection was performed and concluded that the walls are not fully grouted.

This Engineering Package (EP) provides the details/requirements for grouting the voids in block walls 79, 84, 84A, 85, 92A, 114, 115, and 115A. This grouting will be performed in two phases. A HOLD POINT is placed on construction activities at the completion of Phase I work. Phase II construction activities will resume following engineering approval of the Phase II grouting material.

This modification does not involve an unreviewed safety question, has no effect on plant safety and operation, and does not involve a change to any plant Technical Specification. Upon completion of this modification, the action in Technical Specification 3/4.7.12 will no longer be required for the walls modified. This EP is classified Quality Related since all of the walls involved are required per 10 CFR 50 Appendix R to be fire barriers.

SAFETY EVALUATION

Safety Analysis

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

When a concern was raised that the walls modified by this EP might not be fully grouted, a report (Reference 6.5) was written to evaluate the safety implications if the walls were found to be not fully grouted. This report demonstrated that, if an ungrouted condition was confirmed, no unreviewed safety questions exists and continued operation of the plant is justified. This EP provides the details/requirements for grouting the walls so that they are in conformance with the design bases established in Subsection 3.11.2 of the St Lucie Unit 1 FSAR Appendix 9.5A; consequently, this modification cannot give rise to an unreviewed safety question.

Although the walls do not perform a safety-related function, this EP is classified Quality Related, since all of the walls are required per 10 CFR 50 Appendix R to be fire barriers.

Based on the above, the following provides the justification that an unreviewed safety question does not exist:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased. Since the walls located in the vicinity of safety-related equipment maintain their seismic qualification, no accidents due to structural failure are postulated. The only other type of accident potentially associated with the walls affected by this modification involves damage that could occur if the walls fail to provide three hours of fire protection. The JCO discussed above, however, demonstrated that no single fire event could impair the plant's ability to achieve safe shutdown. Consequently, there are no accidents or malfunctions of equipment important to safety previously evaluated whose probability of occurrence or consequences are increased by this modification.

Construction activities will stop when Phase I is completed. Phase II construction will continue after Phase II materials have been reviewed and approved with respect to their density and their structural, radiation resistance, and thermal resistance properties, and the use of these materials has been shown not to degrade the seismic qualification of Walls 85 and 114. This item is identified as a HOLD POINT and must be resolved prior to the implementation of Phase II. The safety evaluation will be revised upon resolution of this item.

(ii) There is no possibility for an accident or malfunction of a different type than any evaluated previously since the modification provides the walls with a three hour fire rating while the design ensures that the structural integrity of the seismically designed walls is maintained.

(iii) This modification does not change the margin of safety as defined in the bases for any Technical Specification. The basis for Technical Specification 3/4.7.12 indicates that fire barriers ensure that fire damage will be limited and the possibility of a single fire event involving more than one fire area prior to detection and extinguishment will be minimized. The referenced JCO indicated that the current situation, in combination with compensatory measures, does not violate this basis. When the walls are fully grouted, the barriers will be fully operational, eliminating the need for the said compensatory measures.

The implementation of this PCM does not require a change to plant technical specifications.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

TITLE CEA MG SETS LOCKOUT RELAY

DESCRIPTION OF CHANGE/ABSTRACT: This change modifies drawings (see drawing list) to show lockout relay 52Y contact (16-17) as normally closed per vendor manual representation. No physical change is required, only correction of drawing. No unreviewed safety question or change to technical specification is involved.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes ____ No X A change to the plant as described in the FSAR?

Yes ____ No X A change to procedures as described in the FSAR?

Yes ____ No X A test or experiment not described in the FSAR?

Yes ____ No X A change to the plant technical specifications?

EFFECT OF CHANGE

Yes ____ No X Will the probability of an accident previously evaluated in the FSAR be increased?

Yes ____ No X Will the consequences of an accident previously evaluated in the FSAR be increased?

Yes ____ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?

Yes ____ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

Yes ____ No X Will the margin of safety as defined in the bases to any technical specification be reduced?

ST LUCIE PLANT - UNIT NO 1
SAFETY INJECTION TANK AND CONTAINMENT FAN COOLER INSTRUMENTATION UPGRADE
REA-SLN-86-076-11, -13, -21, -23

ABSTRACT

This Engineering Package addresses level, temperature, and flow instrumentation upgrade for the Safety Injection Tank (SIT), Component Cooling Water and Containment Fan Coolers.

The Safety Injection Tanks are part of the Safety Injection System which automatically discharges borated water into the Reactor Coolant System on depressurization of RCS as a result of a Loss of Coolant Accident (LOCA). The level instrumentation being upgraded measures the Safety Injection Tank water level and provides indication at the RTGB.

The Containment Fan Coolers are part of the Containment Cooling System which provides the means of Containment heat removal during normal operations and in the event of a LOCA. The flow instrumentation being upgraded detects low Component Cooling Water flow through the Containment Fan Coolers, providing local indication and remote annunciation. The temperature detecting elements (thermocouples) at the inlet and outlet of the Containment Fan Coolers used to measure the duct air temperature are also being upgraded.

These instruments currently are designated as Non-Nuclear Safety Related. This effort will upgrade selected instrumentation, associated electrical circuit loops and structural support to Nuclear Safety Related meeting the requirements of USNRC Regulatory Guide 1.97, Rev 3, Category 2, Type D Variable.

This U S Nuclear Regulatory Commission requirement is defined as those instruments that remain functional during all accident conditions and provide indication and records for many variables required to follow the course of the accident. Specifically Type D variables are defined as those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. Category 2 provides for equipment qualification which is less stringent in that it does not include seismic qualification, redundancy or continuous displays and requires only a high-reliability power source.

Based on the usage of these instruments to monitor safety related equipment, this EP is classified as Nuclear Safety Related.

The safety evaluation of this package has shown that the implementation of this PCM does not constitute an unreviewed safety question and prior commission approval for its implementation is not required.

This EP has no impact on plant safety and operation or Plant Technical Specifications.

SUPPLEMENT 1

This engineering package revision revises control wiring diagrams and cable splice details dealing with the revised Conax thermocouple electric conductor seal assembly. Environmental Qualification Documentation Package 8770-A-451-6.0, Continental Wire and Cable, has been updated to include references to model CC-2200 (XLPE) B/M D5-1.

The original safety evaluation has not been affected as a result of this supplement.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification because existing equipment availability, redundance, capacity, or function required to mitigate the effects of an accident are not affected.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created by this modification because replacing monitoring instrumentation with similar replacements having better environmental qualifications does not create changes which could postulate a new accident or malfunction.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since this modification installs qualified thermocouples and flow switches which will enhance the monitoring of the Containment Heat Removal System. Furthermore, this new equipment is seismically and environmentally qualified to withstand the normal and accident conditions anticipated in the areas that they are installed.

This modification is for the upgrade of the Safety Injection System, Component Cooling Water System and Containment Cooling System instrumentation in order to meet the requirements of USNRC Regulatory Guide 1.97, Rev 3, Category 2, Type D Variable. This modification upgrade will provide a more reliable and qualified instrumentation loop to detect and monitor Containment Heat Removal System operation. Hence, this EP is considered Nuclear Safety Related. Since this modification replaces existing monitoring instrumentation with qualified devices and involves no other modifications to safety related equipment, the degree of protection provided to nuclear safety related equipment is unchanged. The probability of

malfunction of equipment important to safety previously evaluated in the FSAR remains unchanged. The consequences of malfunction of equipment important to safety previously evaluated in the FSAR are unchanged. The possibility of malfunctions of a different type than those analyzed in the FSAR is not created.

The implementation of this EP does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST LUCIE UNIT 1
480V SWITCHGEAR 1A2 & 1B2
TRANSFORMER REPLACEMENT
(REA SLN-86-007-10)

ABSTRACT

Due to environmental concerns attendant to polychlorinated biphenyl (PCB) cooling/insulating liquids, all transformers filled with PCB are being eliminated from FP&L's system. The station service transformers for 480 volt switchgear 1A2 and 1B2 are filled with PCB cooling/insulating oil. Each transformer contains 370 gallons and 254 gallons respectively of PCB liquid. This Engineering Package provides for the replacement of the existing PCB filled station service transformers with equivalent transformers of dry type construction and for the removal of the concrete curbs surrounding the transformers. The curbs are no longer required since their function was to retain leakage of cooling/insulating liquid which is no longer present in the replacement transformers.

Station service transformers 1A2 and 1B2 perform nuclear safety related functions. Because of their importance in Class 1E service applications the replacement transformers are classified as safety related in this Engineering Package.

Transformers (1A2 & 1B2) are located in the Switchgear Room at Elevation 43'0" of the Reactor Auxiliary Building.

Results of the safety evaluation conclude that modifications presented by this Engineering Package do not constitute an unreviewed safety question, do not require any changes to the Plant Technical Specifications and do not require prior NRC approval for the implementation of this PC/M.

The implementation of this PC/M will not have an adverse impact on plant safety or operations.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This Engineering Package addressed the replacement of PCB liquid filled 480V station service transformers 1A2 & 1B2 located on elevation 43' in the Reactor Auxiliary Building of Unit 1. The replacement transformers will be furnished dimensionally compatible and equivalent in electrical characteristics with the existing transformers.

The physical characteristics of the replacement transformers are different because they are dry type.

The new transformers are safety related because of their importance to essential plant operations. These new transformers perform the same function as the existing transformers 1A2 and 1B2. The replacement transformers have been seismically and environmentally qualified (References 6.18 and 6.19) and will be seismically mounted. The existing seismic qualification of switchgear lineups, 1A2 and 1B2 will not be affected by the replacement of the 1A2 and 1B2 PCB filled transformers with the new dry type transformers.

The curbs do not perform any safety function. They were designed to contain cooling/insulating liquid which will no longer be used; therefore these curbs are no longer required.

Based on the preceding, the following conclusions can be made:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because the existing transformers are being replaced on a one-for-one basis by transformers that are essentially equivalent in function, capacity and qualifications. The curbs did not perform a safety related function. Their removal will not have any safety related implications.

(ii) This modification does not change the operation of the 480V safety related station service transformers and switchgear. Therefore, there is no possibility that an accident or malfunction of a different type than any evaluated in the FSAR may be created.

(iii) The replacement station service transformers are essentially equivalent in purpose and capability to the existing transformers. Therefore, this modification does not reduce the margin of safety as defined in the bases for any technical specification.

The implementation of this PC/M does not require a change to the Plant Technical Specifications.

The foregoing constitutes per 10CFR50.59(b) the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PC/M is not required.

ST LUCIE UNIT 1
480V SWITCHGEAR 1A3 & 1B3
TRANSFORMER REPLACEMENT
(REA SLN-86-007-10)

ABSTRACT

Due to environmental concerns attendant to polychlorinated biphenyl (PCB) cooling/insulating liquids, all transformers filled with PCB are being eliminated from FP&L's system. The pressurizer heater transformers for 480 volt switchgear 1A3 and 1B3 are filled with PCB cooling/insulating oil. Each transformer contains 208 gallons of PCB liquid. This Engineering Package provides for the replacement of the existing PCB filled pressurizer heater transformers with equivalent transformers of dry type construction, and for the removal of the concrete curbs surrounding the transformers. The curbs are no longer required since their function was to retain leakage of cooling/-insulating liquid, which is no longer present.

Pressurizer heaters transformers 1A3 and 1B3 perform non-nuclear safety related functions. Because of their importance in plant operations and because they are fed from Safety Related buses, 4160V 1A3 and 1B3, the replacement transformers are classified as Quality Related in this Engineering Package.

Transformers (1A3 & 1B3) are located on Elevation 43'0" of the Reactor Auxiliary Building.

Results of the safety evaluation conclude that modifications presented by this Engineering Package do not constitute an unreviewed safety question, do not require any changes to the Plant Technical Specifications and do not require prior NRC approval for the implementation of this PC/M.

The implementation of this PC/M will not have an adverse impact on plant safety or operations.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This Engineering Package addressed the replacement of PCB liquid filled 480V pressurizer heater transformers 1A3 & 1B3 located on elevation 43' in the Reactor Auxiliary Building of Unit 1. The transformers supply power to the pressurizer heaters and are located in an area of the plant containing safety-related equipment. The 480V Pressurizer Heater Transformers 1A3 and 1B3 do not perform any nuclear safety related functions, however, because of their importance to normal plant operations and because transformers 1A3 and 1B3 are fed by safety related 4160V Buses 1A3 and 1B3, the replacement transformers are classified as Quality Related in this Engineering Package. The 'dry type' replacement transformers will be furnished dimensionally compatible and equivalent in electrical characteristics with the existing transformers.

The physical characteristics of the replacement transformers are different because they are dry type.

These new transformers perform the same function as the existing transformers 1A3 and 1B3. The replacement transformers will be seismically mounted.

The curbs do not perform any safety function. They were designed to contain cooling/insulating liquid which will no longer be used; therefore these curbs are no longer required.

Based on the preceding, the following conclusions can be made:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because the existing transformers are being replaced on a one-for-one basis by transformers that are equivalent in function, capacity and electrical characteristics. The curbs did not perform a safety related function, their removal will not have any safety related implication.

(ii) This modification does not change the operation of the 480V non-safety related pressurizer heater transformers and switchgear. Therefore, there is no possibility that an accident or malfunction of a different type than any evaluated in the FSAR may be created.

(iii) The replacement pressurizer heater transformers are equivalent in purpose and function to the existing transformers and perform no safety related functions. Therefore, this modification does not reduce the margin of safety as defined in the bases for any technical specification.

The implementation of this PC/M does not require a change to the plant Technical Specifications.

The foregoing constitutes per 10CFR50.59(b) the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PC/M is not required.

TITLE 480V PCB FILLED TRANSFORMER REPLACEMENT 1A1 AND 1B1

DESCRIPTION OF CHANGE/ABSTRACT: EXISTING PCB FILLED 1500 KVA STATION SERVICE TRANSFORMERS 1A1 and 1B1 ARE BEING REPLACED WITH NON-PCB FILLED SILICONE IMPREGNATED DRY TYPE TRANSFORMERS TO SATISFY ENVIRONMENTAL CONCERNS REGARDING PCB'S.

Implementation of this DEEP does not constitute an unreviewed safety question nor affect Plant Technical Specifications. NRC approval is not required prior to implementation. This DEEP has no impact on plant safety or operation.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes	No	
X		A change to the plant as described in the FSAR?
	X	A change to procedures as described in the FSAR?
	X	A test or experiment not described in the FSAR?
	X	A change to the plant technical specifications?

EFFECT OF CHANGE

Yes	No	
	X	Will the probability of an accident previously evaluated in the FSAR be increased?
	X	Will the consequences of an accident previously evaluated in the FSAR be increased?
	X	May the possibility of an accident which is different than any already evaluated in the FSAR be created?
	X	Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
	X	Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
	X	May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
	X	Will the margin of safety as defined in the bases to any technical specification be reduced?

SIT SAMPLE VALVE AS BUILDING MODIFICATION

DESCRIPTION OF CHANGE/ABSTRACT: Revise CWD 8770-B-327 Sh 322 to show as-building state of SIT Sampling Isolation Valve, I-FCV-03-1F wiring as follows: W&B conductors of H-SB to be shown connected to TB639: 9 & 10 respectively instead of TB635: 7 & 8 as per attached marked drawing. This is a drawing change only. It does not affect system function or qualification. It does not require a Tech. Spec. change and it does not involve an unreviewed safety question.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes ____ No _x_ A change to the plant as described in the FSAR?

Yes ____ No _x_ A change to procedures as described in the FSAR?

Yes ____ No _x_ A test or experiment not described in the FSAR?

Yes ____ No _x_ A change to the plant technical specifications?

EFFECT OF CHANGE

Yes ____ No _x_ Will the probability of an accident previously evaluated in the FSAR be increased?

Yes ____ No _x_ Will the consequences of an accident previously evaluated in the FSAR be increased?

Yes ____ No _x_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

Yes ____ No _x_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No _x_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No _x_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

Yes ____ No _x_ Will the margin of safety as defined in the bases to any technical specification be reduced?

ST LUCIE UNIT 1
CONTROL ELEMENT DRIVE SYSTEM & COIL POWER PROGRAMMER PART LENGTH REMOVAL
REA-SLN-86-85

ABSTRACT

This Engineering Package (EP) provides for the removal of unused equipment in the Control Element Drive System (CEDS). The unused equipment was previously employed for power shaping with part-length control elements. The part-length control elements have been removed from the reactor. The electronic components associated with these elements (power supplies, coil power programmer modules, power shaping group modules, displays, etc.) will be removed and maintained as spares.

The Control Element Drive System is not a Nuclear Safety Related System (see FSAR Section 7.1). However, since the CEDS is used to control reactor operation, and since modifications to the RTG Board must be reviewed for their effect on RTGB seismic qualification, this Engineering Package is classified as Quality Related. Implementation of this PCM does not involve an unreviewed safety question or a change to the Plant Technical Specifications. Therefore, prior commission approval for the implementation of this PCM is not required.

Implementation of this PCM does not affect the safety or operation of the plant.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased: or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this Engineering Package consist of the removal of non-functioning equipment, not classified as safety-related, which has no effect on operating plant systems. The modifications do not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated are not increased since no modification is made to any safety related component, system, or function.

ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no new safety-related functions or interfaces with safety-related systems are created by this EP.

iii) This modification does not change the margin of safety as defined in the basis for any Technical Specification, since no equipment removed or modified by this EP affects any parameter referenced in the Technical Specifications.

This EP does not modify equipment which is nuclear safety-related. However, since the Control Element Drive System is used to control reactor operation and since modifications to the RTG Board must be reviewed for their effect on RTGB seismic qualification, this EP is classified as Quality Related.

This EP has no effect on cables or components necessary for safe shutdown of the plant, or on equipment on the Essential Equipment List. Changes to equipment and structures involving 10CFR50 Appendix "R" fire protection requirements have been addressed. Thus, the proposed design is in compliance with applicable requirements for fire protection.

The implementation of this change does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ST. LUCIE UNIT 1
MOISTURE SEPARATOR REHEATER SHELL REPAIR
(REA-SLN-87-031)

Abstract

This design package provides the necessary engineering for adding erosion protection features to the internal surfaces of the Moisture Separator Reheater (MSR) shells.

The effort involves the installation of chromium-molybdenum liner plates to the shells in the area(s) being affected by wet steam impingement/erosion.

Based on the failure modes analysis and 10CFR 50.59 review, this modification does not impact any safety related equipment and is not relied upon for any accident prevention or mitigation. Thus it does not constitute an unreviewed safety question and is correctly classified as non-nuclear safety related. Implementation of this modification, therefore, does not require prior NRC approval. There are no technical specifications affected and the modifications will not affect plant safety or operation.

SAFETY EVALUATION

This design package provides the necessary engineering for adding erosion protection features to the internal surfaces of the Moisture Separator Reheater (MSR) shells.

The effort involves the installation of chromium-molybdenum liner plates to the shells in the area(s) being affected by wet steam impingement/erosion.

This modification is classified as non-nuclear safety related, since the MSRs perform no safety related function and do not interact with safety related equipment, components, or functions.

Based on the above and information supplied in the design analysis, it can be demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  The MSRs are not used in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and have no effect on equipment vital to plant safety.

- The possibility of an accident or malfunction of a different type than any evaluated previously in the safety report has not been created.

  The components involved in this modification have no safety function and no changes have been made to the operational design to the system.

- The margin of safety as defined in the basis for any technical specification has not been reduced.

  Since the components involve in this modification are not included in the bases of any technical specification, the margin of safety has not been reduced.

10 CFR 50.59 allows changes to a facility as described in the FSAR if an unreviewed safety question does not exist and if a change to the technical specifications is not required. As shown in the preceding sections, the change proposed by this design package does not involve an unreviewed safety question because each concern posed by 10 CFR 50.59 that pertains to an unreviewed safety question can be positively answered. Also, no change to the Technical Specifications is required based on the above evaluation. Therefore, prior NRC approval is not required for implementation of this modification.

TITLE Condenser Expansion Joint Impingement Plate Modifications

DESCRIPTION OF CHANGE/ABSTRACT: The existing impingement plate design is inadequate for satisfactory long-term performance. Welded attachments on the plates have continuously failed, causing the plates to fall on and damage condenser tubes. The new plate design involves no welding and will prevent any further failures. The Condenser is a Non-Nuclear Safety Related Quality Group "D" Component. No changes to Technical Specifications are required, and no unreviewed safety questions are involved. This PCM will not affect plant safety or operation.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes _____ No ✓ A change to the plant as described in the FSAR?

Yes _____ No ✓ A change to procedures as described in the FSAR?

Yes _____ No ✓ A test or experiment not described in the FSAR?

Yes _____ No ✓ A change to the plant technical specifications?

EFFECT OF CHANGE

Yes _____ No ✓ Will the probability of an accident previously evaluated in the FSAR be increased?

Yes _____ No ✓ Will the consequences of an accident previously evaluated in the FSAR be increased?

Yes _____ No ✓ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

Yes _____ No ✓ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes _____ No ✓ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes _____ No ✓ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

Yes _____ No ✓ Will the margin of safety as defined in the bases to any technical specification be reduced?

METRASCOPE REPLACEMENT
ST LUCIE - UNIT NO 1
REA-SLN-87-56-10

ABSTRACT

St Lucie - Unit 1 currently utilizes a Metrascope System to monitor and display the Control Element Assembly (CEA) positions. This system will be replaced with a new one which has color graphics and a programmable computer for data processing and display creation. This will alleviate excessive calibration time, provide CEA displays more consistent with Unit 2, and modify Pre-Power and Power Dependent Insertion Limits (PPDIL/PDIL) which result in restricted CEA operation of several inches at full power. Additionally, the replacement will resolve eight open Human Engineering Discrepancies (HEDs) cited against the Metrascope System during the Detailed Control Room Design Review. The HEDs revolve around the existing system's display inadequacies and the lack of operator control over display generation.

The Control Element Assembly Position Display System (CEAPDS) is not a Safety Related system since it does not function to assure the integrity of the reactor coolant boundary, the capability for safe shutdown of the reactor, or the capability to prevent or mitigate the consequences of accidents which could result in off-site exposures described in 10CFR100. However, the proposed components will be seismically mounted in RTGB-104. Therefore this EP is classified as Quality Related.

The safety evaluation concluded that the modifications implemented by this EP do not involve an unreviewed safety question and that prior NRC approval for the implementation of this EP is not required. Since the monitoring function of the system will not be changed by the upgrade, there will be no effect on plant safety and operation. There is no change to the plant Technical Specifications.

SUPPLEMENT 1

This supplement to the Engineering Package adds a cable retractor for the CEAPDS CRT which will help protect its cables and a noise isolator to the Q-power input, which will prevent a potential ground fault from being transferred from the new CEAPDS to the RPS.

Similar to the original issue, this supplement is Quality Related. The modifications implemented by this supplement do not involve an unreviewed safety question, therefore prior NRC approval for its implementation is not required. There will be no effect on plant safety and operation or to the Plant Technical Specifications.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification because it does not modify or affect any Safety Related equipment and the new components are seismically mounted. Therefore it has no effect on the function of any equipment required to prevent or to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created since no new failure modes are introduced which could change the function of any Safety Related equipment.

(iii) The margin of safety as defined in the bases for any Technical Specification is not reduced since this modification does not reduce the operability of the rod block circuit or the CEA position indication systems. Instead, the modifications implemented by this EP will improve the operator's ability to determine the position of the CEAs and to identify limiting conditions.

The Control Element Assembly Position Display System (CEAPDS) is not a Safety Related system since it does not function to assure the integrity of the reactor coolant boundary, the capability for safe shutdown of the reactor, or the capability to prevent or mitigate the consequences of accidents which could result in off-site exposures described in 10CFR100. However, the proposed components will be seismically mounted in RTGB-104 and qualification of the board has been reviewed to ensure its seismic integrity. Therefore this EP is classified as Quality Related.

The implementation of this EP does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PCM is not required.

ST LUCIE PLANT - UNIT NO 1

RCP SEAL COOLER HEAT EXCHANGER TUBE LEAK DETECTION

ABSTRACT

This Engineering Package addresses the replacement of existing limit switches for Component Cooling Water (CCW) outlet valves HCV-14-11-A1, A2, B1 and B2 and minor wiring modifications to the valve control circuits. The replacement limit switches will modify valve position indication so that the indicating lights will discriminate between two (2) conditions: valve fully closed and not fully closed. The wiring modification to the valve control circuits consists of rewiring existing time delay relays to introduce a 60 second time delay. This time delay will allow sufficient CCW flow through the RCP Seal Cooler Heat Exchangers to normalize the temperature, thus, prohibiting the initial temperature differential from initiating inadvertant valve control lockout.

CCW to the RCP is classified as Non-Nuclear Safety Related and non-seismic according to St Lucie Plant - Unit 1 (PSL-1) FSAR Section 9.2.2.3. Also, the valve position indication circuits are Non-Nuclear Safety Related. However, since the function of the seal cooler isolation valves is to isolate reactor coolant leakage into the component cooling system, this EP is classified Quality Related.

The safety evaluation of this package indicates that neither the replacement of the limit switches nor the valve control circuit wiring modifications constitute an unreviewed safety question, and do not require a change in the Plant Technical Specifications. Therefore, prior NRC notification for implementation of this EP is not required.

This EP has no impact on plant safety and operations.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification. Electrical separation is maintained between safety related wiring and components. The modifications provided by this package have no impact on equipment important to safety and introduce no new failure modes. Therefore, this modification does not increase the probability of an accident or malfunction of equipment important to safety.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created by this modification. No new failure modes have been introduced as stated in section 2.1.8 of this EP.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since this modification does not degrade the CCW system, and the CCW Seal Coolers do not form the bases of any Technical Specification.

As described in FSAR section 9.2.2 the Component Cooling System is a closed loop cooling water system that utilizes demineralized water to cool various components. The modifications described in this PCM involve replacing existing limit switches and rewiring the associated CCW outlet valve circuits. These changes do not interrupt the closed loop Component Cooling Water System and are to a Non-Nuclear Safety Related valve indication function which discriminates between a fully closed and not fully closed valve position. However, since the function of the seal cooler isolation valves is to isolate reactor coolant leakage into the component cooling system, this EP has been determined to be Quality Related.

The implementation of this EP does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that these changes do not involve an unreviewed safety question and prior NRC approval for the implementation of this PCM is not required.

RCP VIBRATION MONITORING EQUIPMENT UPGRADE ST LUCIE - UNIT NO 1 REA-SLN-86-018

ABSTRACT

The Reactor Coolant Pump (RCP) Vibration Monitoring System is utilized to monitor the vibration of the RCP shafts on all four RCPs. It is made up of two radial probes (X & Y) located 90^{o} out of phase with each other just above the mechanical seal of each RCP. Vibrations sensed by the probes are monitored by four electronic modules mounted on the rear face of RTGB-104 in the Control Room.

This Engineering Package will implement an upgrade to the RCP Vibration Monitoring System which will include the replacement of the X and Y probes, their relocation to the lower motor shaft area of each pump, and the addition of a third probe (Keyphasor) in the same area of each pump to provide rotational phase position information. The four electronic modules of the existing system will be replaced by two modular instrument racks containing probe monitors for all twelve new probes, two pumps per rack.

The RCP Vibration Monitoring System is not a Safety Related system since it does not function to assure the integrity of the reactor coolant boundary, the capability for safe shutdown of the reactor, or the capability to prevent or mitigate the consequences of accidents. However, the proposed components will be seismically mounted in RTGB-104. Therefore this EP is classified as Quality Related.

The safety evaluation concluded that the modifications in the RTG board as implemented by this EP do not involve an unreviewed safety question and that prior NRC approval for the implementation of this EP is not required. Since the monitoring function of the system will not be changed by the upgrade, there will be no effect to plant safety and operation. There is no change to the Plant Technical Specifications.

SUPPLEMENT 1

Revision 1 of this Engineering Package has been issued to provide the installation of cable, conduit, and probes associated with Reactor Coolant Pumps (RCPs) 1B1 and 1A2, only. The safety evaluation remains valid with the implementation of this supplement; this EP does not involve an unreviewed safety question and prior NRC approval is not required for its implementation. This revision to the EP has no effect on plant safety or operation and does not involve any change to the Plant Technical Specifications.

SUPPLEMENT 2

Revision 2 of this Engineering Package has been issued to lift all remaining hold points to allow the installation of cable, conduit, and probes associated with Reactor Coolant Pumps (RCPs) 1A1 and 1B2 in order to complete the implementation of this PCM. In addition, the proximitors for the two pumps will be relocated in new electrical boxes. The safety evaluation remains valid with the implementation of this supplement; this EP does not involve an unreviewed safety question and prior NRC approval is not required for its implementation.

This revision to the EP has no effect on plant safety or operation and does not involve any change to the Plant Technical Specifications.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by the modifications in the RTG board as implemented in this Engineering Package because it does not modify any Safety Related equipment and involves the seismic installation of all RTG board components. Therefore, it has no effect on the function of any equipment required to prevent or to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created since no new failure modes are introduced which could change the function of any Safety Related equipment.

(iii) The margin of safety as defined in the bases for any Technical Specification is not reduced since this modification does not interface with equipment listed in the Technical Specifications.

The RCP Vibration Monitoring System is not a Safety Related system since it does not function to assure the integrity of the reactor coolant boundary, the capability for safe shutdown of the reactor, or the capability to prevent or mitigate the consequences of accidents. However, the proposed components will be mounted in RTGB-104 such that a seismic event will not cause them to damage adjacent Safety Related equipment. Therefore this EP is classified as Quality Related.

The implementation of this EP does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50/59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and that prior Commission approval for the implementation of this PCM is not requird for work to be performed in the RTG board.

ST LUCIE PLANT - UNIT NO 1 PCM 009-188

UPDATED OF LIMITORQUE EQ DOCUMENT PACKAGE
AND DISCONNECT SPACE HEATERS
(REA-SLN-87-007)

ABSTRACT

Limitorque valve operators whose limit switch compartments have been furnished with space heaters have been recognized by the NRC (IE Notice 86-71, "Recent Identified Problems With Limitorque Motor Operators") to pose a potential hazard to the internal wiring of the Limitorque operator. The hazard arises from internal limit switch compartment wiring potentially making contact with the energized space heater or the heater bracket. The resultant insulation damage could conceivably result in these wires becoming grounded to the limit switch housing. This Engineering Package will facilitate the removal of power to the space heaters thereby eliminating the problem.

The Limitorque valve operators of this Engineering Package are all Safety Related in as much as the valves they control perform nuclear safety related functions. Information relating to disconnecting power to the limit switch compartment space heaters will be included in the EQ Documentation Package for Limitorque motor operators. It is also the intent of this Engineering Package to remove from the Limitorque EQ Documentation Package Marathon 1600 terminal blocks which are not considered, at this time, to be suitable for use in Environmental Qualification (EQ) applications. Additionally the use of 3M taped splices is prohibited in Limitorque operators inside containment and this also will be reflected by revision to the Limitorque EQ Doc Pac.

Results of the safety evaluation conclude that modifications presented by this Engineering Package do not constitute an unreviewed safety question, do not require any changes to the Plant Technical Specifications and do not require prior Commission approval for the implementation of this PC/M.

The implementation of this PC/M will not have an adverse impact on plant safety or operations.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

To determine the effect of removal of the space heater with regard to the criteria outlined in 10CFR50.59(a)(1) which allows plant changes without prior Commission approval, providing that the changes do not involve a change to plant Technical Specifications or an unreviewed safety question, the following criteria were addressed as required by 10CFR50.59(a)(2):

1) The probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

Disconnecting the power to the space heaters does not increase the probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to safety previously evaluated in the FSAR since operability of the Limitorque motor operator is not dependent upon having the space heater energized. Limitorque does not recommend the use of energized space heaters for normal operation and has reported that their qualification testing was conducted with the space heaters de-energized. Limitorque furnishes space heaters for use during long term valve storage in an uncontrolled atmosphere.

Problems relating to potential motor operator inoperability were reported via IE Notice 86-71. This notice was in regard to the possiblity of damage to the control wiring of the motor operated valve due to contact with the space heater/heater bracket. Power to the limit switch compartment space heaters in all valves in this package except for five auxiliary feedwater valves is removed from terminals in the Motor Control Centers. The determinated conductors are taped and otherwise left in place. The five auxiliary feedwater motor operated valves have motor space heaters as well as limit switch compartment space heaters. The limit switch compartment space heaters were paralleled with the motor space heaters for the five auxiliary feedwater valves. Therefore, the determination of the valve operator limit switch compartment space heaters for the auxiliary feedwater valves will be made at the terminals in the valve limit switch compartment in order to maintain the motor space heaters energized. The 120 volt feeder to the motor compartment space heaters will be reconnected to the terminals in the limit switch compartment as will the leads to the valve motor enclosure space heater thereby keeping the motor space heater energized. The leads to the limit switch compartment space heater will be taped and left in place.

SAFETY EVALUATION (Continued)

The safety related valves have not been physically modified and their operation is to remain the same. Therefore, there is no change to their seismic or environmental qualification.

Prohibiting the use of Marathon 1600 terminal blocks for use with safety related valves and prohibiting the use of 3M taped splices inside of containment resolves concerns regarding the use of this equipment in view of NRC question regarding their suitablity.

2) The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR will not be created.

Removing the power to the space heaters will have no affect on the operability of the plant motor operated valves. As stated above, Limitorque does not recommend the use of energized space heaters except for valves in storage and their qualification testing was not conducted with the space heaters energized. Currently there are no Marathon 1600 terminal blocks in use with EQ related valves or 3M splices in use in containment. This Limitorque EQ Doc Pac update will prevent the possiblity of future accidents occurring due to the use of this material.

3) The margin of safety as defined in the basis for any Technical Specification is not reduced.

Since operability of the plant safety related motor operated valves is not affected by disconnecting the space heaters, since no Marathon 1600 terminal blocks are in use in the plant in association with EQ related valves and since no 3M splices are in use in containment, the basis for any Plant Technical Specification is unchanged. Therefore, Plant Technical Specifications are unchanged by disconnecting the space heaters to plant safety related motor operated valves or prohibiting the use of Marathon 1600 terminal blocks or 3M splice in containment.

The foregoing constitutes per 10CFR50.59(b) the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question or change to plant Technical Specifications.

CONCLUSION:

It is therefore recommended that space heaters be de-energized for all safety related Limitorque motor operators at the St Lucie Plant in response to IE Information Notice 86-71. This recommendation is based upon the manufacturer's recommendation, upon favorable industry and FPL experience with motor operators having their space heaters de-energized and upon reported problems relating to potential damage to the internal wiring of the motor operators as described in IE Notice 86-71. Unit 1 motor operators have their heaters fed from circuits which are common to other heaters for fans and motors. The Unit 1 space heaters must have their power leads lifted at the respective MCCs or as described above.

ST LUCIE PLANT - UNIT 1
STATION AIR/INSTRUMENT AIR
PRESSURE INDICATOR REPLACEMENT
REA-SLN-87-13-10

ABSTRACT

This Engineering Package replaces the existing voltage driven dual indicating meter (PI-18-9/PI-18-16) with a current driven device and modifies the 4-20mADC current loop so that the new indicator will be in series in the loop. Replacement of the existing dual indicator is required due to its failure; revision to the loop configuration will allow for the replacement of the indicator with parts maintained in stores inventory.

This meter provides control room indication of station air and instrument air pressure, neither of which are classified as nuclear safety related systems. This EP has no affect on any equipment required for safe reactor shutdown, used to mitigate the consequences of a design bases event (DBE), or control radioactive releases to the atmosphere in the event of a DBE. Since this EP involves the seismic analysis of mounting details for equipment mounted in the Reactor Turbine Generator Board (RTGB), this package is classified as Quality Related.

The implementation of this EP does not constitute an unreviewed safety question nor would its implementation affect the Plant Technical Specifications. Thus, Commission approval is not required prior to implementation.

This EP has no impact on plant safety or operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

SAFETY EVALUATION (Continued)

The modifications included in this Engineering Package do not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased. This is confirmed by the following:

The Compressed Air System (station air and instrument air) serves no safety function per St Lucie - Unit 1 FSAR Section 9.3.1.1. The replacement of the dual pressure indicator PI-18-9/PI-18-16 has no effect on any nuclear safety related equipment and its failure will not increase the probability of occurrence or the consequences of an accident as indicated in Section 2.1.8 of this EP.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated as confirmed by the following:

The modification of the compressed air instrumentation loop uses the same circuit design used throughout the plant and has been previously evaluated for both safety and non-safety loops.

Replacement of the dual pressure indicator PI-18-9/PI-18-16 provides control room indication of station air and instrument air pressure by utilizing a current driven device.

This configuration does not introduce any possibility of accident or malfunction not previously evaluated. See section 2.1.8 of this EP.

(iii) This modification does not reduce the margin of safety as defined in the bases for any technical specification since it has no negative effect on safety related components or systems as defined in any Technical Specifications and provides for station air and instrument air indication as originally specified in the St Lucie - Unit 1 Final Safety Analysis Report.

Since this package does not affect any equipment that is identified as nuclear safety related, this package need not be considered nuclear safety related. However, since the implementation of this PCM requires work to be done inside the reactor turbine generator board (RTGB), this package is classified Quality Related as the RTGB is a seismically designed control panel.

This EP does not involve any equipment on the Essential Equipment List and has no effect on safe reactor shutdown or alternate shutdown. There are no other changes to equipment which involve 10CFR50 Appendix "R" fire protection (see Attachment 7.1).

Implementation of Quality Related PCM 010-188 does not require any change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question nor a change to any Technical Specifications and prior Commission approval for the implementation of this PCM is not required.

ST. LUCIE UNIT 1
RAB RCB WALKWAY ENCLOSURE
REA-SLN-87-037

ABSTRACT

This engineering package covers the modification of the enclosed walkway which connects the Reactor Auxiliary Building and the personnel hatch enclosure at the Reactor Containment Building. The existing fiberglass panels which cover the walkway are being replaced with non-combustible materials. Also, the existing opening located at the south end of the personnel hatch enclosure will be sealed to prevent the entry of stormwater into the RAB RCB walkway and personnel hatch enclosures.

The existing RAB RCB walkway and personnel hatch enclosures do not perform any nuclear safety-related functions so this modification will not be classified as nuclear safety-related. However, this modification does require the installation of concrete expansion anchors in Seismic Class I structures including the Reactor Auxiliary Building. Since reinforcement steel in Seismic Class I structures could potentially be damaged during installation of this modification, quality-related requirements are applied to this design.

A safety evaluation of this modification has been performed in accordance with 10CFR50.59. This evaluation indicates that implementation of this Engineering Package does not involve an unreviewed safety question. Furthermore, the implementation of this modification does not require a change to the plant Technical Specifications and has no detrimental effect on plant safety and operation. Therefore, prior NRC approval for implementation of this modification is not required.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The modifications included in this engineering package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Updated Safety Analysis Report are not increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Updated Safety Analysis Report will not be created by this modification because the modification involves non-nuclear safety-related structures and failure of any items added by this modification will not impact any nuclear safety-related functions.

(iii) The margin of safety as defined in the bases for any technical specification is not affected by this modification since the components involved in this modification are not included in the bases of any Technical Specifications.

The RAB RCB walkway and personnel hatch enclosures are classified as non-nuclear safety-related. Due the location of the modification, failure of the RAB RCB walkway enclosure or the modified portion of the personnel hatch enclosure will not affect any nuclear safety-related equipment. However, this modification does involve the installation of concrete expansion anchors in Seismic Class I structures. Since steel reinforcement in the structures could potentially be damaged during installation of the anchors, quality-related requirements have been applied to this design.

The implementation of this EP does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PCM is not required.

TITLE: IB & ID Instrument Inverter Drawing Change

DESCRIPTION OF CHANGE/ABSTRACT: This change modifies a drawing (See drawing list) to show the correct circuit numbers for the 125VDC feeds to the IB & ID instrument inverters. No physical modifications are required, only a correction to a drawing. No unreviewed safety question or change to technical specification is required.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes	No	
___	X	A change to the plant as described in the FSAR?
___	X	A change to procedures as described in the FSAR?
___	X	A test or experiment not described in the FSAR?
___	X	A change to the plant technical specifications?

EFFECT OF CHANGE

Yes	No	
___	X	Will the probability of an accident previously evaluated in the FSAR be increased?
___	X	Will the consequences of an accident previously evaluated in the FSAR be increased?
___	X	May the possibility of an accident which is different than any already evaluated in the FSAR be created?
___	X	Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
___	X	Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
___	X	May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
___	X	Will the margin of safety as defined in the bases to any technical specification be reduced?

SECURITY LIGHTING PANELS - RELAYS AND CONTACTORS

DESCRIPTION OF CHANGE/ABSTRACT: These changes provide for the replacement of a failed ASCO relay utilized in Security Lighting Panel 2B, and for the documentation of as-built conditions. An unreviewed safety question does not exist and there is no change to technical specifications involved.

Supplement No. 1 - This supplement is to add page 3a (Design Interface Record) and to indicate the purchasing quality level on attachment 1 Sh. 2 of 2. An unreviewed safety question does not exist and there is no change to technical specifications involved.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes ____ No __X__ A change to the plant as described in the FSAR?

Yes ____ No __X__ A change to procedures as described in the FSAR?

Yes ____ No __X__ A test or experiment not described in the FSAR?

Yes ____ No __X__ A change to the plant technical specifications?

EFFECT OF CHANGE

Yes ____ No __X__ Will the probability of an accident previously evaluated in the FSAR be increased?

Yes ____ No __X__ Will the consequences of an accident previously evaluated in the FSAR be increased?

Yes ____ No __X__ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

Yes ____ No __X__ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No __X__ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ____ No __X__ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

Yes ____ No __X__ Will the margin of safety as defined in the bases to any technical specification be reduced?

ST LUCIE PLANT - UNIT NO 1
ICW LUBE WATER PIPE RESTRAINT MODIFICATIONS

ABSTRACT

The tornado missile barriers around the intake cooling water pumps are exposed to salt water spray from the pump packing. The barriers are constructed of coated carbon steel and have suffered corrosion from the salt water exposure. In particular, the structural members near the bottom of the enclosure on the east and west faces have experienced severe deterioration. Several of these members furnish support for ICW lube water system pipe restraints.

An evaluation of the as-found condition is being performed as part of the overall effort associated with the disposition of NCR 1-133. Pending a long term solution to correct the root causes of the corrosion problem, this Engineering Package is being issued to modify those corroded structural elements which are integral parts of the pipe restraints.

This Engineering Package does not involve an unreviewed safety question and has no effect on plant safety or operation, nor does it require a change to the plant Technical Specifications The system involved is classified as Safety Related, consequently this Engineering Package is also classified as Safety Related.

SAFETY EVALUATION

Safety Analysis

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This Engineering Package provides modifications to pipe restraints for the ICW lube water system, which is a safety related system. Accordingly, this Engineering Package has been classified as Safety Related. It does not involve an unreviewed safety question. The following are the bases for this conclusion:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification restores and enhances the original design margin of the affected restraints and will be performed in accordance with Safety Related requirements, hence there can be no impact on any Safety Related structures, systems, or equipment.

(ii) There is no possibility for an accident or malfunction of a different type than any evaluated previously since there is no potential for the interaction of these modifications with any Safety Related equipment or systems other than the ICW lube water system itself and the ICW pump missile barrier to which the affected restraints are attached; the restoration of these components to their original design margin will not affect any safety related systems or equipment.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification as the modifications have been designed to the same criteria as the restraints of which they are a part.

The implementation of this Engineering Package does not require a change to plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question or a Technical Specification change and thus prior Commission approval for the implementation of this Engineering Package is not required.

FLORIDA POWER & LIGHT COMPANY
ST LUCIE PLANT - UNIT NO 1
CONDENSATE PUMP DISCHARGE SAMPLING LINES
REA-SLN-86-061-92

ABSTRACT

This Engineering Package provides details for the addition of condensate sampling points downstream of each condensate pump (1A, 1B, 1C) and in the common discharge line for all the pumps. It also provides for the connection of these sample points, through a valve manifold, to an existing sample line to the Chemical Analyzer in the Cold Chemistry Laboratory.

This EP is classified as non-safety related since it provides for a modification to a non-safety related system. The safety evaluation has shown that this EP does not constitute any unreviewed safety question. The implementation of this EP does not require a change to the Plant Technical Specification; therefore prior NRC notification for implementation of the EP is not required.

This sampling system is non-safety related and will have no effect on equipment vital to plant safety, nor will it effect plant operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This EP provides for the addition of sampling points in the condensate lines, connection of these points to a valve manifold station and then to the existing common line to the Cold Chemistry Lab. Stainless steel tubing with compression fittings will be used for the sample lines. The EP has been classified as non-safety related and does not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The St Lucie Unit No 1 FSAR, Section 10.4, "Steam and Power Conversion System", states that the features and components of this system, which includes the condensate system, serve no safety function since they are not required for safe shutdown or to mitigate the effects of a LOCA. This modification is on a non-safety related system and will have no effect on equipment vital to plant safety.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.