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L-89-133
10 CFR 50.59

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

Gentlemen:

Re: St. Lucie Unit 2
Docket No. 50-389
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 October 7, 1987 and October 6, 1988 and correlates with the information included in Revision 4 of the Updated Final Safety Analysis Report.

Very truly yours,

W. F. Conway
Senior Vice President - Nuclear

WFC/JRH/gp

Enclosure

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

IE47
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Re: St. Lucie Unit 2
Docket No. 50-389
Report of 10 CFR 50.59 Plant Changes

ST. LUCIE PLANT UNIT 2

REPORT OF CHANGES MADE

UNDER THE PROVISIONS OF

10 CFR 50.59

FOR THE PERIOD ENDING OCTOBER 6, 1988

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PLANT CHANGE/MOD REVIEWED FOR PSL2 FSAR AMENDMENT 4

NUMBER	REVISION	TITLE
095-283	0-2	SHUTDOWN COOLING PURIFICATION SYSTEM
323-283	0	SECURITY CONSOLES LED GRAPHIC DISPLAY-UNIT2 POWER SUPP ANN
400-283	0	UGS LIFT RIG TRIPOD MOD
027-284	2	MFIV - MAIN FEEDWATER ISOLATION VALVE PRESS SWITCH REPL
045-284	0	CEDMCS CABINET COOLING
054-284	0	FOXBORO RECORDER MODEL 226S CHANGE
129-284	0	MAIN PURGE SYSTEM/LLRT TAPS
156-284	0-1	CONTINUOUS MONITORING EQUIPMENT CABLE MODIFICATION
163-284	3	SPENT FUEL GATE STORAGE AREA MODIFICATION
203-284	1-2	HYDROGEN DETECTION INSIDE EXCITER HOUSING
247-284	0	480V BUSES CV-2 UNDERVOLTAGE RELAY MODIFICATION
[illegible]-284	0	CONTAINMENT ANNULUS AIR SUPPLY
009-285	0	ICW SYSTEM ORIFICES
011-285	0	STEAM TRAP DRAIN PIPING AS-FAIL REPLACEMENT
035-285	0	FUEL TRANSFER TUBE SHIELDING
048-285	0	GE SAM RELAY PC CARD REPLACEMENT
058-285	3	NEW FEED TO 480V POWER CENTER 2A5
085-285	2	ICW PUMP EXPANSION JOINT REPLACEMENT
093-285	0-2	PRESSURIZER MANWAY LIFTING LUG MODIFICATION
102-285	0	REMOVAL TEMP S/U STEAM SUPP PIPING
106-285	0-1	PSB-1 UNDERVOLTAGE RELAY CABINET ENHANCEMENT
138-285	0	2B CHARGING PUMP DISCHARGE RESTRAINT ADDITION
143-285	0	LINEAR TRIPTEST POTENTIOMETER REPLACEMENT
149-285	0	CONDENSATE PUMP MINIMUM RECIRCULATION SYSTEM MODIFICATIONS

PLANT CHANGE/MOD REVIEWED FOR PSL2 FSAR AMENDMENT 4

NUMBER	REVISION	TITLE
150-285	0-1	CONDENSATE PUMP MINI-RECIRC PIPING
163-285	0	REACTOR HEAD VENT LINE RESTRAINT MODIFICATION
196-285	0-1	ANALOG DISPLAY SYSTEM GRAPHIC DISPLAY SPARES
203-285	0	CCW BACKFLUSH STRAINER DRAIN
208-285	0	RTD & THERMOWELL REPLMT-RCS
211-285	0	MAKEUP AIR FOR CONTAINMENT HYDRO PURGE SYS TEMP VLV MOD
008-286	0	ATMOS DUMP VLV MOV MOD
011-286	0-1	DG GOVERNOR POWER SUPPLY
014-286	0-1	TARGET ROCK VALVES-STEM ASSEMBLY UPGRADE
024-286	0	RDF-RTD TEMPERATURE TRANSMITTER REPLACEMENT
[illegible]-286	0	CCW PUMP BEARING MATL CHANGE
038-286	0-2	PCB TRANSFORMER REPLACEMENT
042-286	0	QUENCH TANK PMW ISOLATION VALVE REPLACEMENT
062-286	0	ADD FLANGE - PENET P-50
064-286	0	RCP INSULATION REPL
069-286	0	TORQUE SEATING-ATMOSPHERIC DUMP VALVES
075-286	0-1	HEATER DRAIN PUMP MECHANICAL SEAL DEMINERALIZED WATER SUPPLY
087-286	0	MISAPPLICATION OF LIMITORQUE OPERATORS
091-286	0-1	CLOSE INTERCEPT VALVE CIRCUIT MODIFICATION
092-286	0	ADDITIONAL APPENDIX R FIRE SPRINKLER AND FIRE WRAP IN RAB
108-286	0	HIGH INITIAL RESPONSE EXCITATION SYSTEM
113-286	0	DIESEL GENERATOR CRANK CASE OIL DEFLECTOR PLATE
120-286	1	10 CFR 50.49 ENVIRONMENTAL QUALIFICATION LIST REVISION
[illegible]3-286	0	PRESSURIZER MISSILE SHIELD ACCESS LADDER SAFETY CAGE
[illegible]24-286	0	ICW LUBEWATER FLOWRATOR MODIFICATION

PLANT CHANGE/MOD REVIEWED FOR PSL2 FSAR AMENDMENT 4

NUMBER	REVISION	TITLE
129-286	0	S/U XFMR L/O DISC SWITCHES
134-286	0	QSPDS SOFTWARE MODIFICATION
002-287	0	IE BULLETIN 85-03 MOV SWITCH SETTING
006-287	0	NRC IE BULLETIN 85-03 MOV POSITION INDICATION
007-287	0	HP TURBINE INNER GLAND AND GLAND DIAPHRAGM ENHANCEMENTS
019-287	0-2	DIESEL GENERATOR TORSIONAL VIBRATION ISOLATION
026-287	0	FIRE PROTECTION STRUCTURAL STEEL FIRE PROOFING
029-287	0	TURBINE GENERATOR ADDITONAL OIL SEAL FOR 1 AND 2 BEARING
033-287	0	REPLACEMENT OF VALVE V3734
040-287	0	CONDENSATE RECIRC TO COND PNEUMATIC SQ RT EXTRACTOR REPL
[illegible]-287	0	MFRV POSITION INDICATOR
048-287	0-1	REACTOR CAVITY SEAL RING SEAL
050-287	0	CONDENSATE PUMP EXPANSION JOINT REPLACEMENT
051-287	0-1	CONDENSER HOTWELL NITROGEN INJECTION CONNECTIONS
052-287	0	CONDENSATE POLISHER TIE-IN
055-287	0	MSR PARTITION PLATE NUT REPLACEMENT
056-287	0-1	480V SWITCHGEAR 2A1 AND 2B1 TRANSFORMER REPLACEMENT
058-287	0	BORIC ACID MAKEUP SYSTEM RELIEF VALVE MODIFICATIONS
059-287	0-1	LOW POWER FEEDWATER CONTROL SYSTEM
061-287	0-1	INSTALL VERNIER MERCURY MANOMETERS
062-287	0	ANNUNCIATOR NUISANCE ALARMS
063-287	0	NEW FUEL CRANE INTERLOCK ADDITION
068-287	0	MFIV TERM STRIP REPL
069-287	0	RELOCATION OF THE SBVF HEATER CONTROL PANELS
[illegible]-287	0	REPLACEMENT OF FISCHER AND PORTER CONTROLLERS

PLANT CHANGE/MOD REVIEWED FOR PSL2 FSAR AMENDMENT 4

NUMBER	REVISION	TITLE
077-287	0-1	ERDADS
079-287	0	EXTRACTION STEAM PIPE AND FITTING MATERIAL UPGRADE
083-287	0	MISCELLANEOUS ICW SYSTEM MODIFICATIONS
086-287	0	CONDENSER OUTLET TUBESHEET AND WATERBOX COATINGS
089-287	0	REMOTE REACTOR VESSEL LEVEL INDICATION
091-287	0	REACTOR HEAD TORUS RING MODIFICATION
092-287	0-2	REPLACEMENT OF SAFETY RELATED BATTERIES 2A AND 2B
096-287	0-1	PRESSURIZER INSTRUMENT NOZZLE REPLACEMENT
102-287	0	RCA PROTECTIVE CLOTHING BINS
103-287	0	TSI THRUST BEARING PROBE RELOCATION
[illegible]-287	0	MISCELLANEOUS SNUBBER MODIFICATIONS
114-287	0	LC XFRMR VLV PACKING MODS
115-287	0	INSTRUMENT AIR AFTER FILTER ISOLATION VALVES AND BYPASS LINE
117-287	0	CONDENSER EXPANSION JOINT IMPINGEMENT PLATE MODIFICATION
118-287	0	CABLE SUPPORT STRUCTURE CONNECTION MODIFICATION
120-287	0	GROUTING OF MASONRY BLOCK WALLS
121-287	0	STEAM GENERATOR TUBE PLUGGING CE DESIGN PLUGS
122-287	0	2A/B SPARE STM GENERATOR INSTR NOZZLES CLOSURE MODIFICATION
126-287	0-1	INCORE INSTRUMENTATION (ICI) PLATE MODIFICATION
131-287	0	AS-BLD CCW SUPPORT
133-287	0	WELD - LINE WM-B-80
136-287	0	CHECK VALVE V27101 REPLACEMENT
137-287	0	RCS INSTRUMENT NOZZLE INSULATION TEMPORARY MODIFICATIONS
[illegible]-287	0	SPLICE BOXES B2124,34,35
139-287	0	REPLACEMENT OF I-FCV-25-7,8 ACCUMULATOR CHECK VALVES

PLANT CHANGE/MOD REVIEWED FOR PSL2 FSAR AMENDMENT 4

NUMBER	REVISION	TITLE
150-287	0	CEA MG SETS LOCKOUT RELAY
155-287	0	EXCORE S/U & CNTL CHAN MODS
027-288	0	ROSEMOUNT XMTR F19021
028-288	0	REPL PT-22-23
044-288	0	EQ LIST REV - SPARE PARTS
048-288	0	DWG CLARIFICATION-GEN PROT RLY & ICW INSTR
056-288	0	ICW & CW PUMP PACKING REPL
066-288	0	EQ DOC PAC LIMITORQUE MOTOR OPERATORS
080-288	0	NAMCO LIM SW REPL-PCV-8801-5
096-288	0	DWG CLARIFICATION-ENG SAFEGRDS CAB
[illegible]-288	0-1	PDIS REPLACEMENT
102-288	0	EMDRAC DRAWING 2998-730, REV 4
108-288	0	FUEL POOL PURIFICATION SYS PUMPS MECH SEAL REPLACEMENT
110-288	0	CONDENSATE RECOVERY SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT
112-288	0	TURBINE GLAND SEAL SYSTEMS PUMPS MECHANICAL SEAL REPLACEMENT
149-288	0	DOC CORR PS 29-4,4-1,4-2
227-984	1	TURBINE GANTRY CRANE SEPARATION REQUIREMENTS
087-985	0	HYPOCHLORITE CELL FLUSH SYSTEM
020-986	0	INTAKE CANAL DREDGING AND SLOPE RESTORATION
039-986	0	BLOWDOWN BUILDING RADIATION MONITORING SYSTEM
111-986	0	SIMULATOR TRAINING FACILITY PIPING TIE-INS
106-988	0	SG BLOWDOWN TREATMENT FACILITY SYSTEM PUMPS MECH SEAL RPLCMT

SHUTDOWN COOLING PURIFICATION SYSTEM

INTRODUCTION

The plant requested via DIR M-43 that the Shutdown Cooling Purification System, which is presently arranged using hoses connected to flanged pipe taps in the Safety Injection and Chemical Volume Control Systems, be hard piped. Such a modification would result in a savings in set-up and maintenance time at the start and completion of a refueling outage.

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 SDC Purification System is composed of piping, supports and manual valves and does not consist of any active components. The system is utilized during shutdown conditions only, when RCS temperatures are less than 140°. The piping and supports are designed to ASME III, Class 3 criteria and appropriate portions of the piping are seismically supported to withstand the applicable loadings listed in Chapter 3 of the FSAR and to maintain the seismic separation criteria from safety related equipment and piping. Therefore, there is no increase in the probability of an accident or malfunction previously evaluated in the safety analysis report.

Isolation valves separating the SDC Purification system from the Safety Injection (SI) and CVCS systems are designated as normally closed and locked closed during normal plant operation. As such, these valves are verified closed by plant administrative procedures prior to reactor start-up. Therefore, the possibility of an accident or malfunction of a different type than any previously evaluated in the safety analysis report is not created.

The SDC purification system performs no safety function and is used only as a clean-up system for the RCS during plant shutdown. Therefore, the margin of safety as defined is the basis for any technical specificatin is not reduced.

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

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

SECURITY CONSOLES LED GRAPHIC DISPLAY - UNIT 2

INTRODUCTION

The NRC has determined that annunciation of the Security System power supplies is required for compliance with 10CFR Part 73 (i.e. requirements for security systems for nuclear power plants). To meet the intent of this requirement, status lights shall be installed on the security system alarm consoles to indicate the "at hand" condition of the power input to the security SUPs and therefore, to the entire security system.

SYSTEM DESCRIPTION

The status of the security system is continuously monitored by security personnel, who utilize the central Alarm Station (CAS), and the Secondary Alarm Station (SAS). CAS and SAS are independent, manned stations.

This PC/M package provides the design and installation details that are necessary to install indicating lights on the CAS and SAS consoles, which will provide visual indication of the status and lineup of the power supply to the security SUPS and therefore to the security system. These lights provide the following information:

1. "Normal" - Indicates that the security SUPS is powered by the normal operating plant equipment lineup.

2. "Diesel" - Indicates that the diesel generator breaker is closed and the diesel generator is supplying power to the plant loads.

3. "Bypass" - Indicates that the static transfer switch has been placed in the manual bypass position.

   Note: The "Bypass" indication will be accompanied by a "Normal" or a "Diesel" indication.

4. "Battery" - Indicates that AC power is not available.

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 security system is a non-safety related plant system. the Central and Secondary Alarm Stations are components of this system. The modifications presented in this PC/M affect both safety and non-safety related plant equipment.

SAFETY ANALYSIS (Continued)

The modifications to the CAS and SAS control panels, i.e. installation and wiring of the annunciator circuitry, and the inputs to these annunciators are non-safety related. The alarm stations are non-safety, non-seismic structures. The majority of required cable to these areas will be routed in non-safety related cable tray. The balance of cable will be routed through appropriately dedicated raceway.

Diesel generator breaker position is monitored to provide input to the "Normal" and "Diesel" annunciator circuits. This portion of the diesel generator control circuitry is safety related. Therefore, this signal will be isolated from the non-safety security annunciation circuitry by utilizing existing isolation relays. These relays were provided as part of PC/M 015-283 and have already been qualified to the applicable industry standards.

The balance of the control relays that are required in this modification have been purchased and will be installed as non-safety related equipment.

Control power to all relays is from the associated plant power train (safety to isolation relays, non-safety to the non-safety control relays). All cables will be routed through the appropriate raceway and the raceway will be seismically supported as required (i.e. inside the RAB).

This modification has no impact on 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 any unreviewed safety question, therefore prior Commission approval is not required for implementation of this PC/M.

UPPER GUIDE STRUCTURE LIFT RIG TRIPOD MODIFICATION

ABSTRACT

This engineering package provides the details to modify the UGS lift rig tripod to achieve the required clearance to permit the refueling machine to pass over the lift rig when the lift rig is mounted on the UGS in the refueling pool. This modification is necessary in order to minimize time consuming efforts to disassemble and move the tripod out of the path of the refueling machine during each outage.

Although the UGS lift rig does not perform a safety-related function as defined by FSAR Section 3.2.1, its failure during its operation could result in damage to nearby safety-related equipment. Accordingly, quality-related requirements have been applied to this design.

NUREG 0612, "Control of Heavy Loads", and the associated requirements of ANSI N14.6 have been reviewed and any applicable requirements of these documents have been incorporated into the design of the new tripod.

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 commission approval for the implementation of this modification is not required.

SAFETY EVALUATION

The new tripod assembly installed by this engineering package does not perform or affect any safety-related function and will only be used during refueling operations. Since failure of the tripod while lifting the UGS or CSB 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 requirements have been applied to the design.

The new tripod assembly has been structurally analyzed 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 the analysis demonstrate that the new components are all within allowable stress levels. Additionally, a review was performed to verify the acceptability of storing the original tripod in the refueling pool while attached to the core support barrel lift rig.

The implementation of this modification has negligible impact on the containment heat sink, hydrogen generating source, and free volume analyses described in FSAR Section 6.2.

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 9.1-13, Section 9.1.4.2.2.6 and Tables 6.2-7 and 6.2-8 must be updated to reflect the addition of the new tripod assembly.

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

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 this modification does not affect the availability, redundancy, capacity, or function of any equipment to mitigate the effects of an accident.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the function of the tripod has not been altered.

(iii) The margin of safety as defined in the bases of any technical specification is not reduced since the modified tripod performs no safety-related function and is not included in the bases of any technical specification.

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 is not required.

MAIN FEEDWATER ISOLATION VALVE PRESSURE SWITCH REPLACEMENT

SYSTEM DESCRIPTION

For each main feedwater isolation valve (HCV-09-1A, 2A, 1B & 2B), there is a pressure switch that senses air reservoir pressure and another switch that senses air supply pressure. The air reservoir pressure switches, PS-09-1A2, PS-09-2A2, PS-09-1B2 & PS-09-2B2, are presently Barksdale series model B2TA12SS from Transamerica Delaval. Their function is to monitor actuator air loss past the air check valve by signaling if pressure drops below 70 psig in the air reservoir. The air supply pressure switches, PS-09-1A3, PS-09-2A3, PS-09-1B3 & PS-09-2B3 (same manufacturer and model number), monitor plant air supply loss to the actuator by signaling if pressure drops below 80 psig at the air filter. These switches are very inaccurate since the setpoints are near the low end of their range (50-1200 psig). In addition, the existing switches are not rated to the DC voltage being supplied.

The purpose of this PC/M is to replace the referenced switches with Static-O-Ring models 6N6-BB5-NX-C1A-JJTTX6 (air reservoir) and 6NN-LL5-C1A (air supply). These switches have an adjustable range of 20 to 180 psig and will maintain the same setpoints of 70 and 80 psig respectively.

The function of the air reservoir and air supply pressure switches is not nuclear safety related. The switches are used strictly for annunciation of pressure drops below the assigned setpoints. However, because the air reservoir pressure switches are considered pressure boundary safety related, the PC/M is nuclear safety related. In addition, since the pressure switches are electrically connected to nuclear safety related power supplies these switches will be evaluated and demonstrated that their failure does not preclude the actuation of the main feedwater isolation valves (MFIVs).

SAFETY ANALYSIS

This PC/M is nuclear safety related because the air reservoir pressure switches (PS-09-1A2, PS-09-2A2, PS-09-1B2 & PS-09-2B2) on the Main Feedwater Isolation valves are part of a safety related pressure boundary. The air supply pressure switches (PS-09-1A3, PS-09-2A3, PS-09-1B3 & PS-09-2B3) are considered safety related because they are electrically connected to a nuclear safety related power supply, however, after performing a failure mode and effects evaluation it is shown that any failures to the air supply pressure switches do not propogate and affect the actuation of the MFIVs. Therefore, these switches can be considered non-nuclear safety related. Appendix D provides an evaluation of the failure modes for the air reservoir pressure switches and the air supply pressure switches. The function of the air reservoir and air supply pressure switches being replaced by this modification is not nuclear safety related since their function is to provide annunciation and lamp indication whenever pressure drops below their assigned setpoints. However, since the air reservoir pressure switches are pressure boundary safety related, the portion of the PC/M pertaining to these switches is considered nuclear safety related. The air reservoir pressure switches are qualified seismically to prevent the loss of the safety related pressure boundary.

The replacement pressure switches assigned to monitor air reservoir pressure (PS-09-1A2, PS-09-2A2, PS-09-1B2 & PS-09-2B2) shall be Static-O-Ring model 6N6-BB5-NX-C1A-JJTTX6. These switches have been qualified to IEEE-344-1975 standards as per test report no 17344-82N-D prepared by Acton Environmental Testing Corporation (AETC) for Static-O-Ring, Inc. Mounting of these switches, have been evaluated for seismic category I loadings,

This change is not an unreviewed safety question because: the probability of occurrence or the consequence of an accident of malfunction previously evaluated in the FSAR has not been increased. The new pressure switches are of approximately the same weight as the presently installed switches, and will be installed per the same requirements that applied to the existing switches. The process inputs will remain the same as the existing switches: 1/4 NPT.

This modification will improve annunciation by replacing the existing switches with new switches which are less subject to setpoint drift and have a wider adjustable range. In addition, the new switches will satisfy the specifications with respect to voltage and amperage ratings.

For the same reason, no possibility for an accident or malfunction of a different type from any evaluated previously in the FSAR has been created by this modification. Additionally, the margin of safety, as defined in the bases for the technical specifications, has not been decreased. In conclusion, this modification does not involve and unreviewed safety question.

CONTROL ELEMENT DRIVE MECHANISM/CONTROL SYSTEM (CEDMCS) CABINET COOLING

Description of Change

With a CEDMCS cabinet area ambient temperature or 76°F, cabinet discharge temperatures in excess of 120°F have been measured, accompanied by persistent cooling failure alarms. Such excessive inner-cabinet temperatures will reduce component lifetimes, resulting in costly premature failures.

This PC/M will add eighteen (18) exhaust fan assemblies (one per bay) to CEDMCS Cabinets C2 through C5 for the purpose of reducing internal cabinet temperatures.

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; of (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 devined in the basis for any technical specification is reduced.

For the following reasons, C-E concludes this change does not involve any unresolved safety questions as defined in items (i), (ii), or (iii) above:

1. CEDMCS is a non-safety grade system;
2. This modification does not affect any of the isolation devices used to interface the CEDMCS Cabinets with Safety Related Equipment/Systems;
3. The system affected by this modification has not been used as a basis for any technical specifications;
4. This modification will reduce the probability of equipment malfunction by reducing the thermal stresses exerted on electronic components.
5. Since the CEDMCS is not seismically qualified, this modification does not require a seismic reanalysis.

The implementation of the 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 and prior commission approval for the implementation of this PC/M is not required.

FOXBORO RECORDER MODEL 226S CHANGE

System Description

The St. Lucie Unit 2 Instrument List calls for Foxboro Recorder Model 226S for the following tag numbers in the Safety Injection System.

Recorder	Pens	Board	Recorder	Pen	Board
JR-001A	1	RTGB-204	PR-3301	1	RTGB-206
JR-001B	1	RTGB-204	PR-3302	1	RTGB-206
JR-001C	1	RTGB-204	PR-3305	1	RTGB-206
JR-001D	1	RTGB-204	PR-3306	1	RTGB-206
LR-110X/PR-1108	2	RTGB-203	TR-3303W	1	RTGB-206
FR-3301	1	RTGB-206	TR-3303Z	1	RTGB-206
FR-3306	1	RTGB-206	tr-3351	2	RTGB-206
FR-3313/FR-3323	2	RTGB-206	TR-3352	2	RTGB-206
FR-3317	1	RTGB-206	PR-8013D/PR-8023D	2	RTGB-206
FR-3327	1	RTGB-206	PR-9013D/LR-9023D	2	RTGB-206
FR-3333/FR-3343	2	RTGB-206			

This PC/M allows the use of either the Foxboro Model 226S or Model 227S for the instruments identified above. Either model recorder can be removed and replaced with the other model recorder without any wiring changes or input signal modifications. The change is simply the replacement of the Model 226S with the Model 227S or vice versa.

The Instrument List will be "as-built" for these recorders to read Foxboro Recorder Model 226S or Model 227S. See Appendix A for the specific changes.

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.

These recorders are safety related since they are for the Safety Injection System. There are no unreviewed safety quesitons, since both recorder models are seismically qualified Category I and functionally equivalent. The environmental design of the Foxboro recorders satisfies the mild environment of the Control Room and the design specifications for the control boards, that they are bein installed in. The Model 227 recorders shall also be ordered with a certificate of compliance that they are functionally interchangeable with Model 226 recorders that have been previously qualified.

MAIN PURGE SYSTEM/LLRT TAPS

SYSTEM DESCRIPTION

The purge system exhausts to the environment via the plant stack. The system has a capacity of 42,000 cfm and is operated during refueling mode only. The system is not required to operate during short term access to the containment.

During normal refueling purge, the containment air is drawn through penetration P-10, which includes butterfly isolation valves I-FCV-25-4, -5 and -6 into a filter casing. (See FSAR Fig 9.4-8). A valved (I-V-25-207-324P) test tap and plug is provided in the penetration pipe in the containment side of isolation valve FCV-25-5.

The air make-up side of the purge system includes penetration P-11 and isolation butterfly valves I-FCV-25-1, -2 and -3 in the direction of flow.

All six isolation valves in penetrations P-10 and P-11 close automatically on Containment Isolation Actuation Signal (CIAS).

Similarly to penetration P-10, a valved (I-V-25-210-324P) test tap and plug is provided in the penetration in the containment side of isolation valve FCV-25-2.

Technical Specification 4.6.1.7.3 Surveillance Requirements states that "At least once per 6 months on a STAGGERED TEST BASIS each sealed closed 48 inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

To perform the above surveillance requirement with the present facilities, the personnel performing the Local Leak Rate Test, has to enter the annulus, with the test equipment and transport the equipment and himself past hot piping areas and use a ladder to reach the present valves and test plugs.

This PCM will modify the means to perform the Local Leak Rate Tests for penetrations P-10 and P-11 from floor elevation 23 feet by providing test stations inside the annulus near the SW access door.

Each new LLRT station consists of an isolation valve and a plugged test connection to duplicate the facilities provided by the original design. A bleed valve open to atmosphere has been added to each test station to provide the means for controlled depressurization of the penetration or additional instrument connection.

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 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 possiblity 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 inthe basis for any technical specification is reduced.

The containment purge system is not a safety related system and is not required to operate following a design basis accident. It is required to purge the containment to allow required access time for the plant personnel during planned shutdown and refueling operations. The system requires approximately 15 hours to reduce c/mpc to 1.0. A radiation monitor is provided in the plant stack to monitor the radiation level of gases being discharge.

Isolation valves and containment penetrations are designed to Quality Group B and seismic Category I. The extended installation to the new location for the LLRT test taps in the annulus is designed as seismic Category I.

The containment purge system penetration is not subject to bypass leakage testing since its penetrations are filtered by the shield building ventilation system. This modification will not change that statement. The addition of the test tubing will not adversely affect the limits allowed by the Technical Specifications.

This Plant Change Modification does not change the philosophy of the main purge system or the Local Leak Rate Test Tap for penetrations P-10 and P-11. The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analsyis Report or the Safety Evaluation Report has not been increased nor has a new situation been created. The margin of safety as defined in the basis for the Technical Specifications has not been changed.

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 PCM.

CONTINUOUS MONITORING EQUIPMENT CABLE MODIFICATION

System Description

The Continuous Monitoring Equipment is a system that has been engineered and purchased by Florida Power and Light Co. for the purpose of monitoring and recording the electrical generation parameters of the St Lucie Plant Unit #2 (a similar installation exists for St Lucie Plant Unit #1). Inputs are taken from various locations in the plant system. This data is collated and recorded at the Continuous Monitoring Equipment cabinet located at Elevation 43.0 in the Reactor Auxiliary Building.

The following CME cable modifications are addressed under this PC/M package:

1. Cables 20916F and 20917F referenced previously on CWDs 2998-B-327 sheets 916, 917, 918 and 919, were intended to provide metering signal from 4160 volt switchgear to CME cabinet via RTGB 201. These cables, however were not installed. Review of the cable routing indicated that due to inaccessibility of conduits in RTGB 201, it was not feasible to utilize the route via RTGB 201. Revised control wiring diagrams employ the alternate route which provides cabling from 4160 volt switchgear to the CME cabinet. This change results into the routing of four cables 20916F, 20917F, 20918H and 20919 H. Pull cards for these cables are included in this PC/M package.

2. Points D15 and D16 on the Data Acquisition Package (DAP) are being used as junction points to tie the field current ammeter (AM-872) to the CME channel 20 (CWD #871), this is not as shown on CWD 2998-B-327, sheet 872. The internal wiring from T17-61 and 62 (sh.872) to D15 and D16 (sh.871) on the DAP will be removed. New wiring will be provided from T17-61 and 62 to T13- 90, 91 and 92. The CWDs have been corrected, to incorporate this change.

3. Underfrequency relays 81F2 and 81F4 on CWD 882 show an internal jumper between actuating and seal-in contacts. This jumper was missing so an external jumper was added. This is shown on revised CWD.

4. Watt-hour meter WHM/881 on CWD 881 had input connections reversed. Connections to terminals 3-4 should be 4-3, 7-8 should be 8-7. The CWD has been revised.

5. Startup transformer MOD indicating lights have connections reversed. Cable connections to lights for startup transformer A and B should be swapped. This will be shown on revised CWD 1105. Cables remain in place. Internal jumpers will be used to prevent repulling.

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 malfunciton 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 operation of the Continuous Monitoring Equipment enhances the St Lucie Unit 2 generation monitoring system by providing a capability to record parameters such as output voltage, current and power (KW and kVAR), generator field current and circuit breaker status.

The modifications entailed in this PC/M are non safety related and are required for the complete operation of the Continuous Monitoring Equipment.

The cable routings have been designed in accordance with the St Lucie 2 cable ampacity, tray fill and support criteria. Furthermore, the existing raceway is seismically supported.

The foregoing constitutes, per 10 CFR 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.

SPENT FUEL GATE STORAGE AREA MODIFICATION

INTRODUCTION

PCM 163-284, "Spent Fuel Gate Storage Area Modification", was initially implemented to modify the spent fuel cask pit bulkhead storage rack located on the north wall of the spent fuel pool. The modified design prevented the potential interference between a fuel element as it was being handled and one of the cross members on the bulkhead storage rack. A steel plate was added to the front of the rack to present a smooth surface to any fuel element which should contact it.

Supplement 1 to PCM 163-284 was implemented to replace two level switch support brackets and two temperature element support brackets which projected 18" from the spent fuel pool wall. The new design reduced the potential for interference between the brackets and fuel elements during handling by modifying the brackets so that they were no more than 11" from the pool wall.

Supplement 1 to the PCM introduced a new interference between the level switch supports and the refueling machine trolley above the supports. Consequently, Supplement 2 was implemented to lower the supports to provide additional clearance between the level switches and the trolley mechanism.

Implementation of Supplement 2 to the PCM introduced calibration and setpoint problems for the existing level switches which resulted in the loss of the fuel pool high level annuniciation function. The scope of Supplement 3 entails the restoration of the spent fuel pool high level detection capability. Implementation includes replacement of the two level switches (LS-4420 and LS-4421) with switches which are short enough in height to clear the refueling machine trolley and are capable of being calibrated to the high and low level setpoints. A new flanged spool piece will be fabricated and added to each support bracket to elevate the switches to the correct height for the fuel pool level.

SYSTEM DESCRIPTION

The redundant level switches are safety-related and meet Class 1E and Seismic Category I requirements. They will be calibrated for their annunciation functions at high and low level setpoints of 60.5 and 59.5 feet respectively. The flanged 4" spool pieces, shown on BCS-163-284.3001 R2 in sections C-C and D-D, rest on the lower flange of the support brackets. They are held in place by (8) 5/8 x 3" stainless steel hexagonal head nuts with bolts tack welded in place. The level switches will be located on top of the spool pieces with the instrument displacement devices suspended through the spool pieces into the spent fuel pool water. The level switches will also be held in place by (8) 5/8 x 3" stainless steel nuts and tack welded bolts. Existing wiring will be used to connect the switches to the annunciators.

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 replacement of two level switches and the addition of spool pieces to the level switch support brackets will restore the spent fuel pool high level alarm function without losing the low level alarm function and without sacrificing clearance between the instruments and the refueling machine trolley. The new level switches are safety-related and meet Seismic Category I and Class 1E requirements as demonstrated by Wyle Labs test report No. 43235-1 Rev A. The low level alarm function consists of redundant annunciation in the Control Room which completely meets the requirements of Technical Specification 3/4.9.11. The high level alarm function is provided solely to identify a high level condition to the operators and is not required by NRC regulation or for accident prevention as a result of FPL's implementation of PCM 052-283 "Transfer Canal Bulkhead Modification". In PCM 052-283, an opening was cut in the bulkhead door to allow drainage of spent fuel pool water in the unlikely event of an overfill.

The implementation of this PCM does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report since the level switches are not required for accident prevention or equipment protection. In addition, the possibility for creating an accident or malfuncion of a different type than any evaluated previously in the safety analysis report is not created since the function of the level switches has not been modified and the new switches are seismically qualified and mounted to ensure that the requirements applicable to Seismic Category I are met and are qualified Class 1E. Also, an analysis of the additional weight, due to the new spool pieces required for the level switch installation, shows that the modifications do not exceed the load limits allowed for the brackets.

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 any unreviewed safety question, therefore prior Commission approval is not required for implementation of this PCM.

HYDROGEN DETECTION INSIDE EXCITER HOUSING

SYSTEM DESCRIPTION

The Generator Exciter Housing Hydrogen Monitoring System, as added by this PC/M is a combustible gas detection system which includes:

1. One (1) combustible gas detector mounted on the top of the exciter housing toward the generator end,
2. One (1) combustible gas detector mounted on the top of the generator removable end cover.
3. Two (2) control/indicating modules located in the turbine building on the mezzanine floor level,
4. One (1) terminal box located inside the generator appearance skirting. A removable access cover to be installed on the appearance skirting in order to provide terminal box access,
5. Provisions for calibrating the sensors,
6. The design incorporates the easy removal of the sensor and sensor assembly from the turbine-generator exciter housing,
7. Electrical enclosure for the control modules, provided with a viewing window.

The Combustible Gas Detection System, is an instrument package specifically designed to continuously monitor for flammable gases and vapors and to activate a warning or alarm when predetermined concentration levels are reached.

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.

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

The modifications included in this PC/M affect only the turbine-generator and exciter. The hydrogen detection system provides an additional margin of safety by providing early warning indication of combustible gases. In addition, these components are all non-safety related and non-seismic.

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

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 implementation for this PC/M is not required.

480V BUSSES CV-2 UNDERVOLTAGE RELAY MODIFICATION

Description of Change

Dual contacts in the CV-2 relay could not be adjusted so that both operate at the same voltage.

This PC/M revises the wiring to the Westinghouse CV-2 relay for the 480V Busses 2A2/2B2 and 2A5/2B5 Undervoltage Protection System.

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 previouslyin 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 PC/M includes rewiring the contacts on the Westinghouse CV-2 undervoltage relay. No modifications to these Class 1E relays are being made. As a result of these modifications no change in the system operation results. The system operates identically to the previous design and only the wiring external to the CV-2 relay is revised.

No modification to the existing undervoltage protection scheme other than the CV-2 wiring are included in this PC/M.

This change alleviates maintenance procedures and ensures proper operation of these undervoltage relays.

This PC/M does not result in a change to the FSAR or Technical Specifications.

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

CONTAINMENT ANNULUS AIR SUPPLY

SYSTEM DESCRIPTION

Operation

The operation of the Instrument Air (IA) System is affected by this modification since the filter element needs replacement periodically at intervals recommended by this design document. The inlet, outlet and bypass valves added by this modification should be included in section 8.13 of Operating Procedure 2-1010022.

Function

The filter added by this modification will function to ensure that particulates do not interfere with the operation of valves inside the annulus which are supplied by the Instrument Air System. Elimination of these particulates will increase the reliability of the affected system loads, (i.e., those valves which must be cycled within Technical Specification limitations).

Design Description

This modification adds one particulate filter and its associated inlet, outlet and bypass valves to 3/4-IA-73 which supplies the annulus through penetration 62. This filter will be located near the penetration in the RAB.

SAFETY EVALUATION

This modification has been reviewed with respect to 10 CFR 50.59 and has been deemed not to involve any unreviewed safety question because of the following:

1.1 The portions of the IA System affected by this modification are not within the ASME Class III boundary and the components added by this modification do not perform a safety function. Therefore, this modification is classified as non-nuclear safety-related, Quality Group D.

1.2 These modifications do not–interact with any safety related systems or components.

1.3 No safety-related equipment or components are compromised by any assumed failure of existing or new equipment or components.

Therefore, failure of the filter will not increase the probability of an accident or malfunction of equipment important to safety previously evaluated.

1.4 No parameters relating to Technical Specifications are adversely affected and no Technical Specifications are altered.

2.0 Care has been taken from the design bases to system design phases to recognize and eliminate, mitigate or control all potential features which could be hazardous to the safety of equipment and/or personnel. This review constitutes, per 10 CFR 50.59, the safety evaluation; thereore this modification does not require prior Commission approved for implementation.

ICW SYSTEM ORIFICES

MODIFICATION DESCRIPTION

This PCM provides guidelines and details for replacing Intake Cooling Water (ICW) System orifices I-SO-21-1A and 1B (downstream of the Component Cooling Water Heat Exchangers) and SO-21-2A and 2B (downstream of the Turbine Cooling and Open Blowdown Cooling Water Heat Exchangers). The existing orifice plates are known to be deteriorated due to stress corrosion and/or flow erosion.

These orifices were installed early in plant life and are designed to stage the pressure drop downstream of the ICW system temperature control valves (TCVs) to reduce flow erosion (due to cavitation) of piping downstream of the TCVs.

This PCM provides details for new orifice plates, to be constructed of titanium.

SAFETY ANALYSIS

Regarding I-SO-21-1A and 1B:

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

The replacement of these orifices with new orifices, which have identical flow-pressure drop characteristics but are to be constructed of an upgraded material, will have no impact on the probability of accidents previously evaluated in the FSAR since no system design parameters or margins have been changed.

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

The consequences of accidents previously evaluated in the FSAR have not been made more serious since these orifices will produce the same flow-pressure drop characteristics as those they replace, and ICW system heat removal capability has not been reduced or altered.

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

Same as 1a

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

Same as 1a

2a. With respect to the possibility of an accident of a different type then previously evaluated in the FSAR:

There is no possibility of an accident of a different type than previously evaluated in the FSAR, since the modification only upgrades the material of the orifices, making them more reliable. Assuming failure (i.e., degradation) of an orifice, potential ICW flows would only increase, therefore, increasing containment heat removal capability.

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

Other than a straight replacement of the existing orifices with new orifices of an upgraded material, no new equipment is added by this PC/M. Additionally, no other existing equipment is modified by this package, therefore, there is no possibility of equipment malfunction of a different typed than previously evaluated in the FSAR.

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

No ICW system design parameters have been altered since the new orifices have the same flow-pressure drop characteristics as the existing orifices.

Based on the above arguments, it is concluded that no unreviewed safety question exists as defined by 10 CFR 50.59.

Regarding SO-21-2A and 2B:

The upgrade of material for these orifices is considered non-nuclear safety related for the following reasons:

A) These orifices are installed in a non-safety related portion of the ICW System.

B) Postulated failures of these orifices would have no impact on safe shutdown of the plant.

C) The orifices are not required to prevent postulated accidents, mitigate the consequences of such accidents, maintain safe shutdown conditions or adequately store spent fuel.

STEAM TRAP DRAIN PIPING AS-FAIL REPLACEMENT

Description of Change

Existing carbon steel fittings and piping have experienced several failures due to corrosion-erosion effects.

This PC/M provides guidelines for replacing fittings and piping in the steam trap-to condenser drain lines with upgraded materials (chrome-molybdenum) on an "as-fail" basis.

Safety Analysis

This modification is considered non-nuclear safety related for the following reasons:

A. the steam trap drains are non-safety related.

B. No postulated failures of any of the steam trap drains would have an impact on safe shutdown of the plant or safety related systems.

C. The steam trap drains are not used to prevent postulated accidents, mitigate the consequences of such accidents, maintain safe shutdown conditions or adequately store spent fuel.

FUEL TRANSFER TUBE SHIELDING

ABSTRACT

This Engineering Package (EP) details the requirements for the installation of additional concrete and lead shielding in the vicinity of the fuel transfer tube. The additional shielding is required to reduce personnel dose rates in the area during fuel transfer operations.

This modification is classified Safety Related because it involves an attachment to the containment vessel, which is Nuclear Safety Class 2.

This modification has been evaluated in accordance with 10CFR 50.59. The safety evaluation has shown that the implementation of this Engineering Package does not constitute an unreviewed safety question and prior Commission approval for its implementation is not required. This modification will have no effect on plant safety or operation.

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

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 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 Engineering Package provides for the installation of additional shielding in the vicinity of the fuel transfer tube to reduce personnel dose rates in the area. 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 will be performed in accordance with Safety Related requirements, hence the seismic capability of the existing structures in the area is not compromised. Therefore, there can be no impact on any adjacent 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 the modification will ensure that the additional shielding will have no interaction with safety related equipment and hence will have no effect on plant safety.

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

The implementation of this Engineering Package does not require a change to 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 Engineering Package is not required.

GE SAM RELAY PC CARD REPLACEMENT

System Description

Presently, St. Lucie Plant Unit #2 uses the SAM 11B relays for circuit breaker failure back-up protection schemes. The following is a list of their application at St. Lucie Plant, per reference 1.B..

St. Lucie Unit #2

Switchgear/Cubicle

6.9KV-2A1-01
6.9KV-2A1-02
6.9KV-2B1-04
6.9KV-2B1-05
4.16KV-2A2-01
4.16KV-2A2-02
4.16KV-2B2-09
4.16KV-2B2-10
4.16KV-2A4-1
4.16KV-2A4-5
4.16KV-2B4-1
4.16KV-2B4-5

This PC/M will replace the existing printed circuit board for the above relays with a new PCC #0165B1987G10 printed board. This will eliminate any time delay problems due to an unusual initiating contact bounce while maintaining the existing function.

Safety Analysis

This modification has been reviewed with respect to Title 10 of the Code of Federal Regulations, Part 50.59, which states that 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 modification being performed under this PC/M will enhance the operation of the G. E. Sam 11B relay assuring that if the unlikely event of an initiating contact bounce occurred, the relay will time out appropriately.

The G. E. Sam 11B relay affected are utilized for circuit breaker failure back-up protection schemes and are not in any safety related circuit or performed a safety related function.

Environmental qualification is justified by the fact that these relays and thus their internal PC cards are located in a mild environment.

There is no seismic concerns affected by this modification, the relays have no seismic requirements associated with them.

Therefore, the probability of a previously reviewed accident is not increased, the possibility of an accident of a different type has not been created and the margin of safety has not been reduced. The implementation of this PC/M does not require a change to the plant 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 for implementation of this PC/M is not necessary.

NEW FEED TO 480V POWER CENTER 2A5

SYSTEM DESCRIPTION

This PC/M provides the design details to add a new safety related "SA" 4.16kV breaker to feed the 480V Power Center 2A2, including control, indication and annunciation functions previously associated with the bifurcated feeder breaker. The additional 4.16kV breaker is being located in a new cubicle which is being added to the existing switchgear 2A3.

To allow this modification, the Isolation Panel IP-2E3, which is presently located at the end of 4.16kV switchgear 2A3, is being relocated to a new location.

The new 4.16kV switchgear cubicle complete with 1200A, 250MVA short circuit rating, 80kA momentary interrupting current capacity breaker and all appurtenances (relays, bus and miscellaneous devices) are being procured from Westinghouse. Indicating lights and control switch for installation on RTGB 201 are being procured from General Electric. Environmental and Seismic Qualifications for the above material have been provided by Westinghouse and General Electric, respectively.

Environmental and Seismic Qualification for the 4.16kV switchgear cubicle addition have been provided by Westinghouse via the following report: "Westinghouse Qualification Report to Florida Power & Light Company for DHP Medium Voltage Metal Clad Switchgear Cubicle Addition at St Lucie Plant - Unit 2", dated May 1986.

This report has been reviewed, found acceptable and entered into the EMDRAC system under drawing number 2998-18321.

Seismic analysis to determine the effect of the new loads on the existing seismic qualifications of RTGB 201 is being provided by Acton Environmental Testing Corporation.

Also as part of this package redundant fuses (per Appendix 'R' Requirements) will be added to the control circuits of the new breaker in the 4.16 kV Switchgear 2A3 Cubicle 1A.

SAFETY ANALYSIS

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 basis for any technical specification is reduced.

The 4.16kV breaker in Switchgear 2A3 feeding power centers 2A2 and 2A5 as well as the associated controls on RTGB 201 are all safety related train "SA".

The addition of a new safety related "SA" 4.16kV breaker to feed the 480V power center 2A2 including control, indication and annunciation functions on RTGB 201 previously associated with the bifurcated feeder breaker, constitutes an enhancement to safety. By installing the new 4.16kV breaker for feeding power center 2A2, total loss of the safety related 480V and 120V systems in case of a single fire in the "B" area, where the 2A5 power center is located, is being prevented.

The equipment required to implement this package includes a safety related 4.16kV switchgear cubicle, to be installed at the end of existing switchgear 2A3 and control switch/indicating devices to be installed on the RTGB 201. Isolation Panel IP-283 is being relocated to allow that the additional cubicle be attached to the existing 4.16kV Switchgear 2A3.

This implementation will also require the addition of redundant fuses (per Appendix 'R' Requirements) to the control circuit of the breaker in the 4.16 kV Switchgear 2A3. This scheme will permit continued control power to the breaker feeding 480V switchgear 2A2 upon isolation from the control room because of control/cable spreading room fire.

The switchgear cubicle addition has been environmentally and seismically qualified (Qualification Report No 2998-18321, Revision 1) for its installed location/configuration. The new conduit runs are seismically supported in accordance with the Flectrical Installation Notes and Details.

The addition of the new devices to the previously qualified RTGB and 4.16 kV switchgear 2A3 has been seismically evaluated with no significant impact on the dynamic characteristics of the RTGB and the 4.16 kV switchgear 2A3.

This modification does not require a revision to the 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; therefore, prior Commission approval is not required for implementation of this PC/M.

ICW PUMP EXPANSION JOINT REPLACEMENT

SYSTEM DESCRIPTION

Function

The intake cooling water pumps provide cooling water from the intake structure to the CCW heat exchangers and the TCW heat exchangers. Expansion joints between the pump nozzle and the piping system reduce stresses on the nozzle imposed by the movements of the piping system. Although they are not necessary to meet code design stresses, the expansion joints reduce vibration induced stresses and ease reassembly of the system.

Design Description

This PC/M changes the bellows material from Monel (ASME-SB-127) to Inconel 625 (ASME-SB-443). The liner material is changed fron 316 SS to Inconel 625. The attachment of the liner and bellows is changed from a welded design to a Van Stone flange design. These design changes are made to reduce the current rate of corrosion exhibited by the existing expansion joints. Inconel 625 is superior with respect to corrosion fatigue strength, pitting and crevice corrosion resistance when compared with Monel. The new expansion joints should have a significantly longer service life than the existing Monel expansion joints.

The new expansion joints shall be designed and fabricated in accordance with ASME Section III Class 3 requirements, except no N-Stamp is required.

SAFETY ANALYSIS

This change does not represent an unreviewed safety question since it does not affect any accident addressed in the FSAR, present any new accident not previously analyzed in the FSAR, nor does it affect the margin of safety for any technical specification.

The operation of the intake cooling water pumps or the piping system has not been affected by the use of an alternate material as specified in this PC/M package, as this alternate material is equal to or better than the original material in all aspects. Therefore, this material change does not increase the probabilities or consequences of accidents or equipment malfunction important to the safety of the plant previously evaluated in the FSAR.

PRESSURIZER MANWAY LIFTING LUG MODIFICATION

SYSTEM DESCRIPTION

FUNCTION

This modification functions to provide for removal of part of the pressurizer manway cover lifting lugs such that accessibility for the Kleiber and Schulz stud tensioning ring is provided.

DESCRIPTION

This modification provides for removal of a small portion from the end of each pressurizer manway lifting lug that presently interferes with the use of the Kleiber and Schulz stud tensioning ring.

The lug shall be modified by boring a new 5/8" hole located such that sufficient metal will exist on all sides. Also, this change provides for removing sufficient lug material to preclude usage of the existing lifting holes.

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 evaluation 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 provides for removing a small portion of the pressurizer manway cover lifting lug. The intent of this modification is to provide for removal of interference for the manway cover stud tensioning ring. Modification of this lug in no way affects the integrity or function of the manway cover and, therefore, does not increase the probability of occurrence or consequences of an accident or malfunction previously addressed in the Safety Analysis Report. Additionally, the modification does not affect or require a change in the Technical Specifications.

The forgoing 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 PCM.

REMOVAL TEMP S/U STEAM SUPPLY PIPING

ABSTRACT

This PCM package was developed to support the removal of temporary steam supply lines which were installed during the construction of St. Lucie Unit #2 and are no longer in use. The removal of these lines and the replacement of the existing eroded flange connections at the points where they tie into the steam lines is a non-nuclear safety related modification.

SAFETY ANALYSIS

Because the lines to be modified by this PCM (3-MS-35, 2½-MS-46, and 4-MS-62) are components that are not involved in the FSAR analysis of accidents the probability of occurrence of accidents previously addressed in the FSAR is unaffected, the consequences of the accidents addressed in the FSAR are unchanged, and the possibility of new accidents not considered in the FSAR is not increased. The three lines are not equipment that is important to safety, thus the modification does not affect the probability of malfunction of equipment important to safety previously evaluated in the FSAR, does not change the consequences of malfunction of equipment important to safety previously evaluated in the FSAR, and does not create the possibility of malfunctions of a different type than those analyzed in the FSAR. The three lines are not equipment that is considered in the bases of the Technical Specifications, so no margin of safety defined therein is affected by the modification. Failure modes evaluated as described in the design analysis also demonstrate that there is no potential interaction with safety related equipment or functions.

Based on the above discussion, it can therefore be concluded that the implementation of non-nuclear safety related PCM 102-285 will not create an unreviewed safety question.

PSB-1 UNDERVOLTAGE RELAY CABINET ENHANCEMENT

ABSTRACT

This Engineering Package (EP) modifies circuits and components in the PSB-1 Undervoltage Relay Cabinets to provide improvements to the cabinets as follows:

1) Replace existing potential transformers (PTs) with those with a center tap to evenly divide voltage in the event of umbalanced loads on the secondary windings.

2) Install test switches and indicating lights to facilitate periodic relay testing.

3) Modify existing ITE-27N undervoltage relays to correct operating anomalies. Brown Boveri letter to USNRC dated March 13, 1984 provided a bulletin regarding relay misoperation.

This EP is classified as Nuclear Safety Related since it provides for modification to Nuclear Safety Related Class 1E equipment.

The safety evaluation will be completed upon review and approval of all outstanding qualification documentation, after all HOLD POINTS have been lifted.

Supplement 1

This EP has been revised to lift all HOLD POINTS. National Technical Services - Acton Report No 23462-88N (EMDRAC No 2998-18510) has been reviewed and approved for seismic qualification of the potential transformers (PT-6S) as well as seismic qualification of the PSB-1 cabinets as modified by this PCM.

As a result, all HOLD POINTS have been lifted and the safety evaluation has been updated. The implementation of this PCM does not affect the Plant Technical Specifications and does not constitute an unreviewed safety question. Therefore, Commission approval is not required prior to implementation of this PCM.

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 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. This is confirmed by the following:

- This EP provides for evenly divided voltage in the event of imbalanced load via the new potential transformers with the center tap. This modification provides more accurate undervoltage sensing and reduces the chance of unanticipated trip; this aspect of the EP does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

- This EP provides for modifications to the ITE-27N definite time undervoltage relays in order to improve reliability and assure actuation in the event of a degraded grid voltage condition. This serves to decrease the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

- This EP provides for the installation of test switches and indicating lights in order to isolate the undervoltage relays for testing purposes wihout the possibility of inadvertent propagation of 4160V switchgear trip. This does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

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

- Replacement of the existing potential transformers with those with the center tap does not introduce the possibility of an accident or malfunction not previously evaluated in the FSAR since the primary and secondary circuits are in no way changed and this represents design configuration as originally evaluated in the FSAR.

- Modification of the ITE-27N undervoltage relays does not introduce the possiblity of an accident or malfunction of a different type than any previously evaluated in the FSAR since the two failure modes of the relays (fail trip and fail no-trip) have been evaluated in the FSAR.

- The installation of the test switches and indicating lights does not introduce the possiblity of an accident or malfunction of a different type than any previously evaluated in the FSAR since these are non-active, in-line components for which failure modes resulting in an accident or malfunction are not postulated.

(iii) These modifications do not change the margin of safety as defined in the basis for any technical specification since they have no negative effect on undervoltage protection and/or plant onsite AC power.

Since this EP affects equipment that is identified as Nuclear Safety Related (the PSB-1 Cabinets provide undervoltage protection for Class 1E buses), this package is considered Nuclear Safety Related. Seismic qualification of the modification of the PSB-1 Cabinets and the potential transformers have been reviewed and approved; the structural integrity of the PSB-1 cabinets will be maintained with the implementation of this PCM and the potential transformers have been qualified for Nuclear Safety Related service (see NTS-Acton Report No 23462-88N, EMDRAC No 2998-18510 and Attachment 7.8).

This EP involves equipment on the Essential Equipment List, but does not modify their intended operation or function. This package does not affect safe reactor shutdown or alternate shutdown. There are no other changes to equipment which involves 10CFR50 Appendix "R" fire protection (See Attachment 7.1).

Implementation of Nuclear Safety Related PCM 106-285 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.

2B CHARGING PUMP DISCHARGE RESTRAINT ADDITION

Introduction

Pulsation and vibration testing of 2B Charging Pump discharge line, conducted by FPL, indicated that line I-2"-CH-136 is experiencing excessive vibration when the pump is running. These vibrations have resulted in the need to make frequent repairs to this line.

System Description

The additional restraint proposed on BCS 138-285.3000, when implemented, will eliminate the undesirable vibrations which has caused the need for frequent repairs to the 2B charging pump discharge line.

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 additional restraint proposed by this PC/M, when implemented, will eliminate the undesirable vibrations experienced by the 2B Charging Pump discharge line and will provide additional restraining capabilities during the seismic event. Accordingly, it does <u>not</u> increase the probability of occurrence or the consequences of any previously analysed accident, nor does it create a new accident or reduce the margin of safety of any technical specifications.

The implementation of this PC/M 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 PC/M is not required.

LINEAR TRIPTEST POTENTIOMETER REPLACEMENT

ABSTRACT

This Engineering Package covers modifications in the Reactor Protective System Safety Channels. The PC/M will eliminate the combination pot/switches that are presently installed to perform functional testing of the linear trip function and replace them with high-resolution 10-turn potentiometers and separate toggle switches. The new components will ensure the required sensitivity to calibrate and perform testing.

The modifications are classified as nuclear safety related because the components being replaced are part of the Reactor Protective System.

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:

(a) 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.

(b) 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

(c) if the margin of safety as defined in the basis for any technical specification is reduced.

For the following reasons, C-E concludes this change does not involve any unresolved safety questions as defined in items (a), (b) or (c) above:

The changes to the Reactor Protective System described in this PC/M improve the testing and calibration characteristics of the RPS and decrease the probability of spurious indications and actuations. These changes do not adversly affect the functions, test/surveillance requirements or design of the RPS as described in the Technical Specifications. Therefore, Technical Specification revisions are not necessary.

The changes to the Reactor Protective System described in this PC/M do not affect the performance of its design safety function, nor are any other plant operations or design characteristics adversely affected. Therefore, the safety analysis transients are not affected and the consequences of accidents previously evaluated in the safety analysis report are not increased.

Also, since the RPS safety functions are not affected, since plant operation and design are not adversely affected, and since accidents previously evaluated in the safety analysis report are not assumed to be initiated by faults within the RPS, the probability of occurrence of accidents previously evaluated is not increased.

Additionally, the changes to the RPS do not adversely affect the performance, testing, calibration, or design features of the RPS, and therefore the possibility for a new kind of accident is not created.

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

CONDENSATE PUMP MINIMUM RECIRCULATION SYSTEM MODIFICATION

ABSTRACT

This Engineering Package (EP) is for the replacement of the existing 4 inch Condensate Pump Minimum Recirculation Flow Control Valves (FCV 12-3A, 3B and 3C) with 8 inch valves. It also adds an 8 inch manually operated isolation gate valve upstream of each of the new flow control valves and replaces the single stage restriction orifices in these lines with multistage orifices.

This EP is classified non-safety related, since the Condensate Pumps Minimum Recirculation lines, where this modification will be implemented, does not perform any safety function.

The safety analysis has correctly concluded that no unreviewed safety concern exist and no changes to the Technical Specifications are required as a result of this modification. Therefore, prior NRC approval for the implementation of this modification 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 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 is for the replacement of the existing 4 inch Condensate Pump Minimum Recirculation flow control valves with 8 inch flow control valves and for replacing and relocating the existing restriction orifices in the minimum recirculation lines. It also provides the installation of an 8 inch manually operated isolation gate valve upstream of each of the new flow control valves.

The portion of the Condensate System where this modification will be implemented does not perform any safety function. Accordingly, components in that portion of the Condensate System are classified non-safety class, Quality Group D; therefore this modification is not safety related.

Based on the above, this Engineering Package does not constitute an unreviewed safety question and the following are the basis 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 Systems where this modification will be implemented 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.

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 component involved in this modification are not included in the bases for 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.

CONDENSATE PUMP MINI-RECIRC PIPING

ABSTRACT

The 2B condensate pump mini recirculation line was found detached from the condenser at the nozzle weldment following the plant trip which occured in December 1984.

Examination of the restraint system for the mini recirc lines revealed that the condensate pump mini recirculation lines are supported for dead weight conditions only and that the condenser nozzles are not protected against the dynamic effects of vibration.

This Engineering Design Package provides engineering and design for additional vibration restraints in order to control the vibrations in the condensate pump minimum recirculation lines and to prevent future weld failure at the condenser nozzles.

The condensate system performs no safety related function. Accordingly, the system and its components, including pipe supports/restraints, are classified as non-nuclear safety related, quality group D and non-seismic.

This PCM does not constitute an unreviewed safety question and enhances the existing condensate pump system. The addition of vibration restraints to the condensate pump mini recirculation lines provides additional protection for condenser nozzle and does not affect any safety related equipment.

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

ADDENDUM 1

Supplement 1 provides designs for all support/restraint related additions and modifications needed to address all the changes in piping which includes routing change, removal, addition/replacement of new valves, etc, being proposed in PCM 149-285.

This supplement does not affect the safety evaluation that was performed for Rev. 0 of this PC/M and does not require a change to plant technical specifications.

This supplement has no affect 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 occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluted in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluted 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:

i Section 10.4.7 of the FSAR states that the condensate system is non-safety and non-seismic. The condensate system neither initiates nor mitigates any of the accidents analyzed in the FSAR, therefore this PC/M is non-safety related. The additional restraints provided in this modification will control vibration of the line and reduce the dynamic effect of vibration on the condenser nozzle without adversely affecting thermal flexibility. The pipe stresses are within the limits allowed in ANSI B 31.1 "Power Piping". Therefore, the implementation of this PC/M does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

ii The pipe stresses have remained within the code allowable limits. Integral attachments to the pipe, which could affect the pressure boundary of the piping, are not used for this modification. This modification does not create any possibility for an accident or malfunction of a different type than evaluated previously in the Safety Analysis Report (SAR)

iii Since the probability of failure induced by vibration is reduced, there is no decrease in the margin of safety at the condenser nozzle as calculated in the original design or as defined in the bases for technical specifications.

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 and prior Commission approval for the implementation of this PCM is not required.

REACTOR HEAD VENT LINE RESTRAINT MODIFICATION

INTRODUCTION

Restraint RC-98-R2, which is welded to the Reactor Coolant Gas Vent piping, has to be removed with the pipe during each refueling outage. This restraint also interferes with the temporary Reactor Head shielding curtain. Due to the size and weight of the restraint, the removal requires special rigging and handling to prevent bending of the small vent line. In its present configuration the restraint is removable only from the inside of the reactor cooling shroud.

SYSTEM DESCRIPTION

The modification proposed on BCS 163-285.3000, when implemented, will facilitate easy removal of restraint RC-98-R2 from outside the Reactor Cooling Shroud and will also reduce the size and weight of the restraint being removed. This is accomplished by reversing the bolts which attach the restraint to the shroud and by providing a flange type connection near the pipe-end of the restraint.

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.

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

This modification will facilitate easy removal of restraint RC-98-R2 from outside the Reactor Cooling Shroud and will reduce the size and weight of the restraint, which currently requires special rigging and handling for removal. This is accomplished by reversing the bolts, which attach the restraint to the shroud and by providing a flange type connection near the pipe-end of the restaint.

The addition of a flange type connection to the restraint and reversing the bolts does not alter the original design configuration/function nor does it reduce the safety factor calculated during the original design.

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

The foregoing constitutes, per 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.

ANALOG DISPLAY SYSTEM GRAPHIC DISPLAY SPARES

ABSTRACT

This engineering design package covers the replacement of the ADS Video display monitor in the RTG Board Section 204 with Ramtek Model GM-721. The existing video display monitor in the RTGB is a Conrac Model 5211 which is no longer manufactured. The Analog Display System monitors the vertical positions and movements of the 91 Control Element Assemblies (CEA's), utilizing the signals from reed switch position transmitters. The CRT in the Control Room provides the operator with one of the two continuous video graphic displays for the CEA positions. The CEA position system is Non-Safety Related (see FSAR Section 7.5.1). However, the associated mounting assembly in the RTGB must be seismically qualified, mandating this PCM to be 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 initiated without prior commission approval.

SUPPLEMENT 1

This EP revision provides for changes to the seismic CRT housing which will be mounted in the Reactor Turbine Generator Board, and which will contain both the ADS CRT monitor and one ERDADS/SAS CRT monitor. The CRT housing changes are necessary to allow the ERDADS/SAS CRT monitor to fit into the housing.

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 initiated 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 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 described in this PC/M replaces existing CRT monitor associated with the Analog Display System. The vertical position and movement of the 91 Control Element Assemblies (CEA's) are graphically displayed on the CRT. A CEA backup display panel associated with the ADS is also available for operator's use. No modification to the system is initiated by this PC/M since it utilizes a one for one CRT replacement.

The failure of this component to function would not affect the safe shutdown of the unit since it is not required to shutdown the reactor, cool the core, or cool another safety system in the reactor containment (after an accident); nor is it part of any system that reduces radioactive release during an accident. The housing is required to withstand loadings induced by the design basis earthquake. Therefore, this PC/M is classified "Quality Related".

The modifications to the RTGB-204 is anlayzed as to maintain the seismic integrity of the equipment.

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 Nuclear Regulatory Commission approval for the implementation of this PCM is not required.

CCW BACKFLUSH STRAINER DRAIN

Abstract

This engineering design package (EDP) modifies the CCW Strainer Backflush Drain piping. Existing cast iron and fiberglass drain piping, which is routed to the CCW sump, will be replaced with stainless steel piping which ties into the ICW discharge line. This will eliminate the flooding problem in the CCW pit area, which is causing corrosion of structural steel and piping supports mounted on or near the floor.

This EDP is classified as nuclear safety related since it modifies a safety related system. The safety evaluation has shown that this EDP does not constitute an unreviewed safety question.

This EDP has no impact on plant safety and operation.

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 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 modification included in this engineering design 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 connection of a CCW strainer backflush drain line to the ICW discharge line will have no effect on the safety performance of the ICW or CCW systems or any of their components.

(ii) There is no possiblity for an accident or malfunction of a different type than any previously evaluted since no changes have been made to the operational design of the CCW strainer backflush system.

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

Implementation of this engineering design package 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.

RTD AND THERMOWELL REPLACEMENT FOR REACTOR COOLANT SYSTEM

ABSTRACT

This Engineering Package (EP) provides for the removal and replacement of the existing resistance temperature devices (RTDs) and thermowells with spring loaded tapered RTDs and tapered thermowells. The purpose of this change is to alleviate difficulties experienced in the maintenance of the existing equipment (e.g. prying of RTD from thermowell and thermowell damage sufficient to necessitate its replacement).

This EP is classified as Nuclear Safety Related since it involves the reactor coolant pressure boundary and components which are part of the Reactor Protection System (RPS). This EP also involves components (RTDs) which are identified as post accident monitoring instrumentation (PAMI) and provide control room indication and recording. A review of the changes to be implemented by this PCM was performed against 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 or the proposed Revised 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.

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:

(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 replacement RTD/thermowells meet the requirements of the FSAR. The replacement RTDs provide the same input as the existing equipment and do not alter the function of any of the components, cabinets, or systems that receive RTD input.

(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.

Installation of the RTD/thermowell assemblies are controlled by site procedures and the FPL Welding Control Manual. Welding of the thermowells to the sleeve is to be in accordance with FPL General Welding Standard for Nuclear Piping and Piping Components, Rev 1. The RTD/thermowell assemblies have been subjected to non-destructive examination (NDE) per the codes and standards listed in Section 2.2.2 of this EP. By adhering to these codes and standards in the implemention of this PCM, there is no possibility for an accident or malfunction different than any previously evaluated involving the Primary Coolant Pressure Boundary. The replacement RTD/thermowells have approximately the same weight as those being replaced. Therefore the insignificant change in weight does not have any affect on the pipe stress and/or the support restraints.

(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 the replacement items meet the same Technical Specification limitations as the existing items and the fact that the design limitations of the reactor coolant pressure boundary, as delineated in FSAR Section 5.1, are maintained with the implementation of this PCM.

Since this EP affects equipment that is identified as Nuclear Safety Related (the RTDs are class 1E; the thermowells are ASME Class 1), this package is considered Nuclear Safety Related.

No hydrostatic pressure test is required after the installation of the RTD/thermowell assemblies per ASME Section XI, Paragraph IWA-4400. An in-service leak test will be performed to ascertain that the implementation of this PCM has met the requirement of no allowable leakage of reactor coolant.

The only effect this EP has on cables essential to safe reactor shutdown and alternate shutdown components is in the disconnection of the existing RTDs and the reconnection of the new RTDs. There are no other changes to equipment which involves 10CFR50 Appendix "R" fire protection (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 and prior Commission approval for the implementation of this PCM is not required.

MAKEUP·AIR FOR CONTAINMENT HYDRO PURGE SYSTEM
TEMPORARY VALVE MODIFICAITON

INTRODUCTION

The continuous containment purge/hydrogen purge system is designed to: provide a sufficiently low concentration of radionuclides in the containment atmosphere, relieving of containment pressure buildup, the capability of ensuring that the containment source term contribution to the annual average off-site doses are maintained as low as is reasonably achievable and hydrogen removal capability.

This system provides a direct air path between the containment atmosphere and the outside. Leak rate testing is required for penetration. During the leak rate testing, it was found that the isolation valve (FCV-25-36) was leaking. This PCM is for the temporary modification to remedy these leaks.

SYSTEM DESCRIPTION

The system consists of a purge make-up penetration line and exhaust penetration line. These containment penetrations provide a direct air path between the containment atmosphere and the outside. The isolation valves have been provided for both air path and isolation.

During the leak rate test it was determined that valve FCV-25-36 was leaking. In order to complete the test successfully the valve was modified by bolting a 1" thick plate to one end of the valve. This modification is considered temporary until the valve is replaced or repaired and the system is restored to its normal operating conditions.

The use of a 1" plate is acceptable for a blind flange for FCV 25-36 for the following reasons:

The required thickness for a 150# flange for this size is 1-1/8". The required pressure for this application is 44 psi versus an allowable pressure of 150 psi for 1-1/8". By engineering judgement the reduction of 1/8" will not affect the ability of this flange to withstand 44 psi. This plate will meet all the other requirements, including documentation, for ASME Section III Class 2 material.

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 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 use of a blind flange does not create any new accident since the valve is used as Containment Isolation and its safety function is to isolate the containment. The blind flange will perform this function passively. The blind flange will be subjected to the same testing requirements as the valves in that system.

This system is used as a non-safety backup to the redundant safety related hydrogen recombiners which maintain the hydrogen concentration below 4% after any accident. This system is non-safety and does not need to meet single failure criteria since the hydrogen recombiners are the design basis for the plant. This system is not considered in the design basis, therefore the loss of function of this system does not affect the ability of the plant to mitigate an accident.

The only Tech Specs involved are Containment Isolation and Containment Pressure. This modification increases the margin of safety of the Containment Isolation Valve Technical Specification since it replaces an active device with a passive device. With regards to the Containment Pressure (normal) Technical Specification, this Technical Specification is unaffected since the mini purge exhaust line will still function to reduce pressure inside containment.

The other design basis of the Continuous Containment Hydrogen Purge System will be affected. However, their impact will be in the form of longer duration of plant outages and in no way impair the safe operation of the plant. The longer plant outages will be the result of the inability to purge the containment during operation. Thus purging must be accomplished during shutdown.

Therefore this modification does not constitute any unreviewed safety question.

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 evaulation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementaion of this PCM is not required.

ATMOSPHERIC DUMP VALVES - MOTOR OPERATED VALVE INDICATION

ABSTRACT

This Engineering Package covers modification to the control circuitry of motor operated valves (MOV) MV-08-18 A/B and MV-08-19 A/B. This involves the utilization of the motor operator's #12 limit switch to provide a limit switch controlled back-up to the primary method of closure-torque switch control.

These valves function as atmospheric steam dump valves (ASDV) which are used to relieve/control system pressure. The ASDV's are part of the main steam supply system. The FSAR, in section 10.3.2 classifies these valves as safety related. This modification is considered safety related and deemed not to constitute an unreviewed safety question.

SAFETY EVALUATION

The function of the atmospheric steam dump valve system is to provide reactor coolant system heat removal capability. The modification is directed only to valve position indication circuitry to provide more reliable position indication while eliminating rotor adjustment difficulty and does not affect valve power circuits.

The proposed design ensures that ASDV's will be closed with the torque switch with a limit back-up closure switch. The change proposed to the ASD's MOV does not introduce any change to the functional configuration of the system required to meet the safety related design function. This design does not alter the original requirements specified in the St. Lucie Unit 2 FSAR, Section 10.3, which specified that dump valves are designed to withstand Design Basis Earthquake (DBE) loads simultaneously with the effects of the discharge thrust of steam passing through them, the effect of dead weight, and the effects of internal pressure loads. The FSAR required that these valves shall be powered from a (DC) onsite power source. The proposed modification does not change this feature (FSAR, Section 10.3.3).

The use of switch #12 on rotor #3 of the existing MOV will have no adverse affect on nuclear safety since this modification will not adversely affect the limit backup and torque closure limit circuitry. This modification eliminates the adjustment difficulties of the limit back-up and position indication switches, thus enhancing the circuit overall operability requirements.

All modifications will be made within the motor operated valves. The circuit modification is strictly a hardware modification to exchange the rotor contact presently used for spare contacts on a spare rotor, to aid in contact adjustment. The modified circuit will function as the circuit previous to the modification, and in that no external circuit or cable routing will be required, no Appendix R analysis will be affected.

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. The design change does not alter equipment circuitry used to mitigate accidents. The change

allows proper valve torque closure with limit back-up and appropriate indication. Therefore, the probability of occurrence of analyzed accidents remains unaffected. The margin of safety as defined in the Technical Specification has not been reduced because the atmospheric steam dump valves system operability has not been affected. The capability remains to provide reactor coolant system heat removal and withstand design basis earthquake loads simultaneously with the effects of the discharge thrust of steam passing through them, the effect of dead weight, and the effects of internal pressure loads. Based on the above evaluation and information in the design analysis it can be demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

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 NRC approval and issuance of Technical Specification changes prior to implementation.

DIESEL GENERATOR GOVERNOR POWER SUPPLY

INTRODUCTION

A Lambda power supply was installed under PCM 389-283 in order to increase reliability of the emergency diesel generators. This power supply has been unable to maintain the necessary regulation required by the governor. At present, this power supply has been bypassed.

As presently installed, the governor power supply is derived from plant 125V DC power. This condition is similiar as prior to the implementation of PCM 399-283. It has been demonstrated during testing that any disturbance on the plant 125V DC system will cause erratic governor operation. This erratic operation produces unstable generator electrical output.

The DC governor power supplies installed in St Lucie Unit 1, are Woodward power supplies and have been performing satisfactorily. These power supplies were installed under PCM 372-183. Based on this performance, a modification replacing the Unit 2 Lambda power supply with a Woodward power supply will be implemented by PCM 011-286 supplement 1.

SYSTEM DESCRIPTION

The purpose of the modifications performed by this PCM is to increase the stability and reliability of the Emergency Diesel Generator by installing Woodward power supplies Part No 9903-034 which is similar to that used in St Lucie Unit 1 Diesel Generators.

The existing plant 125V DC power will be used to start the diesel and accelerate it to full rated speed. At this speed (850 RPM), the speed switch will actuate to disconnect the 125V DC plant power and connect the new power supply into the governor circuit. Thus, the governor, which is sensitive to power supply disturbances, will be isolated from the plant system.

The new power supply circuit is also being modified to include contacts which will maintain the voltage regulator in a de-energized state during normal plant operation.

SAFETY ANALYSIS

This modification has been reviewed with respect to Title 10 of the Code of Federal Regulations, Part 50.59, which states that 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 maybe increased; or (ii) if a possiblity 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 possibility of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis is not increased by this PCM supplement. Since the diesel generators will be tested for acceptance in accordance with the Plant Technical specifications for periodic testing 4.8.1.1.2.3, which verify DG performance in a simulated loss of offsite power.

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

a. The installation of the Woodward power supply will minimize the erratic operation of the DG governor due to any disturbances in the 125 VDC power supply.

b. The Woodward power supply has been seismically and environmentally qualified. The environmental qualification evaluation is attached to this PCM supplement.

c. The margin of safety as defined in the basis of the technical specification is not reduced since as previously discussed the operability of the Diesel Generator will be confirmed by the periodic testing discussed above.

d. The power supply mounting duplicates the mounting used for seismic testing performed by NTS Acton Labs.

e. The type of power supply, Woodward Model No 9903-034 is a vendor received power supply to be used with 2301 series electrical governor system.

There is no change on the existing technical specification due to the implementation of this PCM supplement.

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

TARGET ROCK VALVES - STEM ASSEMBLY UPGRADE

ABSTRACT

The Target Rock Valves described herein are model 75C-002, 2" motor operated globe isolation/throttling valves which are installed in the Safety Injection System. Combustion Engineering infobulletin 84-10 provided information regarding the potential of the stem assembly parts galling under long term throttling duty. The Infobulletin discussed possible solutions to the potential problem, which consisted of upgrading the materials of the stem assembly subcomponents.

This PC/M provides the information required to upgrade the eight affected Target-Rock valves on St. Lucie Unit 2 with manufacturer-redesigned stem assemblies constructed of galling resistant materials

The modification described herein is classified as Nuclear Safety Related. No unreviewed safety questions exist as defined by 10 CFR 50.59, therefore commission approval is not required prior to implementation.

Safety Evaluation

This modification involves only the upgrade of materials for the stem assemblies of the High Pressure Safety Injection pump motor operated isolation valves.

10 CFR 50.59 allows changes to a facility described in the FSAR without prior NRC approval if an unreviewed safety question does not exist and if a change to technical specifications is not involved. The following arguments demonstrate that an unreviewed safety question does not exist:

i) The probability of occurrence of a design basis accident is not increased since this modification does not alter existing Safety Injection System operation, design parameters, and since no new equipment is added. Additionally, the new stem assemblies are to be designed, fabricated and inspected to the same code criteria as the existing stems.

ii) The consequences of accidents previously evaluated in the FSAR are not made more serious for the reason provided in Paragraph (i) above.

iii) The possibility of an accident of a different type than any previously postulated in the FSAR is not created for the same reason provided in Paragraph i above.

iv) The margin of safety as defined in the basis for any technical specification is not reduced since Safety Injection System parameters will not be affected by the material change, and since the manufacturers design will meet the original specification requirements.

Since the above arguments demonstrate that an unreviewed safety question does not exist, and since no changes to technical specifications are involved, the modifications to the affected safety injection system isolation valves do not require prior NRC approval.

RDF-RTD TEMPERATURE TRANSMITTER REPLACEMENT

ABSTRACT

This engineering package covers the replacement of four (4) RdF temperature transmitters. The presently installed transmitters are no longer being manufactured and suitable replacements are being provided for maintenance and replacement capability. This engineering design package is considered Quality Related since the replacement temperature devices are being seismically mounted on the RTGB. The instrumentation loops associated with the transmitters are not used to mitigate incidents and accidents and ,therefore, as per FUSAR Chapter 15, this PC/M is not considered to be 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; (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 described in this PC/M is associated with instrumentation loops used for reactor reactivity control, input to data processor and Safety Assessment System, control room indication/recording and annunciation. As per FUSAR Section 7.7, this instrumentation and control system is not essential for the safety of the plant.

The new temperature transmitters are being seismically mounted to RTGB-203. These new transmitters have been addressed by Acton Labs as to the seismic impact on the RTGB. As per Acton Labs letter Att. 7.3 the replacement transmitters will have no impact on the equipment seismic qualification and the dynamic characteristic of the equipment will not be affected.

This modification is a one for one replacement of temperature transmitters only and does not alter or change the original transmitter loop arrangement, as such the implementation of this PC/M does not require a change to 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."

CCW PUMP BEARING MATERIAL CHANGE

ABSTRACT

This engineering package covers replacement of the existing cast iron journal bearing shells on the component cooling water pumps 2A, 2B & 2C 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 2A, 2B and 2C 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 probabilty of occurence 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.

PCB TRANSFORMER REPLACEMENT

ABSTRACT

Due to environmental concerns attendant to polychlorinated biphenyl (PCB) cooling/insulating liquids, all transformers filled with a PCB liquid are being eliminated from FP&L's system. The neutral grounding transformer for the main turbine generator is filled with 38 gallons of PCB cooling/insulating liquid. The neutral grounding transformer for each emergency diesel generator (EDG 2A and EDG 2B) is filled with 15 gallons of PCB cooling/insulating liquid. This Engineering Package provides for replacement of the three (3) generator neutral grounding transformers with equivalent silicone liquid-filled or dry-type transformers.

The main generator neutral grounding transformer does not perform any nuclear safety related function, therefore its replacement is classified as non-nuclear safety related.
Due to their association with the safety related emergency diesel generators, the replacement neutral grounding transformers for the emergency diesel generators are classified as nuclear safety related.
The implementation of this PC/M will not have an adverse impact on plant safety or operations.

SUPPLEMENFT 1

This supplement incorporates vendor and installation drawings, seismic report, associated engineering design calculation certification, design analysis and safety evaluation, design and safety verification and Total Equipment Data Base (TEDB) sheets.
All "On Hold" and "Later" statements affecting the engineering package sections above under Revision 0 are being removed by this supplement.

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.

SUPPLEMENT 2

This supplement incorporates Change Request Notice Nos. 038-286.343 and 038-286.386 which modify the emergency diesel generator neutral grounding transformers' terminal numbers and the wiring to the coil of the ground protection relay, respectively. It also addresses wind loading, electrical clearances and justification for using the 600 volt rated jumper for the emergency diesel generator neutral grounding transformers. In addition, corrections were made to the drawing list to reflect Unit 2 drawing numbers and to Section 1.3.1.2 to reflect revision 1 of Ebasco Specification FLO-E-002. The safety evaluation has been revised to incorporate the wiring modification. This revision, however, has not altered the previous conclusion which indicates that the modifications presented by this Engineering Package do not constitute an unreviewed safety question, do not require any changes to the plant Technical Specification and do not require prior Commission approval for the implementation of this PC/M.

SAFETY EVALUATION

The replacement neutral grounding transformers for EDC 2A and 2B are located inside their respective control cabinets. The replacement transformers have been seismically and environmentally qualified. A flexible connection is provided at the H1 terminal lug of the replacement transformers to minimize stress on the lug under seismic conditions. Ground detection relay coil wiring has been revised in accordance with General Electric Power systems Management Department instructions GEH-1814B. This does not affect the operability of the relay for diesel generator ground detection since the circuit has not been functionally modified. The existing seismic qualification of the control cabinets has not been affected by the replacement transformers.

Based on the preceeding, 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 form, fit and function.

(ii) This modification does not change the operation of the Main Generator or the Emergency Diesel Generators, 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 neutral grounding transformers are equivalent in form, fit 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 Commission approval for the implementation of this PC/M is not required.

QUENCH TANK PMW ISOLATION VALVE REPLACEMENT

INTRODUCTION

The quench tank is part of the pressurizer pressure control system. Excessive pressure in the pressurizer is relieved by discharging steam to the quench tank. The steam is condensed in the quench tank by partially filling it with primary makeup water.

Water is supplied to the hose stations inside containment by the primary makeup water system. A 1 inch solenoid valve isolates the PMW to the quench tank from the hose station supply line. The small size of this valve prevents timely makeup of water addition to the tank. This valve is being replaced with a larger valve to eliminate this problem.

SYSTEM DESCRIPTION

The quench tank and pressurizer relief discharge system are described in Section 5.4.11 of the FSAR. This modification will replace the 1 inch solenoid operated valve (SE-15-2) in line 2-RC-507, with a 2 inch air operated ball valve. The new valve and operator are lighter than the original valve, therefore no additional support/restraints will be required. The air for the operator will be supplied by tying into the air supply for valve V3632. A piston type, spring return operator will be used which will close the valve on loss of instrument air.

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 anslysis 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 modifications included in this PCM 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 evluated is not increased since the quench tank PMW isolation valve is non-safety related and this modification will have no effect on equipment performing a safety function. This modification will decrease the probability of overpressurizing the quench tank, since the new valve will supply makeup water to the quench tank with a substantially greater flow rate.

ii There is no possibility for an accident or malfunction of a different type than any previously evaluated since the quench tank PMW isolation valve has no safety function and no changes have been made to the operational design of the system.

iii This modification does not change the margin of safety as defined in the basis for 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.

ADDITION OF FLANGE TO PENETRATION P-50

ABSTRACT

This Engineering package provides for the replacement of the welded pipe cap on penetration P-50 with a blind flange. The pipe cap is on the outboard side of the concrete shield building. This modification is nuclear safety related, because it deals with a change to the structural loads of the Containment Shield Building which is a Seismic Category I structure. This modification does not affect the Containment pressure boundary. Since this modification is not to a piping system, a Quality Level designation is not applicable. The additional loads are small and do not change the seismic classification of the penetration or Containment. This modification does not constitute an unreviewed safety question as defined by 10 CFR 50.59. As a result of this modification, penetration P-50 can be readily used to support outage related tasks such as eddy current testing.

SAFETY EVALUATION

The subject modification provides for replacement of the pipe cap on the outboard side of Containment Shield Building penetration P-50 with a weld-neck flange, gasket and blind flange. As defined in Section 3 of the FSAR, the Containment Shield Building is Seismic Category I. This modification is considered nuclear safety related because it alters a Seismic Category I structure. Since this modification does not change a piping system, a Quality Level designation is not applicable.

Per the attached Ebasco letter, this modification does not alter the seismic qualification of the Containment Shield Building or penetration P-50. Also, the containment leak rate is not affected because the outboard cap does not form part of the containment pressure boundary. Even so, the seal integrity of the flange replacing the cap is verified by NDE testing and Quality Control verification of flange bolt torquing.

No active components or other safety related systems and/or compoents are impacted by this modification. Accidents considered in Section 6 of the FSAR bound any abnormal condition that could be caused by failure of the new penetration flange. No Technical Specification is impacted by replacing the outboard shield building penetration cap with a flange.

Based on the above evaluation, an unreviewed safety question as defined by 10CFR 50.59 does not exist: (i.e., the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. The possibility of an accident or malfunction of a different type than any evaluated in the FSAR is not changed. The margin of safety as defined in the Technical Specifications is not reduced.) The preceding argument, coupled with the fact that a Technical Specification is not required, leads to the conclusion that prior NRC approval is not required to implement this modification.

RCP INSULATION REPLACEMENT

ABSTRACT

This Engineering Package provides details for replacing the existing blanket type insulation on the Reactor Coolant Pumps with reflective type insulation developed by Diamond Power Specialty Company.

The insulation around the pumps is Quality Related because it must remain in place at all times during plant operation and it will be seismically supported. In addition, the metal reflective design has accounted for effects on the containment recirculation system and sump screen blockage. The insulation design has accounted for potential impact on overall containment heat load to insure that containment ambient temperatures will not increase as a result of this modification.

The safety evaluation has shown that this EP does not constitute any unreviewed safety question, has no adverse effect on plant safety nor does it require a Technical Specification change.

Implementation of this modification is acceptable 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 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 involves the replacement of the blanket type insulation around the reactor coolant pumps with stainless steel reflective type insulation. As discussed in the Design Bases and Design Analysis, this modification is considered Quality related due to the seismic design considerations, the potential impact on the containment recirculation system and sump design and the potential impact on containment ambient temperatures. Based on the failure modes evaluation the insulation added by this modification will not adversely effect any safety related equipment or components. Based on this and information provided in the Design Analysis; this 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 is not increased since the stainless steel encapsulated blanket insulation is being replaced by stainless steel reflective insulation. Both types of insulation presently exist inside containment. Since the stainless steel reflective insulation is similar to other reflective insulation used, is equal to or better in insulating quality to that which it replaces and is seismically supported, it will have no effect on equipment or functions important to 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 replacement of one type of insulation with another type of insulation, both acceptable for use inside containment, does not change any existing Design Criteria, Operating Procedure or Technical Specification.

(iii) This modification does not affect the basis for any Technical Specification and therefore does not reduce the margin of safety as defined in the basis for any Technical Specification.

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

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

TORQUE SEATING - ATMOSPHERIC DUMP VALVES

ABSTRACT

This Engineering Package covers a modification to the Unit 2 Atmospheric Dump Valves (MV-08-18A, MV-08-18B, MV-08-19A and MV-08-19B). This modification changes the control circuit for these valves to allow the closing direction to be limited by the torque switch instead of by the limit switch as presently designed. This PCM is classified as Nuclear Safety Related, and does not constitute an unreviewed safety question.

Safety Evaluation

The atmospheric dump valves (ADV's) provide a means of decay heat removal and cooldown capability when the MSIV's are closed. The ADV's can also be modulated to control primary plant temperature during startup and shutdown. The valve manufacturer, CCI, was consulted and concurs that these valves may be torque seated. The proposed change is therefore acceptable for all four ADV valves. The proposed design, although different than the original, does not change the operation of the atmospheric dump system as discussed in the PSL Unit 2 FSAR Sections 5.4, 6.3, and 10.3. Since the ability of the ADV's to close has not been adversely affected by this change, the probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to safety as discussed in the FSAR Chapter 15 has not been increased. Previously analyzed failure modes for the ADV's remain valid, and thus the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR Chapter 15 is not created. PSL 2 Technical Specificiations Section 3.7.1.7 provides the "Limiting Condition for Operation" for the ADV's. The Technical Specification requirements are not changed by this modification and the margin of safety as defined in the bases for this Technical Specificiation will not be reduced. The safety evaluation thus demonstrates that an unreviewed safety question does not exist.

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 NRC approval prior to implementation.

HEATER DRAIN PUMP MECHANICAL SEAL DEMINERALIZED WATER SUPPLY

ABSTRACT

This design package provides the necessary engineering for adding permanent piping from the demineralized water system to the Unit 2 heater drain pumps' mechanical seals. The piping will make available to the seals the necessary back up flushing water meeting the appropriate chemistry requirements. This backup flushing water 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.

- o 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.

- o 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.

- o 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.

MISAPPLICATION OF LIMITORQUE OPERATORS

ABSTRACT

This Engineering Package (EP) is for the replacement of the motors and gear trains on the following valve motor operators:

Valve Tag No	Location	Parts to be Replaced
I-MV-09-1	FW Pump 2A Discharge	Motor & Gear Train
I-MV-09-2	FW Pump 2B Discharge	Motor & Gear Train
MV-09-3	FW Flow Control Station (Train A)	Motor
MV-09-4	FW Flow Control Station (Train B)	Motor

The replacement of the existing motors with motors having lower RPM and increasing the operator gear train ratio in two of these valves is required to reduce the valve stem speed, to be within the limits recommended by the valve operator manufacturer (Limitorque) for the type of operator (SMB) involved.

This EP is classified non-safety related since the portions of the main feedwater pump discharge piping and flow control stations where the affected valves are installed, does not perform any safety function and they are in the non-safety class portion of the Main Feedwater System.

The safety analysis has correctly concluded that no unreviewed safety concern exist and no changes to the Technical Specifications are required as a result of this modification. Therefore, prior NRC approval for the implementation of this modification is not required.

This EP has no impact on plant safety and/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 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 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 (Section 15.1.2.1) is not increased. The portions of the feedwater system where this modification will be implemented are not considered 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 possiblity 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 component 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 necessary.

CLOSE INTERCEPT VALVE - CONTROL CIRCUIT MODIFICATION

ABSTRACT

This Engineering Design Package (EDP) provides for the removal of the Close Intercept Valve (CIV) anticipatory control circuit from the Westinghouse Digital Electro-hydraulic (DEH) turbine control system.

The original intent of the CIV anticipatory circuit was to provide a temporary closure of the Interceptor Valves in the event of a load mismatch between turbine steam flow and generated electrical output.

This particular circuit does not take into account the dynamic response of the turbine steam cycles, nor does the DEH model P-2000 contain the necessary programming software to perform the required calculations to automatically adjust the turbine governor valves to the new thermodynamic values.

These features, therefore, will, in most cases, maintain the Interceptor. Valves closed with a resultant trip of both the turbine and the reactor.

The CIV control circuit is a downstream extension of the DEH overspeed control channel. System failure would not impact plant safety, since this system is neither required for safe shutdown nor does it perform any safety related functions. However the DEH Control System is required to be operable by the Technical Specifications. Since this modification impacts the subject control circuit, this engineering design package shall be 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 PC/M, this PC/M 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 EDP.

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 of 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 creatd; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The probability of occurrence as the consequences of an accident or malfunction of equipment previously evaluated in the Safety Analysis Report is not increased by this PC/M. This modification to the CIV control circuit does not change or alter the turbine-generator monitoring and control system.

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

- The CIV control circuit is an independent function generated by the DEH control system software.

- The removal of the CIV anticipatory function does not alter the operation of the DEH control system.

- This modification, which will remove the partial load mismatch circuit, will reduce the number of spurious reactor trips which will occur should the Interceptor Valves fail to re-open.

- The turbine overspeed protection channels to both the Reheater Stop valves and the Intercept valves and the mechanical overspeed protection channels are not altered by implementation of this circuit modification. Therefore, the margin of safety for turbine rupture due to the probability of turbine overspeed is not reduced.

The implementation of this PC/M does not require a change to the St Lucie Unit 2 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."

ADDITIONAL APPENDIX R FIRE SPRINKLER AND FIRE WRAP IN RAB

ABSTRACT

An on site inspection by NRC I&E Inspectors identified that in some fire areas in the RAB, existing ceiling level sprinklers are obstructed by cable trays, HVAC ducts, etc. This condition does not provide adequate fire protection to the conduits located below these obstructions. In order to ensure compliance with Appendix "R" requirements and to provide adequate fire protection to the affected conduits, modifications to the existing sprinkler system are required.

This Engineering Design Package (EDP) provides the engineering and design for the addition of new sprinklers below obstructions, isolation for two signal transmitters in the Hot Shutdown Control Panel and revision of the Safe Shutdown Analysis to remove protection requirements from several cables.

This EDP is classified as nuclear safety related since the inputs to the isolated signal transmitters are accepted from safety class equipment and the devices (isolated signal transmitters) have to be qualified as Class IE per IEEE-Section 323 (1974). Changes to the sprinkler system and Safe Shutdown Analysis are considered Quality Related.

The new isolation devices are being installed into circuits that monitor pressurizer pressure and pressurizer level and supply input to the Safety Assessment System (SAS).

The Safe Shutdown Analysis is being revised to delete from the Analysis cables which have been removed from the Essential Cable List or to change the Analysis for cables which are isolated by transfer switches and therefore no longer require protection. This is being done instead of installing additional sprinklers in these areas, or upgrading the conduit wrap from one (1) hour to three (3) hour rating.

The changes to be implemented by this EDP have been reviewed and found to meet the fire protection requirements put forth in 10CFR 50, Appendix "R". As indicated in Section 3.0, this EDP 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 EDP.

This EDP has no impact on 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 basis for any Technical Specification is reduced.

The modification included in this Engineering Design Package does not involve an unreviewed safety question. The following are the bases for the justification.

a) Installation of Isolation Transmitter

This modification to pressurizer pressure and pressurizer level instrumentation does not change or alter the pressurizer instrumentation system or the alternate shutdown procedure.

The existing pressurizer pressure and pressurizer level instrumentation loops are isolated from the cable spread room by way of a 5K ohm resistor providing isolation for cable shorts and discontinuous circuits (open cable) in the Safety Assessment System (SAS) isolation cabinet. The installation of the signal isolators at the hot shutdown panel will prevent any potential cable to cable failure from propagating to the pressurizer level and pressure signals at the hot shutdown panel.

As shown in Attachment 7.4, the RIS isolated signal transmitters, installed in the Hot Shutdown Control Panel, are located in a mild environment. Therefore, Environmental Qualification per 10CFR50.49 is not required. The addition of the transmitters will not adversely affect the seismic qualification of the HSCP. The transmitters themselves are seismically qualified in accordance with IEEE 344-1975.

The 0-10 VDC input/output as provided by the Rochester Isolated Signal trasmitter is compatible with existing loop requirements, limits and constraints and requires no further modifications as there are no other interface points involved in this instrumentation loop..

This modification ensures that conformance to the separation requirements of 10CFR50 Appendix R is met as committed. The probability of occurrence of an accident or malfunction of equipment and systems previously evaluated in the FSAR has not been increased by this change. Compensatory measures in the interim are identical to those invoked for other fire protection modifications; i.e., a roving fire watch (hourly) and operability of the applicable fire detectors are established.

The possibility for an accident or the malfunction of a different type than described in the safety analysis is reduced by this change since isolation superior to the previous design is provided by this PCM.

The possibility of an accident other than that previously evaluated is not created since:

- The pressure level and pressurizer pressure monitoring circuits are redundant control room inputs; only one safety train is modified by this PCM.

- The replacement of the existing isolation device (5K ohm resistor) with the RIS model SC-1302-323-X represents an enhancement of isolation and control room independence and affects no other systems.

The margin of safety as defined in the bases for any Technical Specification is not reduced because isolation for Alternate Shutdown Instrumentation is now provided by a Class IE component (RIS isolated signal transmitter) which when implemented improves the safety margin.

The RIS model SC-1302-323-X isolated signal transmitter is qualified Class IE per IEEE Section 323-1974. This has been identified as an essential equipment in the St Lucie Plant - Unit 2 Safe Shutdown Analysis.

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

b) Additional Sprinkler System

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 because the preaction sprinkler system is non safety related and does not perform any safety related function nor does it have a direct connection with any safety related system or equipment.

A possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report is not created because there are no new connections made to any safety related system or equipment. In areas where failure of the piping and/or supports may cause damage to safety related system or equipment, the piping is seismically analyzed and supports are seismically designed.

The margin of safety as defined in the basis for any Technical Specification is not reduced because based on a hydraulic check of the sprinkler additions it is determined that design adequacy has been maintained for the proper operation of the fire suppression system.

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

(c) Revision to the cables in the Safe Shutdown Analysis

The following is a list of those cables, which are being revised in the Safe Shutdown Analysis, and the associated PCMs by which they were modified to be removed from the Essential Cable List or isolated by transfer switches.

Cable No	Cable Function	Modification PCM
H21631A	I-SE-09-2, AFW P2A Pwr	Removed by PCM 120-285
H20370F	PT-1108, Pressurizer Pressure RTGB	(See Note 1)
H20649A	120V AC, PP201, HSCP	(See Note 2)
H21738M	PY-1108-4, Pressurizer Pressure HSCP	(See Note 3)
H21738N	LY-1105-1, Pressurizer Level HSCP	(See Note 3)
C20250B	V-3481 SDC 2A Isolation Control	(See Note 4)
H21629E	V-1474 PORV Control	Isolated by TS PCM 130-284
H20253B	V-3651 Control	(See Note 4)
H21630E	V-1475 PORV Control	Isolated by TS PCM 130-284

NOTES:

1. Isolated by isolation device during front fit of St Lucie Unit 2
2. Cable has been removed from PP201 and reconnected to PP201A by PCM 121-285
3. Isolated by isolation device added via this PCM.
4. Breaker racked out by operating procedures - valve can be operated manually by Hand Wheel.

The Safety Analysis in the above PCMs provided the bases for the justification that the implementation of these PCMs did not involve any unreviewed safety question. The effects of the implementation of these modifications on the Plant Technical Specification were also addressed in these PCMs.

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.

HIGH INITIAL RESPONSE EXCITATION SYSTEM

ABSTRACT

This Engineering Package covers modifications to the Turbine-Generator brushless excitation system. The brushless excitation system will be upgraded to a High Initial Response (HIR) Brushless Excitation System which will allow the generator to respond quickly to changes in system voltage.

A larger permanent magnet generator, a new stator coil in the brushless exciter, a new voltage regulator and a new voltage regulator enclosure will be required to modify this system.

The Turbine-Generator does not perform a safety related function. The modifications to the Turbine Generator are classified as non safety related. However, since there will be modifications to the RTG Boards, this package is classified as Quality Related.

This EP does not constitute an unreviewed safety question and the modifications described 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 PC/M does not require a change to the plant Technical Specification.

This change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M 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 Turbine Generator High Initial Response (HIR) brushless excitation system is not a safety related system. A larger permanent magnet generator (PMG), stator coil in brushless exciter and voltage regulator will replace the existing equipment and have no impact on any plant system or operation. The HIR excitation system allows the generator to respond quickly to changes in system voltage.

Subsection 3.5.3.2 of the FSAR addresses External Missiles with subpart (b) addressing Turbine Missiles, specifically, missiles generated by the high pressure turbine rotor and the low pressure turbine discs. There are no changes to the high pressure turbine rotor nor the low pressure turbine discs. The modifications required to upgrade the system include a new PMG rotor, PMG stator and exciter stator which are located at the exciter end. The consequences of turbine failure and the potential for damage to critical plant structures, systems, and components from the resulting missiles has not been increased by this modifiction.

The modifications to the Turbine Generator, the voltage regulator, the voltage regulator enclosure and the HVAC system in the Turbine Building are not safety related and do not affect any plant systems.

The cables for the lighting, receptacles and power feeds in the voltage reguator enclosure are routed in cable tray and conduit in the Turbine Building. They do not require seismic support and do not affect safety related equipment.

The modifications to the RTG Boards will involve the replacement of selector switches with an updated version that are the same model size and have the same characteristics as the existing switches. Additional modifications involve the relabeling of annunciator windows and the actuation of an existing spare relay. These modifications do not effect the safety related functions of the affected RTG Boards.

Based on the preceeding, 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 modifications to the Turbine Generator High Initial Response (HIR) System enhances the operability of the equipment. The introduction of the HIR exciter and voltage regulator in the St Lucie Turbine Generator System will have no effect on the turbine generator control system or the steam supply system (See Attachment 7.5). The addition of a larger PMG, a new stator coil in the brushless exciter, and a new voltage regulator will allow the generator to respond quickly to changes in system voltage.

(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 does not affect any safety related equipment. There are no additional missiles generated by the addition of equipment to the Turbine Generator. 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 new HIR system will allow the generator to respond quickly to changes in system voltage. Since the Turbine Generator is a non-safety related piece of equipment the margin of safety provided by the Technical Specification is preserved.

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 Commission approval for the implementation of this PC/M is not required.

DIESEL GENERATOR CRANK CASE OIL DEFLECTOR PLATE

ABSTRACT

The. St. Lucie Unit 2, 2A, 12 cylinder diesel engine had a problem with false crankcase over-pressurization alarms. Analysis indicated this problem was attributed to oil splashing against the diaphragm of the pressure sensor. To resolve this condition, an oil deflector plate, supplied by EMD the engine vendor, was installed, in front of the crankcase pressure detector sensor. This modification has corrected the problem and does not adversely affect the engine operability. NCR 2-028 (Reference 6.4) and its associated Safety Evaluation accommodated the temporary use of the design change. The temporary modification will be made permanent by this Engineering Package.

Based on the FSAR, the diesel generator is safety related but the oil pressure detector performs no safety related function. Since the oil pressure detector is attached to a safety related structure (the diesel engine), the oil pressure detector must be considered nuclear safety related.

Based on a failure mode evaluation and a 10CFR50.59 review, this modification does not involve an unreviewed safety question nor require a change to the technical specifications. Therefore, prior NRC approval is not required for implementation of the modification. This modification has no effect on plant safety.

SAFETY EVALUATION

This Engineering Package will make the temporary modification, provided in the disposition to NCR-2-208 (Reference 6.4), to the 2A 12 cylinder diesel engine permanent. The engine vendor, EMD, supplied the oil deflector plate, which was installed in front of the crankcase pressure detector.

Although the diesel generator is nuclear safety related, the crankcase pressure detector performs a non-safety related function, which is to trip the engine due to high crankcase pressure during testing situations.

This trip function is overridden when the engine is auto started due to SIAS, CIAS, CSAS, or loss of offsite power.

The oil pressure deflector plate will not in anyway affect the safety related components of the diesel engine. A stress analysis (Ref. 6.3) of the deflector plate demonstrated that failure of the plate is not possible when installed properly.

Based on 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.

- 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.

  A failure analysis (Ref. 6.3) was performed for the deflector plate. Based on the results, it was concluded that a failure of the plate was not a credible event. Therefore, the probability of occurrence of accidents previously addressed in the FSAR has not been increased.

- 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 failure analysis (Ref. 6.3) has shown that this modification will not result in a credible failure of the diesel generator. 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 intended function of the diesel generator is not affected by this modification, the margin of safety as defined in the basis for any technical specification has not been reduced.

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 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 10CFR50.59 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 modification.

10 CFR 50.49 ENVIRONMENTAL QUALIFICATION LIST REVISION

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 2 and no physical plant modifications are required by the Engineering Package. The safety evaluation of this package indicates that a change to the Plant Technical Specifications is not required. The equipment removed from the 10CFR50.49 list are listed in Section 2.1.2 of this PCM. Removal of equipment from the 10CFR50.49 list does not affect plant safety and operation.

Supplement 1

This supplement adds additional splicing materials to the 10CFR50.49 list and updates EQ Documentation Package 2998-A-451-16.1 "Raychem Corporation Splices". This supplement also revises maintenance note 24 of the 10CFR50.49 list and updates EQ Documentation Package 2998-A-451-35.6 "Target Rock Solenoid Valves". 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 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. 2'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 event. 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.

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.

The possibility of new Design Basis Events (DBEs) not considered in the FSAR 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 FSAR 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 since the equipment being removed has been shown to be installed in a mild environment or not required to mitigate and monitor the consequences of an accident. The probability of malfunction of 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.

PRESSURIZER MISSILE SHIELD ACCESS LADDER SAFETY CAGE

ABSTRACT

This design package consists of the fabrication and installation of a personnel safety cage for the pressurizer missile shield access ladder and modification of the ladder. The safety cage will be attached to the ladder. The modification of the ladder is required to provide safe access to the top of the pressurizer wall as well as to the missile shield.

The personnel safety cage does not perform or affect a safety-related function. However, this Engineering Package is classified Quality Related since there is a potential that, during a seismic event, the personnel safety cage could damage safety-related items that are in the vicinity. Quality Related requirements are applied to this design.

This modification has been evaluated in accordance with 10CFR50.59. This safety evaluation indicates that implementation of this EP does not involve an unreviewed safety question, and prior Commission approval for its implementation is not required.

The implementation of this modification does not require a change to the plant Technical Specifications and has no effect on plant safety and operation.

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 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.

The pressurizer missile shield access ladder and safety cage do not perform or affect any safety-related system or function. However, this Engineering Package is classified as Quality Related since failure of the access ladder or safety cage during a design basis event (e.g., earthquake) could potentially affect a safety-related system or equipment, since the ladder and cage are located in the containment building which contains safety-related systems. Consequently, the ladder and safety cage have been designed for the design basis conditions specified in the FSAR and Quality Related design requirements have been implemented, thus assuring the integrity of the installation during any design basis event.

The modifications included in this Engineering Package 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 required to shut down the plant and monitor the plant in a safe shutdown condition.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the ladder and cage perform no safety function and no changes have been made to any operational design. Failure of the ladder and cage could not occur since the modification has been designed for the design basis conditions.

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

The implementation of this Engineering Package does not require a change to 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 Engineering Package is not required.

ICW LUBEWATER FLOWRATOR MODIFICATION

ABSTRACT

The Intake Cooling Water (ICW) Pumps Lubewater Flowrators are Brooks, armored magnetically actuated, rotameter type indicating switches.

This engineering package covers the rebuilding of (6) six rotameters by replacing damaged internals. These changes will not modify the present configuration of the lubewater installation.

The function of each rotameter is to:

1. Assist the operators to adjust the lubewater flow rate supplied to each individual pump bearing cooling water flow.

2. Provide a low flow alarm in the control room.

The ICW pump lube water flowrators are considered to be an extension of the ICW system which is nuclear safety related.

The ICW pumps lubewater flow indicating switches indicates flow and actuates a low flow alarm, therefore their failure will not have any effect on the plant operation or safety.

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 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 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.

The proposed modification affects ICW pump bearing Lube Water flow instrumentation The existing flow indicating switches will be removed, rebuilt by the vendor and reinstalled in the system. No configuration changes will be made, however, the range of the instrument will be increased to protect the float from damage when the flow is higher than normal.

This modification does not involve an unreviewed safety question because:

i) The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The ICW Lube Water Flow instrumentation is not used to determine the probability of any accident and as stated in Section 2, failure of this instrumentation cannot block lube water flow and therefore has no consequence for any equipment malfunction.

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 system function and operation remain unchanged and no new failure modes are introduced. These instruments do not provide a control function, therefore cannot cause the failure of equipment important to safety.

iii The margin of safety as defined in the bases for any technical specification is not reduced. These instruments are not used in the bases of any technical specificatons.

The implementation of this PCM does not require a change to the plant techinical specification.

The foregoing constitutes, per 10CFR50.59 (b), the written safey 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.

S/U XFMR LOCKOUT DISCONNECT SWITCHES

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 Technical Specification conditions. 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.

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:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated, in FSAR Section 8.2.1.5, 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, the emergency diesel generator start and loading will occur as previously evaluated in FSAR Section 8.2.

(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 and prior Commission approval for the implementation of this PCM is not required.

QSPDS SOFTWARE MODIFICATION

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 modifications as a result of this PC/M. However, the exchange of identical Erasable Programmable Read Only Memory (EPROM) chips were required as a result of software modifications.

This Engineering Package is safety related because it involves modifications to a nuclear safety related 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 indicate that there is no unreviewed safety question.

The effect of the modifications on Technical Specifications was evaluated. Since the modifications improve the system by, for example, enhancing the readability of the display, it is concluded that there is no technical specification changes required.

The effect of the modifications on plant safety and operation was evaluated. There is no effect on plant safety and there is no effect on normal plant operations other than the operation of the QSPDS itself. The changes of the QSPDS operations is included in the revised version of QSPDS User's Guide.

SAFETY EVALUATION

This engineering package is safety related because it involves a modification to a safety grade system. We have evaluated the effects of this PC/M with respect to regulation 10CFR 50.59, and concluded that it:

a) Does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

There are no major hardware changes due to this PC/M, since the exchanged hardware (EPROM's) are identical to the 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.

b) Does not create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR has not been created for the same reasons stated in item (a).

c) Does not reduce the margin of safety as defined in the basis for any technical specification.

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 to the operator as a result of this PC/M.

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

In conclusion, this proposed change does not involve an unreviewed safety question or a Technical Specification change; therefore, prior NRC approval is not required to implement this modification.

IE BULLETIN 85-03 MOV SWITCH SETTING

ABSTRACT

NRC IE Bulletin 85-03 requires that operating nuclear plants develop and implement a program to ensure that switch settings on selected safety-related motor-operated valves (MOV's) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. Item a) of the bulletin requires that the design basis for those MOV's located in AFW and HPSI systems be reviewed to determine the maximum differential pressure expected during both opening and closing strokes for all postulated events. This effort was performed for St. Lucie Units 1 and 2 by Combustion Engineering as part of the CE Owner's Group (CEOG) Tasks 528 and 531. The results of the Item a) were subsequently transmitted to the NRC via FPL letter L-86-204, dated May 15, 1986.

Item b) of Bulletin 85-03 requires that the licensee establish the correct MOV switch settings based on the previously determined maximum differential pressure. All switches, including torque switches, torque bypass switches, position limit, position indication, overloads, etc., shall be considered. This design package provides the overall switch setting guidelines for each MOV, in addition to the specific design information necessary to set both the open and close torque switches and meet the requirements of Bulletin 85-03.

Once the correct switch settings have been incorporated into the respective MOV, Item c) of IE Bulletin 85-03 requires that each MOV be stroke tested against the maximum differential pressure established in Item a) to verify operability.

Because all of the MOV's associated with Bulletin 85-03 are safety-related, this engineering package has been classified as nuclear safety-related. A review of the switch setting changes to be implemented by this PC/M was performed against the requirements of 10CFR 50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, the modification described in this engineering package does not constitute 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 FSAR is not increased. This engineering package only provides the necessary design information required to set MOV switch settings utilizing MOVATS signature analysis techniques. The recommended switch settings are considered enhancements to the existing settings to further ensure valve operability. Also, FSAR design bases were reviewed to determine the maximum loading conditions on each MOV to ensure the switch settings were properly selected. Furthermore, Item c) of Bulletin 85-03 requires that each MOV be stroke tested under maximum differential pressure conditions to ensure valve operability.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. No hardware modifications are performed as part of this PC/M. The proposed MOV switch settings alter accident mitigating equipment to further enhance operability. However, malfunctions of these MOV's do not in themselves initiate an accident. Therefore, no new accidents have been created.

Additionally, the specified modifications do not introduce any new failure modes for the equipment. Therefore, no different malfunctions of the equipment than those previously analyzed are introduced.

iii) The margin of safety as defined in the basis for any Technical Specification has not been reduced. This modification does not impact the Technical Specification requirements for the associated equipment. Valve stroke times are not impacted. Therefore, the margin of safety controlled by the Technical Specifications is preserved.

In conclusion, the change proposed in this engineering package is acceptable from the standpoint of nuclear safety does not involve an unreviewed safety question and prior NRC approval for implementation is not required.

NRC IE BULLETIN 85-03 MOV POSITION INDICATION

ABSTRACT

This Engineering Package covers modifications to the safety related motor operated valves (MOV's) in the Auxiliary Feedwater (AFW) and the High Pressure Safety Injection (HPSI) systems.

This Engineering Package will provide the engineering and design details required to implement the close to open torque bypass switch and closed position indication wiring modifications for the motor operated valves.

The MOV's in the AFW and HPSI systems are required for plant safe shutdown and classified as Class 1E, are seismically qualified and perform a safety related function. Therefore, this PC/M is considered Nuclear Safety Related.

This EP does not constitute an unreviewed safety question since the modifications described above were reviewed in accordance with 10CFR50.59 and will not have an adverse impact on plant operations or safety related equipment.

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

This change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M 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 install additional rotors and/or internal wiring changes to MOV's in the AFW and HPSI systems. PC/M 002-287 increases the closed to open torque bypass switch settings which impact the closed position indicating light. Increasing the number of rotors from two to four will allow the limit switch for the closed position indicating light to be located on a rotor other than that used for the torque bypass switch. Motor-operated valves that have four rotors will only require internal wiring changes. The addition of the new rotors does not affect the existing equipment qualifications.

The implementation of this Engineering Package increases the availability of the MOV's during safe shutdown conditions and improves the MOV position indication provided to the control room operators.

NRC Regulatory Guide 1.106, Rev 1 discusses and provides guidance directed at ensuring the thermal overload device will not needlessly prevent the motor from performing its safety function. To ensure that the safety related motor operated valves will perform their function, the thermal overload protection devices are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing. The thermal overload heater devices have been sized using the methodology provided by the "Power Distribution and Motor Data" Sheets.

The MOV's that are being modified perform safety related functions within the AFW and HPSI systems and are designed for operation under conditions that could be imposed by a Design Basis Accident (DBA). This EP has been classified as Nuclear Safety Related.

Based on the preceeding, 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 safety analysis report is not increased, since the modifications to the MOV's enhances the operability of the equipment. The addition of rotors and/or internal wiring changes to the valves will prevent the possibility of inaccurate remote closed position indication resulting from the increased bypass limit switch settings.

(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 alters accident mitigating equipment to enhance their operation. There was 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 MOV's will function as assumed during an accident. Thus the margin of safety provided by the Technical Specifications is preserved.

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 Commission approval for the implementation of this PC/M is not required.

HP TURBINE INNER GLAND AND GLAND DIAPHRAGM ENHANCEMENTS

ABSTRACT

This Engineering Package (EP) documents the original equipment manufacturer's (OEM's) material and design changes to the HP turbine inner gland casings, and gland diaphragms. The changes and bases are as follows:

- o HP Turbine Inner Glands Casings

The material for the HP inner glands has been changed from a carbon steel to a 12% Cr stainless steel to minimize erosion potential. Geometrically the design remains the same.

- o HP Turbine Gland Diaphragms

For these components the material has been changed from a carbon steel to a 12% Cr stainless steel to minimize erosion potential. In addition, the design has been simplified to a single wall vessel versus the previously employed double wall.

This modification has been classified as Non-Nuclear Safety Related because the inner gland casings, and the gland diaphragms are sub-assemblies of the turbine generator's high pressure turbine. The turbine generator is not required for operation during OBE or SSE and also is classified as non-seismic per FSAR sect. 10.2.1.

Based on a failure mode evaluation and a 10 CFR 50.59 review, these enhancements do not involve an unreviewed safety question nor require a change to the Technical Specifications. Therefore, prior NRC approval is not required for implementation of the modified components. These modifications have no effect on plant safety.

SAFETY EVALUATION

The components being enhanced by this EP are a part of the turbine generator assembly, specifically the high pressure turbine stationary casing. The components directly interface with the turbine gland steam system. The turbine generator assembly and the gland steam system perform no safety related function, and are non seismic (FSAR sections 10.2.1, 10.4.3.)

A failure mode evaluation has demonstrated that there is no postulated failure of the components being enhanced that would result in the generation of missles from the H.P. turbine. FSAR Section 10.2.3d, supports this analysis by stating that fragments generated by any postulated failure of the HP turbine rotor would be contained by the HP turbine blade rings and casings.

Title 10 of the Code of Federal Regulations Part 50.59 allows changes without prior Commission approval provided the proposed changes does not involve an unreviewed safety question or require changes to the technical specifications. These proposed component enhancements 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 the safety analysis has not been increased.

   As stated, these component enhancements affect only non-nuclear safety related equipment, and have no affects on the potential probability of turbine missles be generated.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety report is not created.

   The component enhancements are basically a like kind exchange of existing components and therefore an accident or malfunction of a different type than any evaluated previously in the safety report is not created.

iii) The margin of safety as defined in the bases for any technical specification has not been reduced.

   The component enhancements affect no technical specification nor are changes to the Technical Specifications required.

In conclusion, the component enhancements performed under this EP are acceptable from the standpoint of nuclear safety since they do not involve an unreviewed safety question as defined by 10CFR50.59 and do not require changes to the Technical Specifications. Implementation of this modification does not require prior NRC approval.

DIESEL GENERATOR TORSIONAL VIBRATION ISOLATION

ABSTRACT

This engineering package covers modifications to the Diesel Generator (D/G) Fan Drive System which will isolate the D/G Cooling Fans and the Fan Drive System from forced torsional vibrations emitted from the diesel engines. The major feature of this package is the installation of a torsionally flexible coupling at the flange between the Power Takeoff (PTO) shaft and the fan drive shaft for each of the 12 and 16 cylinder engines in the 2A and 2B D/G sets. The change proposed by this engineering package is classified as Nuclear Safety Related, is acceptable from the standpoint of nuclear safety, does not involve an unreviewed safety question and does not require a change to the Technical Specifications.

Supplement 1

This supplement is issued because Morrison-Knudsen, Power System Division (PSD) was contracted to procure and dedicate the flexible couplings to be used in the subject modification. PSD requested the use of Lord Corporation Part No. LCD-0300-20R-C in the 16 cylinder fan drive shafts in lieu of Part No. LCD-200-20R-C as delineated in Supplement 0. Part No. LCD-0300-20R-C has a higher torque rating than the LCD-0200-20R-C which provides a higher margin of safety in the design.

Supplement 2

This supplement is issued to document the engineering acceptance of the diesel generator configuration tested per Supplement 1. This Supplement does not affect, amend, or change the original safety evaluation or Technical Specifications.

SAFETY EVALUATION

This change modifies the Diesel Generators by installing a flexible coupling in the 12 and 16 cylinder diesel engine fan drive shafts. The fan drive shafts are part of the Diesel Generator Cooling Water System which provides sufficient capacity to cool the diesel generator set it serves under postulated loading and ambient conditions. Since the diesel generators are required to provide standby emergency power to Safety Related equipment in the event the preferred power supply is not available, any modification to the D/G's is classified as Safety Related. As demonstrated in the design analysis, this modification has been designed in accordance with the safety and regulatory requirements applicable to the components which comprise the D/G Fan Drive System. In addition, the failure modes analysis (paragraph 2.2.1) confirms that this modification will not prevent the diesel generators from performing their design function of providing emergency power.

Based on 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.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety evaluated in the safety analysis report (reference 6.6) has not been increased because the modification has been designed in accordance with the applicable design and safety requirements applicable for the D/G Fan Drive System. Therefore, the probability of a diesel generator failure has not been increased and the consequences of a diesel generator failure remains the same.

- The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report (reference 6.6) has not been created since this modification does not alter the operational characteristics of the diesel generator sets, apart from reducing vibration levels in the fan drive system. The failure of one diesel generator set and the startup of the redundant set is assumed for all accident analyses. This assumption remains unchanged. Finally, this modification affects no other system. Therefore, no new accident or malfunction is created.

- The margin of safety as defined in the basis for any Technical Specification has not been reduced because the redundancy of the diesel generators required by the Technical Specifications is maintained.

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

In conclusion, the change proposed by 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 the Technical Specifications.

FIRE PROTECTION STRUCTURAL STEEL FIRE PROOFING

ABSTRACT

In order to enhance compliance with 10 CFR Part 50 Appendix "R" requirements, this Engineering Package (EP) provides the following:

a) Engineering and design for the addition of new sprinkler heads outside the Aerated Waste Storage Tank (AWST) Room to provide conduit fire protection.
b) Engineering and design for fire wrapping conduit support steel in areas where adequate protection is currently not provided.
c) Replacement of nine (9) existing conduit supports which are attached to cable tray supports with four (4) new supports and fire wrapping of these new supports.

This EP is designated as Nuclear Safety Related because it modifies Safety Related conduit supports by either fire wrapping the existing conduit supports steel or removing the existing conduit supports and adding new Safety Related conduit supports and fire wrapping them. Changes to the existing sprinkler system are considered Quality Related.

The changes to be implemented by this EP have been reviewed and found to meet the fire protection requirements put forth in 10 CFR Part 50, Appendix "R". As indicated in Section 3.0, this EP neither involves 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 EP.

This EP has no impact on 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 basis for any Technical Specification is reduced.

This EP provides the required modifications to expand the existing preaction fire sprinkler, to provide adequate protection to exposed support steel of Safety Related conduit supports and additional new conduit supports in the RAB. This EP is designated Safety Related.

The modification included in this Engineering Design Package does not involve an unreviewed safety question. The following are the bases for the justification.

a) Addition of Sprinkler Heads

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 because the preaction sprinkler system is non safety related and does not perform any safety related function nor does it have a direct connection with any safety related system or equipment.

A possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report is not created because there are no new connections made to any safety related system or equipment. In areas where failure of the piping and/or supports may cause damage to safety related system or equipment, the piping is seismically analyzed and supports are seismically designed.

The margin of safety as defined in the basis for any Technical Specification is not reduced because based on a review of the hydraulic calculation for the existing sprinkler system these sprinkler head additions do not affect the design adequacy for the proper operation of the fire suppression system.

b) <u>Replacement of existing conduit supports attached to CTRs with new conduit supports and wrapping of new/existing conduit support steel</u>

This modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report because it does not change or alter the intended function or design requirements of any safety related conduit originally installed.

This modification does not create a possibility for an accident or malfunction of a different type than any evaluated previously because there are no new connections made to any safety related system or equipment by this modification. The existing conduit supports being removed from the CTRs have been replaced with new conduit supports. These new conduit supports have been seismically designed. The construction note (9.6) requires that the new conduit supports shall be installed prior to removing the existing conduit supports from CTRs; therefore the structural integrity of the conduits affected by this modification are not compromised. The existing conduit supports being wrapped have been evaluated for additional loads of fire wrap material and determined to be adequate for these loads. The seismic block wall to which a new brace from a conduit support is added has been evaluated for the additional load and the structural integrity of the masonry block wall is not compromised.

The margin of safety as defined in the basis for any Technical Specification is not affected by this modification because the components involved in this modification are not included in the bases of any Technical Specification.

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

The foregoing constitutes, per 10 CFR Part 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 EP is not required.

TURBINE GENERATOR ADDITIONAL OIL SEAL FOR 1 AND 2 BEARING

ABSTRACT

This engineering package covers the addition of one supplemental labyrinth to the #1 and #2 bearings oil seals. Oil leakage from these seals could lead to fires due to the proximity of the seals to hot surfaces. To preclude potential leakage, Westinghouse (the turbine generator vendor) has designed and fabricated the supplemental seal which functions as an integral part of the existing seals. Use of the supplemental seal increases sealing capabilities thereby reducing the liklihood of oil leakage. This modification is classified as Non-Nuclear Safety Related, Quality Group D, but the design has been classified as Quality Related due to explicit Quality Control requirements pertaining to the installation effort.

Based on the 10 CFR 50.59 review and the failure modes evaluation, it has been demonstrated that this change does not involve an unreviewed safety question. Additionally, no change is required to the Technical Specifications. This modification does not adversely affect plant safety or operability. Prior NRC approval is not required for implementation of this design.

SAFETY EVALUATION

The Unit 2 turbine generator is located in a non-nuclear safety related system and as such is not required to function for accident mitigation. These modifications affect only non-nuclear safety related Quality Group D equipment.

Based on the failure mode evaluation, failure of the components added by this modification will not inhibit the operation of any existing safety related equipment or components. This evaluation is based on the assumption the new seal is installed according to design. Adequate Quality Control inspections have been specified to verify proper installation and therefore operation. Accordingly, this design is classified as Quality Related.

Title 10 of the Code of Federal Regulations Part 50.59 allows changes without prior Commission approval provided the proposed change does not involve an unreviewed safety question or require changes to the Technical Specifications. This proposed change 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 has not been increased.

   As stated, the modification affects only non-nuclear safety related Quality Group D equipment. In addition, the failure modes analysis demonstrates that no safety related equipment is affected by this modification.

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 failure modes analysis has shown that the possibility of an accident or malfunction of a different type than any previously evaluated in the safety analysis is not created.

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

   This design modification affects no technical specification nor are changes to the Technical Specifications required.

Based on this information, an unreviewed safety question as defined by 10CFR50.59 is not created. Since no accident previously identified in the safety analysis report has been affected, no new accidents or malfunctions are created and no changes to the Technical Specifications are required. An unreviewed safety question does not exist. Prior NRC approval is not required for implementation of this modification.

REPLACEMENT OF VALVE V3734

ABSTRACT

This Engineering Package (EP) provides for replacement of St. Lucie Unit 2 Safety Injection Tank 2A2 solenoid vent valve V3734. The existing valve manufactured by Garrett Pneumatic Systems has failed and a direct replacement is not available. The Garrett valve will be replaced with a Target Rock Model 80B-001 valve.

This modification is classified nuclear safety related since the Safety Injection Tank Solenoid Vent Valves according to FSAR Section 6.3.2.2.1 are nuclear safety related. This EP does not have any adverse impact on plant safety and operation. 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 change to the Technical Specifications. Therefore, prior NRC approval is not required for implementation of the modification.

SAFETY EVALUATION

This EP provides for replacement of St. Lucie Unit 2 SIT 2A2 solenoid vent valve V3734. The existing valve manufactured by Garrett Pneumatic Systems has failed and a direct replacement is not available. The Garrett valve will be replaced with a Target Rock Model 80B-001 valve.

This modification is classified nuclear safety related since the SIT solenoid vent valves according to FSAR Section 6.3.2.2.1 are nuclear safety related. This EP does not have any adverse impact on plant safety and operation. The new SIT solenoid vent valve has been designed to Safety Class 2, Quality Group B, Seismic Category I, and Class 1E requirements. All safety and regulatory requirements specified in FSAR Section 6.3 have been met.

The function of the SIT vent valves is as follows:

During plant cooldown, the SIT solenoid vent valves may be used. When the Reactor Coolant System pressure is 650 psia the SITs are depressurized to 235 psia. The SITs can be depressurized by either opening the SIT vent valves or by draining the SIT to not less than 48 percent full.

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.

- o 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 replacement SIT solenoid vent valve meets all safety and regulatory requirements specified n the FSAR. The operating characteristics of the new valve are shown to be acceptable by Section 2.0. The replacement and original valve are functionally equal. Based on this, the probability of occurrence or the consequences of all analyzed accidents remain unchanged.

- o 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 proposed design change alters accident mitigation equipment, Safety Injection System. There are no accidents that are initiated by malfunctions associated with this system. Therefore, no new accidents have been created.

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

  Technical Specification 4.5.2.a requires once per 12 hours that valve V3734 be verified in a locked close position. This modification does not affect this Tech. Spec. Thus, the margin of safety provided by valve V3734 and controlled by the Technical Specifications are preserved.

  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 are 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. 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 changes to Technical Specifications. Therefore, prior NRC approval is not required for implementation of the modifications.

CONDENSATE RECIRCULATION TO CONDENSER
SQ RT EXTRACTOR REPLACEMENT

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 PC/M was performed against the requirements of 10 CFR 50.59. As indicated in Section 3.0 of this PC/M, this PC/M 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 PC/M.

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 equipment.

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 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.

MFRV POSITION INDICATORS REMOVAL

ABSTRACT

This engineering package covers the removal of two Main Feedwater Regulating Valve position indicators ZI-9011 and 9021 from RTG Board 202 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 202.

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.

The change is not an unreviewed safety question because the probability of occurrence or the consequences of an accident or malfunction important to safety previously evaluated in the FSAR has not been increased.

Internal wiring changes are being performed in the RTG Board to disconnect the subject indicators and to remove (SIS) wiring. When required, only jumpers of the type qualified 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.

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.

> These indicators are not considered by the FSAR in determining the probability of accidents, possible types of accidents or in the evaluation of consequences of accidents. Also they could provide misleading information which their removal would prevent. Therefore it can be concluded that the probability of occurrence or consequences of accidents previously addressed in the FSAR remains unchanged.

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

> As stated above, these indicators are unreliable and could provide misleading valve position indication. Since these indicators are located in the Control Room, misleading information from them could lead to an accident or malfunction of a different type than any previously evaluated in the FSAR. By removing them, this possibility is eliminated since this chance of error is no longer present.

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

> These indicators are not required by any technical specification nor are they included in the basis of any technical specification. Therefore, their removal will not reduce the margin of safety as defined in the loses for any technical specification.

In conclusion, this modification does not involve an unreviewed safety question.

REACTOR CAVITY SEAL RING MODIFICATION

ABSTRACT

This Engineering Design Package covers modifications to the Reactor Cavity Seal Ring. The pneumatic seals have been modified by the vendor. The male studs used to attach the seals to the seal plate have been changed to female threaded fittings. Also, the seal air lines have been changed from neoprene hose to stainless steel braid hose. These modifications are necessary to improve the reliability and operability of the seal and the air lines.

Based on the FSAR, the cavity seal ring is non-nuclear safety related. However, to ensure the Reactor Cavity Seal Ring will perform its intended function, quality requirements are assigned. Therefore, this modification is classified as Quality Related.

Based on a failure mode evaluation and a 10 CFR 50.59 review, these modifications do not involve an unreviewed safety question nor require changes to the technical specifications. Therefore, prior NRC approval is not required for implementation of the modifications. These modifications have no effect on plant safety.

Supplement 1

Supplement 1 incorporates a minor drawing revision. The changes made by this supplement are non-technical and administrative in nature. The drawing was revised to change the drawing number and revision. Design Integration has been reviewed and it has been determined that there are no adverse consequences as a result of revising the drawing. There are no other changes to this package. The safety evaluation remains valid since there are no unreviewed safety questions and no changes to the technical specifications are required. This change has no effect on plant safety.

SAFETY EVALUATION

This Engineering Package covers modifications to the Reactor Cavity Seal Ring. The pneumatic seals have been modified by the vendor, the studs used to attach the seals to the seal plate have been changed to female threaded fittings and the seal air lines have been changed from neoprene to stainless steel braid.

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.

  Since the reactor cavity seal ring is not considered by the FSAR in determining the probability of accidents, possible types of accidents, or in the evaluation of consequences of accidents, 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.

  Since the sealing portion of the cavity seal ring has not been changed, 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.

  Again, since the sealing portion of the cavity seal ring has not changed, 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 changes to the technical specifications. Therefore, prior NRC approval is not required for implementation of the modification.

CONDENSATE PUMP EXPANSION JOINT REPLACEMENT

ABSTRACT

This Engineering Package covers the change out of the St. Lucie Unit 2 Condensate pump expansion joints and the modification to the adjacent pipe supports. The existing expansion joints are made of an elastomeric material which has deteriorated due to aging. The replacement expansion joints are made of stainless steel and will correct the problems associated with the aging deterioration.

These modifications are classified as non-nuclear safety related, according to the FSAR. Based on a failure mode evaluation and a 10CFR50.59 review, these modifications do not involve an unreviewed safety question nor a change to the technical specifications. Therefore, prior NRC approval is not required for implementation of these modifications. These modifications have no effect on plant safety.

SAFETY EVALUATION

This Engineering Package covers the modifications to the condensate pump expansion joints. The elastomeric expansion joints will be replaced with stainless steel expansion joints.

Based on the above and the information supplied in the design analysis, it can be demonstrated that an unreviewed safety question as defined by 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.

Due to the location of these expansion joints, their failure would not cause interaction with any safety related equipment. Also since the condensate system is not considered by the FSAR, Section 10.4.7, in determining the probability of accidents, possible types of accidents, or in the evaluation of consequences of accidents, 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 any safety related function. The operational design of the condensate system has not been affected by the material change of the expansion joints. Also, due to their location, the failure of these expansion joints would not cause interaction with any 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.

10CFR 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 10CFR 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.

CONDENSER HOTWELL NITROGEN INJECTION CONNECTIONS

ABSTRACT

This Engineering Package covers modifications to the St. Lucie Unit 2 condensers to allow the installation of taps for the purpose of injecting nitrogen into the condenser hotwells. Testing (Ref 6.4) has shown that injecting 1 cfm of nitrogen into a condenser shell reduces condensate dissolved oxygen concentration by approximately 2 ppb. The flow of non-condensibles in the air removal section of the tube bundle becomes inhibited when there is low air in-leakage into the condensers. Oxygen is entrained as the condensate drips through the air pockets which form as a result of the stagnant conditions. Injecting an inert gas such as nitrogen enables the air removal section of the condenser to establish the flow required to remove non-condensibles without introducing additional oxygen into the system.

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 affect on plant safety or operability.

Supplement 1

Supplement 1 adds four (4) weld numbers to drawing number JPE-051-287-005. The changes made by this supplement are non-technical and administrative in nature. The drawing was revised to include the weld numbers and revision change. Design Integration has been reviewed and it has been determined that there are no adverse consequences as a result of revising the drawing. There are no other changes to this package. The safety evaluation remains valid since there are no unreviewed safety questions and no changes to the technical specifications required. This change has no effect on plant safety.

SAFETY EVALUATION

This Engineering Package covers the modifications necessary to install condenser taps for the purpose of injecting nitrogen into the condenser hotwells. The condensers are classified as non-nuclear safety related, quality group D. A complete failure of these connections could result only in a loss of condenser vacuum and subsequently a turbine trip. However, no safety related equipment, components or safety related functions are affected.

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

- o 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 piping, valves and control devices associated with this modification, no interaction with safety related equipment will occur in an event of failure. Also, the condenser is not considered by the FSAR in determining the probability of accidents, possible types of accidents, or in the evaluation of consequences of accidents. It can be concluded that the probability of occurrence of accidents previously addressed in the FSAR remains unchanged.

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

In the event of failure, the equipment added by this Engineering Package will not interact with any safety related equipment due to the location of the modifications. Also, the installation of the condenser taps does not change the intended function of the condensers. Therefore, the possibility of an accident of a different type has not been created.

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

Again, since the intended function of the condenser is not affected by this modification, the margin of safety as defined in the basis for any technical specification has not been reduced.

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 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 10CFR 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 the technical specifications. Therefore, prior NRC approval is not required for implementation of the modification.

CONDENSER POLISHER TIE-IN

ABSTRACT

This Engineering Package (EP) is for the installation of the 24 inch tie-in piping and valves required for the future connection of the Condensate Polisher System (CPS) to the Unit 2 Condensate System. It also includes the installation of the by-pass flow control valve required for operating the CPS using Unit 2 condensate and the installation of a connection to the Unit 2 condensate storage tank for providing the capability of using Unit 2 condensate for backwashing the condensate polishers.

This EP is classified non-safety related since the portions of the Condensate System and Condensate Storage Tank piping where this modification will be implemented do not perform any 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 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 piping and valves required for the future connection of the Condensate Polisher System (CPS) to the Unit 2 Condensate System. It also includes the installation of the by-pass flow control valve required for operating the CPS using Unit 2 condensate and the installation of a connection to the Unit 2 condensate tank for providing the capability of using 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 does not perform any safety function or interact with safety related equipment, therefore this package is classified as non-nuclear safety related.

Based on the above description, the modification included in this Engineering Package (EP) is considered to be non-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. The Condensate System 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.

(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 the PCM is not required.

MSR PARTITION PLATE NUT REPLACEMENT

ABSTRACT

This PC/M provides for the replacement of the moisture separator reheater tube bundle hemi-head partition plate nuts with new Westinghouse nuts made of a different material. The existing nuts were found to be susceptible to stress corrosion cracking and failures have been experienced at various Westinghouse Plants, including Turkey Point. Failure of these nuts can result in degraded MSR performance.

A review of the changes to be implemented by this PC/M was performed against the requirements of 10CFR50.59. As a result, this modification is classified as non-safety related, does not constitute an unreviewed safety question, will not affect plant safety, and does not require a change to the plant Technical Specification.

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 modification does not involve an unreviewed safety question because:

i) The probability of occurance or the consequences of an accident or malfunction of equipment important to safety prevoiusly evaluated in the safety analysis report is not increased. The MSR's 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.

ii) The possibility for an accident or malfunction of a different type that 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 component involved in this modification is not included in the bases of any Technical Specification.

The implementation of the 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 the PCM is not required.

480V SWITCHGEAR 2A1 and 2B1 TRANSFORMER REPLACEMENT

ABSTRACT

Due to environmental concerns attendant to polychlorinated biphenyl (PCB) cooling/insulating liquids, all transformers filled wth PCB are being eliminated from FP&L's system. The station service transformers for 480 volt switchgear 2A1 and 2B1 are filled with PCB cooling/insulating oil. Each transformer contains 254 gallons of PCB liquid. This Engineering Package provides for the replacement of the existing PCB filled station service transformers with equivalent transformers filled with an environmentally acceptable silicone cooling/insulating liquid.

Station service transformers 2A1 and 2B1 do not perform any nuclear safety related functions, however, because of their importance to normal balance of plant operations the replacement transformers are classified as Quality Related in this Engineering Package.

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.

Supplement 1

Supplement 1 incorporates vendor drawings and associated engineering design calculation certification sheet for information only. 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 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 addresses the replacement of station service transformers 2A1 and 2B1 which are both non-safety related. The FSAR refers to "two non-safety" related transformers (2A1 and 2B1) in Subsection 8.3.1.1.1.c, on the bottom of page 8.3-4. On FSAR Figure 8.3-2a the 2A1 and 2B1 station service transformers are identified as non-Class 1E, i.e. non-safety related.

Station service transformers 2A1 and 2B1 do not perform any nuclear safety related functions, however, because of their importance to normal balance of plant operations the replacement transformers are classified as Quality Related in this Engineering Package.

The 2A1 and 2B1 station service transformers are located on the ground elevation of the turbine building. The transformers will be replaced on a one-for-one basis by transformers essentially identical except for the silicone cooling/insulating liquid.

Based on the preceeding, 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 classified as non-safety related in the FSAR and they are being replaced on a one-for-one basis by transformers that are identical in form, fit and function.

(ii) This modification does not change the operation of the non-safety related 480 volt auxiliary power distribution system. 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 identical in form, fit 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 Commission approval for the implementation of this PC/M is not required.

BORIC ACID MAKEUP SYSTEM RELIEF VALVE MODIFICATION

ABSTRACT

This Design Package covers the installation of lap joint flanges on the inlet side of twelve (12) 1/2" x 1" relief valves on the Boric Acid Makeup System; V-2123, V-2160, V-2171, V-2630, V-2631, V-2632, V-2634, V-2636, V-2637, V-2639, V-2641, & V-2648. This will allow post-maintenance reassembly of the relief valves without regard to inlet flange bolt hole alignment. In addition, the relief valve manual lift levers will be removed and their activating shafts seal welded to eliminate leakage. The relief valves involved provide thermal relief protection for ASME Section III Class II piping, which makes this modification safety related. Based on a failure mode analysis and 10CFR 50.59 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 Tech Spec change is not required, therefore, prior NRC approval is not required for implementation of this modification. The function of the relief valves is not altered by this modification.

Safety Evaluation

This modification consists of the replacement of the existing socket weld flange with lap joint flanges, the removal of the relief valve manual lift lever, and the seal welding of the lift lever activating shaft. This modification does not affect the design function of the relief valves, and does not introduce any new active components to the system. In fact, the removal of the manual lift lever eliminates one potential failure mode of the relief valves; that of the relief valve inadvertently being manually lifted. 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 9.3.4.1 and 9.3.4.3.2 (h) 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.

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, NRC approval is not required for implementation of the modifications.

LOW POWER FEEDWATER CONTROL SYSTEM

ABSTRACT

In order to reduce the frequency of reactor trips encountered during start-up with manual control of the Feedwater by-pass regulating valves, a new Low Power Feedwater Control System (LPFCS) will be added to the existing Feedwater Control System. The LPFCS is designed to provide stable and automatic control of the by-pass feedwater regulators, which offers an additional advantage over the present manual operation at low power loads in the load range of 2-15%.

The inherent design of this equipment is to provide for a smooth and steady output for automatic control of the by-pass regulators and to significantly reduce the frequency of reactor trips during unit start-up. This equipment does not perform any Safety Related functions and is not required for safe shutdown or alternate shutdown functions. However, this equipment will be installed in the Control Room and will be seismically evaluated. Therefore, this package shall be considered Quality Related.

The implementation of this PCM will have no adverse 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 EP, this EP 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 this new Low Power Feedwater Control System (LPFCS) is an extension of the Feedwater Regulating System and as described in FSAR Subsection 7.7.1, this system function is not essential for the safety of the plant. The installation of the LPFCS will provide control improvements to maintain steam generator water level at set point value during unit start-up with significant reduction in the number of reactor trips due to steam generator level excursions.

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

a) The new equipment mountings and added components have been seismically analyzed for additional loading and it has been concluded that these additions will not alter the original stress conditions or the fundamental frequency of the RTG Boards. Consequently, the seismic qualification of the RTG Boards will not be adversely affected.

b) Also, the LPFCS, which is an extension of the Feedwater Regulation System, is 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 PCM 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.

INSTALLATION OF VERNIER MERCURY MANOMETERS

ABSTRACT

Unit 2 main condenser pressure is measured by two full range, electronic, absolute pressure transmitters connected to Condenser 2A, and one narrow range, absolute pressure transmitter connected to Condenser 2B. Corresponding electronic receivers located in the Control Room panels complete the existing monitoring instrumentation.

The main condensers are classified as non-safety related. However, this Engineering Package (EP) will be classified as Quality Related to assure that good construction practices are followed and to assure added confidence during the design and installation to prevent mercury contamination of the condenser condensate, feedwater and the steam generators.

This EP covers modifications to the Main Condenser Pressure Monitoring System by adding one locally mounted, high precision, 35 inch range, vacuum mercury manometer per condenser and a 35 inch range barometer. These three instruments will be fitted with a 26 to 31 inch range Vernier scale to improve the reading precision to 1/100 of 1 inch. The improved accuracy will help in assessing when condenser cleaning is necessary.

A review of the additions implemented by this PCM was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this package, this EP does not involve an unreviewed safety question, nor does it require a revision to the Plant Technical Specification. Therefore, prior Commission approval is not required for implementation of this PCM.

SUPPLEMENT 1

This EP Revision incorporates the following:

a. Preventing the mercury from entering the condenser.

b. Verification of FSAR commitments for the use of mercury on site.

c. Impact of the use of mercury upon the NSSS equipment guarantees.

d. Special instructions for handling of mercury.

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 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 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, Section 10.4.1, "Main Condenser," is not increased since the mercury manometers and the barometer do not perform any safety function. In addition, these instruments are not essential for the safety of the plant and are not connected to any plant safety related systems.

ii) The possibility of an accident or malfunction of a different type other than any evaluated previously in the Safety Analysis Report, Section 10.4.1, "Main Condenser," is not created since the mercury manometers and the barometers 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 Engineering Package since the components involved in this modification are not included in the bases of any Technical Specification and they do not change the original operational capability of the equipment.

The implementaton of this PCM 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.

ANNUNCIATOR NUISANCE ALARMS

ABSTRACT

The Engineering Package (EP) includes engineering and design necessary to correct annunciator nuisance alarms requiring set point and logic modification as well as alarm circuit deletions. By implementing this EP, these circuits will be consistent with the NUREG 0700 "Guidelines for Control Room Design Review" "Dark Annunciator" concept which allows for alternately flashing annunciators in the alarm state only. Under normal operating conditions no annunciators will be illuminated.

This EP is classified as Nuclear Safety Related since it involves modifications of Nuclear Safety Related circuits, necessary to correct these nuisance alarms. The safety evaluation has determined that this EP does not constitute an unreviewed safety question and does not require a change in the Plant Technical Specifications. This PCM can be implemented without prior Commission approval.

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 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 controlling functions. 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. The Safety Related circuit modifications do not affect the purpose, function or operation of control circuits.

(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. The modified annunciator windows do not perform any Safety Related functions and do not modify the control functions of any Safety Related circuit.

(iii) The margin of safety as defined in the bases of any technical specification is not reduced since the modified annunciator alarms perform non- nuclear safety related functions and are not included in the bases of any technical specification. The Safety Related circuits which were modified have been analyzed, and it has been determined that there is no effect on control circuit set points or response times prescribed by Technical Specifications.

The modified logic of some annunciator alarms is interfaced with the control logic of nuclear safety related equipment, therefore, this EP is classified Nuclear Safety 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 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.

NEW FUEL CRANE INTERLOCK ADDITION

ABSTRACT

This Engineering Package (EP) modifies circuits in and adds components to the New Fuel Crane in the Fuel Handling Building to provide improvements as follows:

> Install photoelectric sensor elements, control relays, and reflective tape as an interlock system to restrict fuel crane movement in order to prevent damage from collision between the fuel crane and observation platform.

This EP is classified as Quality-Related since it provides for modifications to equipment not required to shut down the plant or to mitigate the consequences of a Design Basis Accident but which is used to handle new nuclear fuel. The safety evaluation has shown that the implementation of this EP does not constitute an unreviewed safety question nor would implementation affect 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 basis for any technical specification is reduced.

The modifications included in this Engineering Package, which consist of photoelectric travel limit interlocks in the New Fuel Crane control circuits, 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 this EP provides for increased protection of the New Fuel Crane and structures in the Fuel Handling Building by reducing the potential for damage due to mishandling, and since no anticipated mode of interlock failure will affect equipment required to shut down the plant or to mitigate the consequences of an accident. The modifications do not change the designed function of the crane.

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 of the New Fuel Crane. The addition of the interlocks does not introduce the potential for new accidents because no anticipated mode of interlock failure will affect equipment required to shut down the plant or to mitigate the consequences of an accident, and because the new interlocks provide further restriction of movement but do not otherwise change the operating characteristics of the New Fuel Crane.

iii) This modification does not change the margin of safety as defined in the basis for any Technical Specification, since no anticipated mode of interlock failure changes any parameter referenced in the Technical Specifications.

This EP modifies equipment that is not Nuclear Safety-Related. However, since the equipment is used for handling new nuclear fuel, and since mishandling could result in fuel damage, this EP is classified as Quality Related.

This EP has no effect on cables, structures, or components necessary for safe shutdown of the plant, or on equipment 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 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.

MFIV TERMINAL STRIP REPLACEMENT

ABSTRACT

This Engineering Package provides for the replacement of terminal strips in the terminal boxes of the four Main Feedwater Isolation Valves. The existing terminal strips have experienced a recurring problem with loose connections which causes unreliable valve operation. The replacement terminal strips are already in use in the steam trestle area and have been qualified by Environmental Qualification Documentation Package 17.1, drawing number 2998-A-451-17.1.

This Engineering Package is classified as Nuclear Safety Related due to the classification and safety functions of the Main Feedwater Isolation Valves. Implementation of this PCM 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 PCM 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.

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 this EP provides for the replacement of existing MFIV terminal strips with the more dependable Buchanan Type NQB112 terminal blocks.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since this EP does not affect the existing design philosophy of the MFIV control system.

(iii) This modification does not reduce the margin of safety as defined in the bases for the technical specifications since it improves the mechanical integrity of the MFIV control circuit.

Since this EP affects equipment that is identified as nuclear safety related in the PSL-2 Final Safety Analysis Report, subsection 10.4.7.1, this package is considered Nuclear Safety Related.

Although this EP involves equipment on the Essential Equipment List, it does not affect safe reactor shutdown or alternate shutdown. There are no other changes to equipment which involves 10CFR50 Appendix "R" fire protection (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 EP does not require a change to the Plant Technical Specifications and 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 EP is not required.

RELOCATION OF THE SBVF HEATER CONTROL PANELS

ABSTRACT

The Shield Building Vent System (SBVS) maintains a negative pressure inside the annulus and filters for removal of fission products following a LOCA. Thus the SBVS prevents containment leakage from flowing directly from the annular space, through the Shield Building Structure, to the atmosphere. The SBVS is actuated automatically by a Containment Isolation Actuation Signal or a high radiation signal from the Fuel Handling Building.

The Engineering Package (EP) covers the relocation of the Shield Building Ventilation Fan (SBVF) Heater Control Panels to a mild environment to allow for a reduction in EQ maintenance requirements.

The SBVF Heater Control Panel is part of the Shield Building Vent System and is classified as Nuclear Safety Related. Since this modification only covers relocation of the Heater Control Panel (HCP) with no component changes to the panel, the same classification applies. Plant safety and operation are not affected by this change.

The safety evaluation of this package indicates that the relocation of the HCP does not involve an unreviewed safety question, and does not require a change in the Plant Technical Specifications. Therefore, NRC notification is not required prior to implementation of this EP.

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 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 will not be increased by this modification since there are no component changes associated with the relocation of the control panel. Although the flame test requirements for cables (IEEE 383-1974) were not addressed in the Action Test Report No. 17414 as required by St Lucie Unit 2 FSAR Section 8.3.1.1.4, the cables are in a dedicated conduit from end to end. Therefore, the operation of equipment described in the technical specification is 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 since there is no change in system operation and the new location (a reinforced concrete wall) is more rigid than the location for which the CP was originally qualified (the side of the Shield Building exhaust fan).

(iii) The margin of safety as defined in the bases for any technical specification is not reduced by this modification since the relocation of the equipment does not alter any circuits, and the relocation to a mild environment will reduce maintenance requirements.

The Shield Building Vent System is Class IE (Electrical) and is Nuclear Safety Related, therefore, the Engineering Package (EP) is Nuclear Safety 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 commission approval for the implementation of this EP is not required.

REPLACEMENT OF FISCHER AND PORTER CONTROLLERS

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, redundance, 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.

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.

ERDADS/SAS UPGRADE

ABSTRACT

This Engineering Package (EP) provides for modifications in Control Room equipment to upgrade the Emergency Response Data Acquisition and Display System (ERDADS), which is also known as the Safety Assessment System (SAS) and includes Safety Parameter Display System (SPDS) equipment. This EP will improve the performance and display capabilities of the existing system and will include new display CRTs and keyboards and a new color hardcopier.

The Engineering Package is classified as Quality Related since the SAS system is a computer based data processing and display system which assists Control Room personnel in evaluating the safety status of the plant and since the modifications in the Control Room involve installation of equipment in RTGB-204. Implementation of this PCM does not involve an unreviewed safety question or a change to the Plant Technical Specifications. It can be implemented without prior Commission approval.

Implementation of the PCM will not affect the safety or operation of the plant.

SUPPLEMENT 1

This EP revision provides for modifications in the Control Room in preparation for implementing an upgrade to the ERDADS/SAS equipment. Included in this work are installation of conduit and cable, relocation of existing ERDADS/SAS equipment, and installation of mounting hardware to allow future installation of ERDADS/SAS equipment.

The Engineering Package is classified as Quality Related since SAS is a computer based data processing and display system which assists Control Room personnel in evaluating the safety status of the plant. Implementation of this PCM does not involve an unreviewed safety question or a change to the Plant Technical Specifications. It can be implemented without prior Commission approval.

Implementation of the PCM will not affect the 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 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 does not modify equipment which is 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.

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. (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.

EXTRACTION STEAM PIPE AND FITTING MATERIAL UPGRADE

ABSTRACT

This design package provides details and instructions to replace degraded carbon steel piping and fittings in the extraction steam systems with chromium-molybdenum alloys on an "as-needed" basis. The extent of the replacement required for each situation will be based on ultrasonic inspection data to be reviewed by Power Plant Engineering during the 1987 refueling outage. The required replacement will be reported to construction, and details of each replacement will be added to the package via the CRN process.

This PC/M also provides details to replace two specific sections of extraction steam piping. These sections are identified for replacement since they are similar in terms of design and operating conditions to the section of Unit 1 extraction steam piping which failed during 1986. Theoretical erosion/corrosion rates indicate that ANSI B 31.1 requirements for minimum wall thickness may be violated during the next one to two power cycles.

This PC/M is classified as "Non-Nuclear Safety Related" since it affects only nonseismic, Quality Group D piping 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 EVALUATION

The Unit 2. Extraction Steam System is a Non-Nuclear Safety Related System and as such is not required to function during any existing analyzed accident scenario. Therefore, modifications to these pipes affect only Non-Nuclear Safety Related, Quality Group D equipment.

The modification is a material upgrade only. The new material has been shown, in the Design Analysis, to meet all design requirements of the previous material.

Postulated failures of the extraction steam line would have no impact on safe shutdown of the plant or safety related systems. The extraction steam lines are 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 an extraction steam line 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 report 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.

MISCELLANEOUS ICW SYSTEM MODIFICATIONS

ABSTRACT

This engineering package enables minor modifications to be made to the Intake Cooling Water (ICW) system resulting from disassembly, inspection, repair and reassembly during the 1987 refueling outage. Those modifications that meet the criteria established by this design package shall be initiated via the Change Request/Notice form and dispositioned by engineering. Those modifications which do not meet the criteria established by this design package shall be implemented under separate design packages. Those modifications to the essential portion of the ICW System are classified as nuclear safety related, therefore the PC/M is classified as safety related. Modifications to the non-essential portion of the ICW System are classified as non-nuclear safety related unless the failure mode analysis determines an interaction with equipment important to safety. If so, quality requirements will be applied and the modification classified as Quality Related. The changes proposed in this design package are acceptable from the standpoint of nuclear safety, do not involve an unreviewed safety question, do not require a change to the Technical Specifications and do not require prior NRC approval prior to implementation.

SAFETY EVALUATION

The modifications to the essential portion of the ICW system described in the project scope are classified as nuclear safety-related because the failure of the modified component in conjunction with the worst case single failure as analyzed per FSAR Table 9.2.2 would result in the inability of the ICW system to achieve its design basis safety function. Historically, the types of modifications to the ICW System resulting from the disassembly and reassembly of the piping system for inspection and repair have been:

1. Modifications to pipe vent and drain lines (e.g., replacement of corroded material).

2. Modifications to support/restraints (e.g., documentation of weld symbols required to reassemble S/R's, excessive gap at S/R base plates, replacement of corroded material).

3. Weld repair to ICW pipe (e.g., documentation of pipe welds).

4. Pipe flange bolting material changes or bolt torque valve documentation.

As described in the design bases, these nuclear safety-related modifications shall be made in accordance with the design code requirements for Safety Class 3 pipe and pipe components and for Seismic Class I support/restraints.

In accordance with the requirements specified in the design bases, each modification to the non-nuclear safety-related portion of the ICW system shall have a failure mode evaluation performed to determine if there are any. interactions with safety-related equipment or functions. Since the non-nuclear safety related portion of the ICW system is not relied upon for any accident prevention or mitigation, failures which are determined to not impact the function of the nuclear safety-related portion of the ICW system are acceptable with regard to nuclear safety. No Quality Related requirements will be applied to the design of these modifications. However, if a modification to the non-nuclear safety-related portion of the ICW system is determined by the failure mode evaluation to interact with Nuclear Safety Related equipment, Quality Related requirements will be applied to the design of these modifications.

Based on the above, 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 a Design Basis Accident (DBA) evaluated in the FSAR is not increased because no DBA's deal with specific ICW component failures. The modifications restore the ICW system and original design condition and ensure its safety function will be performed.

ii) The probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased because the modifications proposed by this design package are to passive components only and they will be designed/implemented in accordance with safety class/FSAR requirements. The FSAR does not evaluate passive component failures.

iii) The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR is not created because the modifications permitted by this design package do not alter the ICW system function or mode of operation. The FSAR evaluation of the ICW system envelopes the failure of the described modified components.

iv) The margin of safety as defined in the basis for a technical specification is not reduced. The modifications permitted by this design package have been reviewed and found acceptable. No changes to the design basis, function, or mode of operation of the ICW system is proposed

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 preceeding 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 since the PC/M returns the ICW system to its design condition and no Technical Specification change is required.

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

CONDENSER OUTLET TUBESHEET AND WATERBOX COATINGS

ABSTRACT

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

The condensers and the circulating water system are classified as non-nuclear safety related. A safety evaluation and failure mode evaluation has determined that the modification addressed in this engineering package does not constitute an unreviewed safety question as defined in 10 CFR 50.59. Furthermore, the addition of a protective coating to the condenser outlet tubesheets and waterboxes does not require a change to the plant Technical Specifications.

SAFETY EVALUATION

As noted in FSAR Sections 9.2.1 and 10.4.1, 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.

REMOTE REACTOR VESSEL LEVEL INDICATION

ABSTRACT

This Engineering Package (EP) is for the modification of the Remote Reactor Vessel Level Indicator. This modification will provide more reliable level indication during refueling and reduce personnel radiation exposure since it replaces a temporary system which required more attention for operation.

The modifications considered in this EP are on the Reactor Coolant System. The connections are designated as nuclear safety related and seismically qualified because 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 non-safety, seismic design. 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 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 Part 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question.

REACTOR HEAD TORUS RING MODIFICATION

ABSTRACT

This engineering package covers the modification of the reactor head torus ring. The modification of the reactor head torus ring will enable the ring to be used as an air distribution header for the stud tensioner tuggers.

The reactor vessel head lifting rig assembly including the torus ring is non-safety related. The lifting rig is operated near safety-related equipment including the reactor vessel. Failure of the torus ring during operation of the lifting rig could potentially damage fuel or nearby safety-related equipment. Seismic design criteria as well as the requirements of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants", are applicable to the subject modification. Due to the factors mentioned above, 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 impiementation 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 added nozzles are welded and made as part of the pipe assembly which does not perform a safety-related function.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since the modification does not require any revision to any technical specifications.

The reactor vessel head lifting rig assembly including the torus ring is non-safety related. The lifting rig is operated near safety-related equipment including the reactor vessel. Failure of the torus ring during operation of the lift rig could potentially damage fuel or nearby safety-related equipment. Seismic design criteria as well as the requirements of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants", are applicable to this modification. Due to the factors mentioned above, quality-related requirements are 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.

REPLACEMENT OF SAFETY RELATED BATTERIES 2A AND 2B

ABSTRACT.

This Engineering Package covers the modifications to the Safety Related Station Batteries 2A and 2B 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 15% 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 Nuclear 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.

SUPPLEMENT 1

Supplement 1 removes the holdpoints that were established (environmental and seismic reports received), adds and revises calculations and adds vendor drawings/manuals to EMDRAC system. The original safety evaluation is not affected by this supplement.

SUPPLEMENT 2

Supplement 2 incorporates CRN's ("as found" field dimensions, seismic qualification note, and substitution of cables), revision to drawing list, revised vendor "EQ and Seismic Report" and additional calculations. 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 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 Nuclear 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 Specificatons 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.

PRESSURIZER INSTRUMENT NOZZLE REPLACEMENT

ABSTRACT

This Engineering Package provides the design for the replacement of the four, one inch, steam space pressurizer instrument nozzles. The existing nozzles have been fabricated from a heat of Inconel 600 that has been shown to be susceptible to a form of intergranular stress corrosion cracking (IGSCC). An analysis of the different environments experienced by the nozzles fabricated from this heat has determined that the nozzles with the greatest potential for development of IGSCC are those located in the pressurizer steam space. The replacement nozzles are identical in form, fit and function to the original nozzles with the exception that specific parameters for the Inconel 600 material are more closely controlled to significantly reduce susceptibility to IGSCC.

This Engineering Package is classified as nuclear safety related since it replaces the steam space instrument nozzles which are attached to a safety related component, the pressurizer, and are part of the reactor coolant boundary. The safety evaluation has shown that this Engineering Package does not constitute an unreviewed safety question ncr does it require a technical specification change. Therefore, prior NRC approval is not required for implementation of this PC/M.

This Engineering Package has no adverse impact on nuclear plant safety and/or operation.

Revision 1

This revision removes the construction hold point for material approval, changed the minimum heat treatment temperature from 1800 to 1750 degrees Fahrenheit in order to obtain the minimum ASME Section II yield strength requirements for the SB-166 portion of the replacement nozzle, and modified the ALARA/implementation statements to incorporate plant comments. The safety evaluation has shown that this revision to the Engineering Package does not constitute an unreviewed safety question nor does it require a technical specification change. Therefore, prior NRC approval is not required for implementation of this PC/M.

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 basis for any technical specification is reduced.

This engineering package replaces the four steam space pressurizer instrument nozzles with identical nozzles in design, dimensions, weight, and ASME Section II material specifications. This modification is considered to be 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 is not increased since the replacement of the nozzles will not impact the operation of the pressurizer, affect downstream instrumentation or affect the parameters measured by such instrumentation.

(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 the pressurizer and the new nozzles are equivalent in design.

(iii) This modification does not change the margin of safety as defined in the bases for any Technical Specification because the pressurizer nozzles are not included in any Technical Specification bases.

Implementation of this P/CM 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 P/CM is not required.

RCA PROTECTIVE CLOTHING BINS

ABSTRACT

This engineering package is being issued in response to a request from Plant Mechanical Maintenance. This package will provide the engineering documentation required for the installation of protective clothing bins in the Reactor Auxiliary Building (RAB) and the Fuel Handling Building (FHB). The bins are being installed to provide convenient locations for distribution of protective clothing, and to replace mobile carts currently used.

The protective clothing bins do not perform or affect any safety related function. However, this PC/M is classified Quality Related to provide the Q.C. inspections necessary to ensure the location and installation of the bins are in accordance with the provisions of this engineering package. Quality Related requirements are applied to this modification.

This PC/M does not constitute an unreviewed safety question. The implementation of this PC/M does not require a change to plant technical specifications. This modification does not affect plant operations or safety. Based on the above, implementation of this PC/M does not require prior NRC approval.

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 report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The protective clothing bins do not perform or affect any safety related system or function. However, this PC/M is classified as quality related to ensure Q.C. inspection of the installation.

Consequently, the storage bins and support structures have been analyzed for the design basis conditions specified in the FSAR and Quality Related design requirements have been implemented, thus assuring the integrity of the installation.

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 required to shut down the plant and monitor the plant in a safe shutdown condition.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the protective clothing bins perform no safety function and no changes have been made to any operational design. Failure of the support structures could not occur since the modification has been designed for the design basis conditions.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification since installation of the protective clothing bins does not effect the basis for any technical specification. The implementation of this PC/M 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 PC/M is not required.

TSI THRUST BEARING PROBE RELOCATION

ABSTRACT

This Engineering Package (EP) is for the relocation of the Turbine Generator thrust bearing probes from their existing location to the original Westinghouse probe location. These probes function as position detectors, that is, to monitor the shifting of the rotor with respect to the thrust bearing.

This EP is classifed as non-safety related since these probes neither perform any safety function nor do they interact with safety related equipment. The safety evaluation has determined 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 notification for implementation of 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 EP is for the relocation and replacement of thrust bearing Bently Nevada probes, with similar shorter length probes, from their existing location to the original Westinghouse probe location. The modification implemented via this EP neither performs any safety function nor does it interact with safety related equipment, therefore this package is classified as non-nuclear safety related.

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 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 Turbine Supervisory Instrumentation 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.

(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 Specification.

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 or a change to the Plant Technical Specifications and prior Commission approval for the implementation of the PCM is not required.

MISCELLANEOUS SNUBBER MODIFICATIONS

ABSTRACT

This Engineering Package (EP) provides the engineering and design information for typical modifications to snubbers which may be required as a result of the Inservice Inspection findings. The anticipated typical modifications are expected to be replacement of the existing snubber with a snubber, on a one-for-one basis, of equivalent capacity of the same or a different style (e.g., replacement of a Pacific Scientific snubber with an equivalent Anchor/Darling snubber).

This EP has been classified as Safety Related because the snubber modifications may affect nuclear safety related piping systems. The safety evaluation has determined that this EP does not involve an unreviewed safety question, and implementation of the EP does not require a change to Plant Technical Specifications. Therefore, prior NRC approval is not required for implementation of this EP.

Modifications other than the typical ones shall be reviewed individually to determine if they involve an unreviewed safety question as defined by 10CFR 50.59 or if they will affect any Technical Specifications. Documentation of these reviews shall be included in revisions to this EP.

Modifications performed under this EP shall be documented via Change Request Notices (CRNs) and/or revisions to this EP.

A final revision may be issued, if deemed necessary, after the implementation of this PCM to include the following: a summary of all the modifications included in the project scope, affected Support/Restraint Mark Numbers, documents, affected drawings, TEDB, and changes related to other sections of this EP.

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 basis for any Technical Specification is reduced.

This EP provides typical modifications to snubbers which may be found necessary during the Inservice Inspection of snubbers. These typical modifications are limited to the replacement of snubbers or components with snubbers and components of equivalent load rating. Since these typical modifications may affect Nuclear Safety Related piping systems, this EP is classified as Nuclear Safety Related.

This EP has been determined not to involve an unreviewed safety question, based on the following:

(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 restraints for the piping will remain functionally identical to the existing configuration. In addition, since the restraint configurations are not changed, all previous analyses and conclusions are still valid.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in a safety analysis report is not created because the system remains functionally identical to the configuration depicted in the existing stress analysis of record. Also, the affected restraints have been qualified to the same code requirements as those they replace.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this modification because the replacement components utilized perform the same restraining function as those they replace.

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 and prior Commission approval for the implementation of this PCM is not required.

LOAD CENTER XFRMR VALVE PACKING MODIFICATIONS

ABSTRACT: Load Center Transformers (2B2, 2A5, 2B5) radiator shut-off valves (Tranter Valves) have leaked silicone fluid. This change documents modifications performed to the Tranter Valves' (36 Total) stem packing to prevent the leakage.

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?

INSTRUMENT AIR DRYER AFTERFILTER ISOLATION VALVES AND BYPASS LINE

ABSTRACT

This Engineering Package covers installation of isolation valves and a full flow bypass line with a valve on the Instrument Air (IA) dryer afterfilter. These modifications will provide for isolating the afterfilter while maintaining IA System operation. Provision for isolation capabilities is required to facilitate installation of the IA upgrade modification (PC/M 051-286), during plant operation.

This modification covers equipment located in the IA System which is classified as Non-Nuclear Safety Related. Based on the failure modes evaluation and 10CFR 50.59 review, this modification does not impact any safety related equipment or functions, is not relied upon for any accident prevention or mitigation, and does not impact plant safety. This EP does not constitute an unreviewed safety question and is correctly classified as Non-Nuclear Safety Related. In addition, this modification does not require changes to the Technical Specifications. Implementation of this modification, therefore, does not require prior NRC approval.

SAFETY EVALUATION

The subject modification provides for installation of inlet and outlet isolation valves and a full flow bypass line with a vlave to the IA System afterfilter. As defined in Section 9.3 of the FSAR, this system is considered Non-Nuclear Safety Related, Quality Group D and is not required to perform a safety function. These modifications are therefore considered Non-Nuclear Safety Related. Based on the failure modes evaluation, as provided in the Design Analysis, failure of the IA System has no effect on Nuclear Safety.

Title 10 of the Code of Federal Regulations Part 50.59 allows changes without prior commission approval provided the proposed change does not involve an unreviewed safety question or require changes to the Technical Specifications. The proposed change does not involve and unreviewed safety question because:

- 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 occurrences or consequences of analyzed accidents remain unchanged.

- The possibility of any 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 Specification has not been reduced.

  This change has no effect on any existing Technical Specifications.

In conclusion, this modification is acceptable from the standpoint of nuclear safety since it does not involve an unreviewed safety question, as defined by 10 CFR 50.59, and does not require changes to the Technical Specifications. Implementation of this modification does not require prior NRC approval.

CONDENSER EXPANSION JOINT IMPINGEMENT PLATE MODIFICATION

ABSTRACT: The existing impingement plate design is inadequate for satisfactory long-term performance. Welded attachments on the plates have continuously failed, causing plates to fall on the damage condenser tubes. The new plate design will involve no welding and will allow plate installation which will prevent any further failure.

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?
____	✓	A change to procedures as described in the FSAR?
____	✓	A test or experiment not described in the FSAR?
____	✓	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?
____	✓	Will the consequences of an accident previously evaluated in the FSAR be increased?
____	✓	May the possibility of an accident which is different than any already evaluated in the FSAR be created?
____	✓	Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
____	✓	Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
____	✓	May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
____	✓	Will the margin of safety as defined in the bases to any technical specification be reduced?

CABLE SUPPORT STRUCTURE CONNECTION MODIFICATION

ABSTRACT: Reinstallation of the reactor head cable support structure at the end of each refueling outage requires the replacement of connection bolts. These bolts, specified on drawing 2998-B-791, SH.11, are unique and must be special ordered. This modification will specify standard connection bolts which are readily available. This modification does not involve an unreviewed safety question and no technical specification changes are 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?
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?

GROUTING OF MASONRY BLOCK WALLS

ABSTRACT

In the course of preparing the Fire Protection Appendix of the Unit 2 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 recently performed and concluded that the walls are not fully grouted.

This Engineering Package (EP) provides the details/requirements for pressure grouting the voids in block walls 127, 128, 129A, 129B, and 137. The remaining walls will be addressed in a future revision to this EP.

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 3/4.6.12 will no longer be required for the walls modified. This EP is classified Quality Related since all of the walls involved are seismically designed and 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 not be 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 pressure grouting the walls so that they are in conformance with the design bases established in Subsection 3.11.2 of the St Lucie Unit 2 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 failure of the walls could damage safety-related equipment.

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 these walls are seismically designed, no accidents due to structural failure are postulated. The only other type of accident potentially associated with these walls 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 postulated which involve these block walls.

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 seismic integrity of the 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 3/4.6.12 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 PC/M 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 PC/M is not required.

STEAM GENERATOR TUBE PLUGGING/CE DESIGN PLUGS

ABSTRACT

This PCM documents Engineering review and concurrence for the use of Combustion Engineering expanded type plugs in the St. Lucie Unit 2 steam generators. This PCM also provides the information necessary to as-build affected documents.

Since the steam generator tubes are nuclear safety related, the tube plugs described herein are also nuclear safety related.

Based upon a failure mode evaluation and 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. The modification has no adverse affect on plant safety or operability.

SAFETY EVALUATION

This modification involves documenting the maintenance practice of plugging steam generator tubes. Steam generator tubes are nuclear safety related, therefore this engineering package is classified as nuclear safety related. The PC/M provides engineering concurrence for the use of the Combustion Engineering expanded tube plug design (previously utilized on the Unit 1 steam generators), the required 50.59 review of the modification, and the information required for update of affected documents.

10 CFR 50.59 allows a change to a nuclear facility without prior NRC approval if an unreviewed safety question does not exist and if changes to Technical Specifications are not involved. The following arguments demonstrate that an unreviewed safety question does not exist relative to this modification:

i) The probability of occurrence of a design basis accident is not increased since this modification does not decrease the design margin of the RCS pressure boundary (the tube plugs meet or exceed all design requirements for ASME Section III, Class 1 components).

ii) The consequences of a previously postulated design basis accident are not made more severe for the same reasons given in (i) and since no existing accident mitigation equipment or systems are altered by this modification.

iii) The possibility of an accident of a different type than previously addressed in the FSAR does not exist since no new systems or equipment are introduced by this modification. Failure of a tube plug would be no more severe than a steam generator tube rupture, a previously evaluated condition. Therefore, no new accidents are created.

iv) The margin of safety as defined in the basis for any technical specification is not reduced since the total number of tubes plugged in the steam generators following this modification is less than assumed in the Cycle Four Reload Analysis.

Since the above arguments demonstrate than an unreviewed safety question does not exist, and since a revision to the Technical Specifications is not required, the addition of the Combustion Engineering tube plugs to the Unit 1 steam generators does not require prior NRC approval.

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2A/B SPARE STEAM GENERATOR INSTRUMENT NOZZLES CLOSURE MODIFICATION

ABSTRACT: The existing blank flange connections have degraded, resulting in steam leaks. In order to prevent further leakage, and since the nozzles are no longer required, the flanges will be removed and welded caps will be installed. No unreviewed safety questions exist as defined by 10 CRF 50.59, and no Technical Specifications are affected by this modification. Therefore, prior Commission approval is not 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?

INCORE INSTRUMENTATION (ICI) PLATE MODIFICATION

ABSTRACT

This Engineering Package covers repair modifications to the Incore Instrumentation (ICI) Plate Assembly to correct local deformations in the plate. The major features of this package include the repair process and the acceptability of the repaired ICI plate for continued service. Included is information pertaining to the repair tooling, repair process procedures, repaired plate acceptance criteria and an assessment of the safety impact of continued use of the repaired ICI plate.

Work performed under this Engineering Package has been classified as Quality Related. This classification was selected because the repair work discussed herein was conducted over the reactor vessel and its internal components. Further, during operation the ICI plate has the potential to interact with safety related components. The ICI plate assembly itself, however, is a non-safety related component since it neither prevents nor mitigates the consequences of accidents. Changes in the ICI plate configuration resulting from the repair will have no effect on its design function.

Based on a Failure Modes and Effects Analysis (FMEA) and 10CFR50.59 review, the repair modifications implemented by this engineering package are acceptable from the standpoint of nuclear safety as it does not involve an unreviewed safety question and does not change plant Technical Specifications. As such, prior Nuclear Regulatory Commission approval is not mandatory to implement this Engineering Package.

<u>Supplement 1</u>

This supplement revises only the design interface record of the package to add two signatures. This supplement does not affect the conclusions of the safety evaluation that the change does not involve an unreviewed safety question and does not change plant Technical Specifications.

SAFETY EVALUATION

This Engineering Package has considered the safety related consequences of the repair process proposed to straighten the deformed Incore Instrumentation (ICI) Plate Assembly at St. Lucie Unit 2 and its subsequent return to service. The package has been classified Quality Related due to the location of the repair (over the reactor vessel) and the potential for interaction with safety related components. The ICI plate assembly itself, however, performs no safety related function. No unreviewed safety question or Technical Specification change was identified as a result of these considerations. As such, it has been concluded that the repair of the plate and its return to service can proceed as planned.

In order to straighten the deformed ICI plate, it was necessary to develop repair tooling which could be used to reverse the load path on the plate that originally caused the deformation. This tooling took the form of various pieces of hardware which could be used to grip or push on the plate and apply the necessary loads. The loads applied were aimed at either straightening the plate in a global sense by pushing down at its center or by loading the locally deformed areas to bring them back into proper alignment. In practice, only the center push tool had to be employed. All jacking load paths for the plate bending process were through the ICI plate itself and the UGS lift rig and did not involve any other plant structures or reactor vessel internal components. Straightening of the ICI plate was actually accomplished through two center push operations. Each operation was accomplished over a series of controlled increments.

In making its determination regarding the safety aspects of this process, C-E considered containment integrity, shutdown cooling system operation, fuel damage, impact on plant structures, loose particles due to grinding operations , heavy loads, fire hazards and the acceptability of repair hardware materials. The details of this evaluation were transmitted (L-MPS-87-033) to FPL prior to the repair and are included with this package in Attachment 7.

After straightening the ICI plate, it was discovered that one of the T-brackets which supports incore detector guide tube runs interfered with the outside wall of an adjacent Control Element Assembly (CEA) guide tube shroud during the lowering of the ICI plate. The interference was sufficient to prevent the ICI plate from seating completely.

A plan to reduce the T-bracket deformation and eliminate the interference was developed. The corrective action called for a suitably rated chain-fall (or come-along) to be attached to the T-bracket near its top and to a bracket mounted on the southwest corner of the "A" steam generator biological shield wall. FPL determined that this bracket was capable of withstanding a load substantially greater than the maximum 5000 pound load limit for the T-bracket corrective procedure. To faciliate bending of the T-bracket, a relief slot was cut in the vertical leg of the bracket. The slot allowed the T-bracket to undergo a one-time local plastic deformation when tensioned with the come-along. In this manner, the T-bracket was pulled into a more upright position eliminating the interference.

To support the feasibility of this procedure to straighten the T-bracket, a test was performed on a similar structure at C-E's Windsor Test Facility on Nov. 6, 1987. A T-bracket was deformed approximately 1/4 to 1/2 inch by applying a load of less than 3500 pounds. In the St. Lucie Unit 2 plant, the actual load was applied at an angle of approximately 35° above vertical because of access limitations in the work area. The C-E test demonstrated that the T-bracket could be deformed without affecting adjacent ICI plate structures (e.g., guide tube clusters).

The in plant procedure was performed with the ICI plate raised and supported by the UGS lift rig and with the ICI plate compressed by the center push tool applying a load of approximately 8000 pounds. An analysis was performed to evaluate areas of potential concern:

a) Motion Within the Lift Rig

Analysis showed that the vertical component of the bending force was less than the seating force at each leg of the lift rig. Therefore, lift off was not anticipated and tipping could not occur

since the moment balance about a potential tipping point was stable.

b). Lift Rig Column Bending (Lift rig in bending and shear)

Analysis showed the bending stress to be below the minimum material yield strength of 30 Ksi. As such, column bending was not a concern. Combined cable side and vertical loading due to dead weight and center jacking stabilizing loads were combined. The total stress in the lift rig leg column was < 8000 psi (i.e., combined bending and axial shear).

c) UGS Stability

A moment balance about the base of the UGS showed that the system was stable and would not tip under the maximum applied 5000 pound pulling load.

d) Acceptability of Slotting T-Bracket

The flow loads in the relatively isolated upper head region were reviewed and found to be below those required to adversely impact the slotted T-bracket.

The pull will not affect the ability of the T-bracket to perform its design function of supporting the instrument guide tubes in any way. Further, the slotted and straightened bracket will not interfere with the function of any other components in the area of the ICI plate.

During the repair, it was observed that the surface of the ICI plate covered with metal chips in the localized area in which grinding of the locating pins took place. The chips were characterized by FPL divers as ranging from fine particles to slivers of between 1/8 - 1/4 inch in length (maximum) with a thickness of less than 25 mils on average. The surface density of the chips was reported as approximately 10 - 15 chips per square inch in the grinding area falling off to sparse coverage in areas outside the grinding location.

If the chips enter the reactor vessel, the potential for interference with control rod motion and possibility of degraded core thermal margin and/or fuel damage can be postulated. In order for the small metal chips to impede control rod motion they would have to become wedged between the control element assembly (CEA) and the guide tube. The diametral gap between the stainless steel sleeve and the CEA is 54 mils and the metal chips are less than 25 mils. Therefore, it is very unlikely that any chips that happened to fall into the guide tube could cause the CEA to jam. Degraded core thermal margin could only occur if a significant flow blockage were to occur as a result of the metal chips. Critical heat flux tests have been run in which flow blockages were simulated. Blockage of 11 out of 34 subchannels of a 5X5 test section showed little effect on bundle critical heat flux capability compared to similar tests without the blockage. Considering the distribution and sizes of chips observed on the ICI plate, it is concluded that is is not possible for a sufficient quantity of chips to agglomerate in one local region of the core and block the inlet of more than 11 subchannels. Therefore, no premature DNB due to subchannel inlet flow blockage is expected. It is conceivable that a small number of chips could become trapped at other spacer grids above the core inlet. Again, the total blockage that might occur would be extremely small for any one subchannel and would not be expected to adversely impact DNB margin.

The potential for fretting-induced fuel failure due to the chips entering the active fuel region, however, does exist. It is not likely, though because of their small size. Any fretting-induced fuel failures would occur gradually over time, and become apparent to the reactor operator in the form of higher coolant activity levels. Technical Specifications on coolant activity level preclude plant operation at levels which would pose an undue risk to the health and safety of the public. Therefore, with regard to the issue of control rod motion and DNB performance and fuel damage, the metal chips found on the ICI plate do not create the possibility of an accident or impact the operation of equipment important to safety.

In addition to the repair process, this Engineering Package has considered the acceptability of the repaired ICI plate assembly for continued service. Evaluation of the loads experienced by the plate during its deformation, during the repair (straightening), process and during operation ensure that the plates design function were not adversely impacted.

Using a finite element model of the ICI plate, the loads necessary to cause the observed deformation were determined. This information was used to evaluate the strain levels within the plate and its acceptability for repair. Results of this investigation indicated that the plate was not strained beyond limits which would preclude a repair process to straighten the deformed areas. This same finite element model was employed to determine the loads and their points of application to be applied in order to straighten the ICI plate. The strain levels imposed during the straightening process were also evaluated to assure that plate integrity would not be compromised. Using the as-repaired dimensions of the ICI plate, its acceptability for re-installation in the reactor vessel was assessed along with its ability to carry out its design function.

The safety evaluations discussed above were conducted to determine whether any unreviewed safety question or change to Technical Specifications was involved in the proposed ICI plate assembly repair process or in its return to service. The overall conclusions, which are elaboratored below, indicate that the use of the special tooling and procedures developed for this repair effort as well as the continued use of the repaired ICI plate assembly does not involve an unreviewed safety question or require a change to the plant Technical Specifications. Specifically, the safety evaluation conclusions are that the ICI plate repair and continued use does not:

1) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

As indicated in Attachment 7, the repair process was conducted in such a fashion that containment integrity and shutdown cooling system operation were unaffected, therefore, the safety functions for both systems were not impacted. Furthermore, examinations of the possibility of dropping tools into the reactor vessel or of impacting Heavy Loads considerations during the repair process were conducted. The repair process was found to not introduce any additional loads to the UGS lift rig that exceed its design limits. Heavy Loads, therefore, were not impacted in any way by the ICI plate repair. Furthermore, since the repair work was conducted above the UGS, which was in its normally seated position, there was no increase in the probability of occurrence of fuel damage due to potentially dropped tooling.

The ICI plate and fine alignment pins are passive components and do not provide a safety function. The modification of the fine alignment pins could, however, increase the lateral movement of the ICI plate, resulting in some slight contact of the ICI instrumentation thimbles and HJTCs. Even if contact did occur, it would not affect the function of these instruments. The potential for increased wear of any of these components was considered insignificant because flow loads in the isolated upper head region are small or non-existent and the ICI plate assembly is heavily damped by 56 incore instruments and two HJTC probes. Instrumentation guide tube cluster engagement with the reactor vessel head also provides damped restraint. Therefore, the probability of unanalyzed equipment malfunction is not increased, nor is the probability of an occurrence or consequences of an accident previously evaluated in the safety analysis report increased.

Evaluation of the load to be applied during the pull to eliminate intereference between the T-bracket and the CEA shroud were well within the capability of the UGS lift rig and "A" steam generator biological shield wall. The modified T-bracket is capable of performing its design function and will not adversely impact any other components in the area of the ICI plate. As such, the probability of occurrence or the consequences of accidents evaluated in the safety analysis report with respect to use of the lift rig or the function of the ICI plate are unaffected.

As stated in Section 3.9.5.4.2 of the St. Lucie Unit 2 UFSAR, the ICI plate assembly by itself is not a safety-related component since its satisfactory performance does not prevent accidents nor mitigate the consequences of accidents that could cause undue risk to the health and safety of the public. Nothing in the repair process nor the continued use of the ICI plate has any impact on the UFSAR statement.

2) Create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report;

In assessing the possibility of creating new accidents or malfunctions as a result of the repair process, examinations of pool contamination, fire hazards, loose parts and impacts on crane handling design capability were conducted. The findings indicate that the grinding operations performed on the two ICI plate locating pins were executed in a controlled manner. Particles that could potentially have escaped the vacuuming operations would be of insufficient size to create the possibility of an accident or impact operation of any safety related equipment. Hydraulic fluid used in conjunction with the plate straightening tools has been found acceptable for unrestricted use at St. Lucie Unit 2. With respect to additional fire hazards being generated due to the repair process, only the short time use of a cutting torch (under water) was necessary to cut a slot in the T-bracket. This operation was controlled by FPL in accordance with their fire protection procedures. No fire hazards were, therefore, created. The size and mass of all repair tooling was examined and found to be well within the design loading capacity of the tool handling cranes. Protection against inadvertently losing tooling was ensured through use of lanyards and administrative controls.

The possible affects of the modification to the ICI plate fine alignment pins could have some slight effect on the potential for interaction between the incore instrumentation thimbles and HJTCs and their respective guide tubes. The slight contact that may potentially occur does not adverseley impact the ability of these instruments to provide their design function. The ICI plate will not interfere with other reactor vessel internal structures in its repaired configuration. Therefore, the possibility of an unanalyzed accident or malfunction is not created by this modification.

Evaluation of the load to be applied during the T-bracket pull indicated that the UGS lift rig will not tip nor will its columns deform. Similarly, the UGS itself is stable and cannot be tipped by the load to be applied. The integrity of the T-bracket is not compromised by the slot and, in its new orientation, it does not interfere with other components. The design function of the T-bracket in supporting the instrumentation guide tubes will, therefore, be met.

As stated above, the ICI plate assembly does not have any design function related to the prevention or mitigation of accidents. The evaluations of the as-repaired ICI plate indicate, furthermore, that its structural integrity is not compromised and that it can continue to provide its design function. In the extreme, if a hypothetical failure of the plate (complete through plate crack) were assumed in any of the repaired areas, no adverse consequences have been identified by the Failure Modes and Effects Analysis. The CEA shrouds extending from the upper guide structure pass through the plate in numerous locations over its surface area and would act to hold any separated segments in place. Further, since the upper head region in which the ICI plate resides is a relatively low flow area, the probability that any adverse movement of the plate or segments thereof would occur is also extremely low. Being a highly isolated region from the remainder of the reactor vessel internals will preclude the generation of any postulated loose parts from adversely affecting plant operation.

3) Reduce the margin of safety as defined in the basis of any technical specification;

An examination of the repair process has not led to the identification of any Technical Specifications that will be impacted. In addition, the modifications to the ICI plate, locating pins and T-bracket during the repair do not affect the Technical Specification margins for safety because these structures do not serve any safety related function. As a result, the margin of safety in the bases in plant Technical Specifications will remain unchanged.

Based on the evaluations discussed above, C-E concludes that the proposed corrective actions do not involve an unreviewed safety question or a change to the plant's Technical Specifications. Further, Combustion Engineering has found no reason to preclude the repaired ICI plate assembly from being returned to service and being able to carry out its design function for the remainder of the St. Lucie Unit 2 design life.

St. Lucie Unit 2 Incore Instrumentation Plate Repair Program
Safety Evaluation

Summary:

Combustion Engineering (C-E) has considered the safety related consequences of the repair proposed to straighten the deformed incore instrumentation (ICI) plate assembly at St. Lucie Unit 2. In making its determination, C-E considered containment integrity, residual heat removal system operation, fuel damage, impact on plant structures, loose particles due to grinding operations, heavy loads, fire hazards and the acceptability of repair hardware materials. No unreviewed safety question nor technical specification change was identified as a result of these considerations. As such, C-E concludes that the repair can proceed as planned.

Discussion:

In order to straighten the deformed ICI plate it was necessary to develop repair tooling which could be used to reverse the load path on the plate that originally caused the deformation. This tooling is in the form of various pieces of hardware which can be used to grip or push on the plate and apply the necessary loads. The loads applied are aimed at either straightening the plate in a global sense by pushing down at its center or by loading the locally deformed areas to bring them back into proper alignment. All jacking load paths for the repair process are through the ICI plate itself and the upper guide structure (UGS) lift rig and do not involve any other plant structures or reactor vessel internal components.

An evaluation of the repair hardware or the repair process for any adverse safety related consequences was conducted in the following areas:

<u>Containment Integrity</u>

Neither the repair process nor the results of the repair will impact containment integrity in any way. The equipment required is self contained and can be powered from sources inside containment normally used during refueling or plant maintenance operations. No outside containment support is required for the operation of repair hardware. Normal access hatches will be used only to bring in and remove equipment. No breaches of containment integrity, therefore, will be required before or during the repair.

Shutdown Cooling System Operation

The repair process does not impact the shutdown cooling system (SDC) operation. The SDC system will be in its normal refueling configuration throughout the repair process. The repair process will have no direct interaction with the SDC system. All repair actions will be preformed in the refueling pool above the normal elevation of the ICI plate and above the reactor vessel.

Fuel Damage

There is no increased potential to damage fuel during the repair process. The size and mass of repair tools and equipment are within the envelope of tools and equipment normally used for work in this area. The repair work will all be conducted above the upper guide structure which is located so as to preclude any dropped equipment from reaching the core region. As such, the repair evolution will not result in any risk of damage to the fuel. Further considerations regarding the potential for fuel damage are discussed in relation to heavy loads below.

Loose Particles (Grinding)

Grinding of any ICI plate components as part of the repair evolution will be done in a controlled manner. A vacuum system will be used to minimize the potential for particles entering the refueling pool water and possibly making their way into the reactor vessel or primary coolant system. Any random particles not captured by the vacuum system would be of insufficient size or volume to create a safety hazard.

Impact of Tools on Any Plant Structure

The size and mass of the repair tools are within the capabilities of the cranes to be used in support of the repair process. All tools are hand-pump operated metal working tools which operate independent of other plant systems. The recommended repair process does not involve any repair equipment coming into contact with any of the plant structures other than the ICI plate and the UGS lift rig. Tools will be administratively controlled and procedures will include the use of lanyards to assure that no tools are inadvertently lost in the refueling pool. No adverse impact, therefore, is expected as a result of the repair process or any postulated tool or equipment failure.

Acceptability of Material Being Put in Pool Water

No material will be introduced into the refueling pool water that is not typically used at refueling outages. Repair tooling is fabricated from carbon steel and stainless steel. The hydraulic fluid that will be used for the repair tooling has been evaluated and found to be acceptable for unrestricted use in C-E designed NSSSs. The potential for contamination of the water or for adverse reaction with reactor system components is, therefore, precluded.

Heavy Loads

The Upper Guide Structure (UGS) lift rig was designed and load tested in accordance with the guidance of NUREG-612 and ANSI N14.6-1978. The repair process does not introduce any additional loads from repair tooling which exceed the lift rig design limits. Abnormal load transfer through the lift rig structural components due to the repair process has been reviewed and determined not to adversely impact structural integrity. The recommended repair process will limit the applied loads to the ICI plate and the lift rig and will not result in any loads being applied to fuel, core support components or plant structures.

The potential consequences of dropping heavy loads into the open reactor vessel have been considered. The UGS will be in its normally seated position and, therefore, heavy load drop is not an issue pertinent to this repair. Only the ICI plate, which weighs less than 10% of the UGS, will be within its normal range of raised positions for the repair evolution. Should the ICI plate drop, the impact load would be taken up by the reactor vessel flange and not the internal components. A drop such as this is conservatively bounded by reactor vessel head drop analyses which have been shown not to have any fuel damage consequences. The long slender ICI thimbles which extend down from the plate are guided in the UGS and fuel regions by surrounding tubes. These tubes would preclude damage to the fuel even in the unlikely event the ICI plate were to drop.

Fire Hazards

The repair program does not involve the use of any flammable substances or open flame. All of the repair tooling is prefabricated outside containment. The repair process, therefore, will not cause any fire hazards.

Conclusions:

A safety evaluation was conducted to determine whether any unreviewed safety question or change to technical specifications is involved in the proposed incore instrumentation plate assembly repair process. The overall conclusions, which are elaborated below, indicate that the use of the special tooling and procedures developed for this repair effort will not involve an unreviewed safety question or require a change to the plant technical specifications. Specifically, the safety evaluation conclusions are that the ICI plate repairs does not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

   As indicated previously, the repair will be conducted in such a fashion that containment integrity and shutdown cooling system operation will be unaffected, therefore the safety functions for both systems will not be impacted. Furthermore, examinations of the possibility of dropping tools into the reactor vessel or of impacting the Heavy Loads Analysis during the repair process have been conducted. The repair process was found to not introduce any additional loads to the UGS lift rig that exceed its design limits. The Heavy Loads analysis will, therefore, not be impacted in any way by the ICI plate repair. Furthermore, since the repair work will be conducted above the UGS, which will be in its normally seated position, there is no increase in the probability of occurrence of fuel damage due to the dropped tooling.

2. Create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report:

   In assessing the possibility of creating new accidents or malfunctions as a result of the repair process, examinations of pool contamination, fire hazards, loose parts and impacts on crane handling design capability were conducted. The findings indicate that the grinding operations of any ICI plate components will only be executed in a controlled manner that minimizes the potential for particles entering the pool or reactor coolant system. The vacuum system that will be used will ensure that any random particles that may escape will be of insufficient size to create any safety concern. Furthermore, the hydraulic fluid that will be used has been found acceptable for unrestricted use at St. Lucie 2. With respect to additional fire hazards being generated due to the repair process, it was found that no flammable substances or open flames will be used. No fire hazards will therefore be created. The size and mass of all repair tooling have been examined and found to be well within the design loading capacity of the tool handling cranes. Protection against inadvertently losing tooling has been ensured through use of lanyards and administrative controls on all.

3. Reduce the margin of safety as defined in the basis of any technical specification;

   An examination of the repair process has not led to any technical specifications that will be impacted. As a result, the margin of safety for plant technical specifications will remain unchanged.

Based on the information set forth above, C-E has concluded that it is acceptable to proceed with the incore instrumentation plate assembly repair as planned.

AS-BUILT CCW SUPPORT

ABSTRACT: NCR 2 - 123 has been generated to resolve discrepant field conditions for component cooling system restraint Mark No. CC-2074-44. This Engineering Package documents the evaluation performed for the 'as found' condition and provides a mechanism for permanent plant drawing update. This modification does not involve an unreviewed safety question and no technical specification changes are 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?

ADD A WELD ON LINE WM-B-80

ABSTRACT: In order to clean blockage in the vicinity of the BAM pump branch connections into the header on WM-B-80, pipe was cut, removed, and cleaned. Reinstallation required a new weld with a new weld number (FW-8A on WM-B-80). Piping isometric WM-B-11 requires a revision to incorporate this new weld number.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?

CHECK VALVE V27101 REPLACEMENT

ABSTRACT: Existing check valve V27101 by Rockwell has been damaged and is to be replaced with an equivalent valve by Kerotest, on FPL PO 87630-90117.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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	
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	
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?

RCS INSTRUMENT NOZZLE INSULATION TEMPORARY MODIFICATIONS

ABSTRACT: Minor Insulation modifications are outlined in this package for five hot leg RTD locations and one pressrizer lower head nozzle location. The changes are required to support requirement of JPE-M-87-112, Revision 0, "RCS Instrument Nozzle Cracking - Justification for Continued Operation."

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?
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?

SPLICE BOXES B2124,34,35

ABSTRACT: This EP documents the acceptability of replacing FD type conduit boxes with larger splice boxes to prevent violating the bend radius of the Raychem splices contained within. This modification does not involve an unreviewed safety question and does not require a change to the Plant Technical Specifications. This is evidenced by the attached Nuclear Safety Evaluation Checklist.

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?

REPLACEMENT OF I-FCV-25-7 AND 8 ACCUMULATOR CHECK VALVES

ABSTRACT: Replacement of the check valves for the Instrument Air supply to the accumulators for FCV 25-7,8 (Containment Vacuum Relief). The replacement valves are NUPRO SS-4CP2-1 purchased on FPL PO C38610 98267.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?

CEA MG SETS LOCKOUT RELAY

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?

EXCORE START/UP AND CONTROL CHANNEL CIRCUIT MODIFICATION

ABSTRACT

This engineering package covers modifications to the Excore Start/up and Control Channel Linear Power Circuits. The major feature of this package is the modification of the feedback loop on the linear power subchannel inputs to increase the channel gain. This modification will compensate for the lower values of leakage flux at the excore detectors which result from the current St. Lucie Unit 2 Fuel Management Program.

Because the Excore Start/up and Control Channels are non-safety related and since this modification does not impact any safety related systems, all work covered by this engineering package is classified as non-safety related.

Based on a 10CFR50.59 safety evaluation this modification does not affect plant safety or operation, nor does it involve any unreviewed safety questions or require changes to the plant Technical Specifications. As such, prior NRC approval is not required to implement this engineering package.

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 change described herein does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

This engineering package has considered the safety related consequences of the modifications proposed to the excore start/up and control drawers. Because the excore start/up and control channels are non-safety related, all work covered by this engineering package is classified as non-safety related.

All of the modifications implemented by this package are confined to the linear amp and summer cards (LASI-2) which are located inside the excore start/up and control channel drawers. The chnages consist of modifying the first stage feedback loop on the two linear subchannels to accommodate the actual, and anticipated future flux levels as seen by the detectors. Section 10 describes the modification in detail. These changes do not impact the functionality of the drawer:

There are no changes to any other equipment interfacing with the excore start/up and control channels.

These changes were reviewed to determine the impact on the existing seismic and environmental requirements with no negative findings.

Based on the above, the modification is confined only to the excore start/up and control drawer, and has no impact on existing analysis or design basis. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety has not increased.

ii. The change does not create the possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report.

The changes will not result in any new functional circuitry being added to the equipment, therefore, the procedures for performing maintenance and calibration for the drawer are essentially identical, and there is no impact on personnel performing maintenance on this equipment.

Based on the above, the change has no effect on existing setpoints, or system operation, and therefore, does not create the possiblity for an accident or malfunction of a different type than evaluated previously.

iii. The change does not reduce the margin of safety as defined in the basis for any technical specification. The change does not result in an increase in the surveillance requirements. In addition, no operational parameter or technical specifications are impacted by the changes in this Engineering Package, therefore, no change to the technical specifications are required.

Because the Excore Start/up and Control Channels are non-safety related and since this modification does not impact any safety related systems, all work covered by this engineering package is classified as non-safety related.

The implementation of this Engineering Package 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 or require changes to the Plant Technical Specifications. As such, prior Commission approval for the implementation of this PCM is not required.

ROSEMOUNT XMTR FT9021

ABSTRACT: Correct a model number discrepancy between installed hardware and affected documents for Rosemount instrument FT 9021. This is in response to RFD-009-87 and NCR 2-047. It is a document 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?
____	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?

REPLACEMENT OF PT-22-23

ABSTRACT: Replace obsolete Fischer & Porter transmitter PT-22-23 (Turbine Bearing Oil Pressure) with new Rosemount 1151GP transmitter. This task is a late addition to REA-SLN-86-079. This changeout 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?
____	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?

EQ LIST REVISIONS - SPARE PARTS

ABSTRACT: The EQ List has an "X" in the SPEER column to designate the components/replacement parts which must be evaluated by engineering prior to ordering. Equipment included in this DEEP has been preapproved by engineering for purchase so the "X" in the SPEER column is being deleted.

This change does not involve a change to the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59 so prior NRC approval is not required for implementing this change. This change has no impact on plant safety or operation since it is being implemented to assure that procurement of replacement parts for EQ components is being done in accordance with the FPL QA program requirements to assure compliance with the provisions of 10CFR50.49.

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?

DRAWING/INSTRUMENT LIST CORRECTIONS REGARDING ICW PUMPS

ABSTRACT: To correct a drawing error (CWD 2998-B-327 SH 882) for a Main Generator relay by Interchanging the contact designations and to correct designations. In the Instrument List and TEDB flow indicating switches used on Lubewater for the ICW pumps on the Instrument list and TEDB. An unreviewed safety question does not exist and there is no change to technical specification 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?

ICW & CW PUMP PACKING REPLACEMENT

ABSTRACT: The existing CW & ICW pump packing contains asbestos. The packing will be changed to an all graphite material. This DEEP provides for this change including guidelines for procurement.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?

EQ DOC PAC UPDATE FOR LIMITORQUE MOTOR OPERATORS

ABSTRACT: Update Limitorque EQ Doc Pac to prohibit use of Marathon 1600 Terminal Blocks throughout the plant and 3M taped splices inside the RCB. The Doc Pac will also be revised to document the de-energization of Limitorque Limit Switch Compartment Space Heaters, and information relating to "T-Drain," Torque Switch Color and Motor Size.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?

NAMCO LIMITS SWITCHES FOR PCV-8801 THRU 5

ABSTRACT: Designate replacement limit switches for PCV-8801, 8802, 8803, 8804 and 8805. Revise affected drawings and documentation. The existing limit switches D2400X to be replaced with limit switches model EA170-11100.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?

DRAWING CLARIFICATION-ENGINEERED SAFEGUARDS CABINET

ABSTRACT: This PC/M clarifies a Unit 2 Control Wiring Diagram to more clearly show the appropriate terminal board numbers for the termination of certain wiring within the engineering safeguards cabinet. Improper interpretation of the terminal board designation by plant personnel has previously contributed to a plant trip. This is merely a drawing clarification. An unreviewed safety question does not exist and there is no change to technical specifications.

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?

PDIS REPLACEMENT

ABSTRACT: Replacement of failed PDS-2216, Barton Model No 288A with a Barton DPIS Model No 288A currently in stores. The replacement Barton 288A switch was procured as spare for Unit No 1 PDIS 02-1, which performs the same function in Unit 1 as the failed PDS-2216 in Unit No 2.

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?

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?

EMDRAC DRAWING 2998-738, REV 4

ABSTRACT

The ASME Sect III Class 2 check valves on drawing 2998-738 Rev 4 were never installed. Hydraulic operated gate valves were installed in place of these valves for the main feedwater system containment isolation function. Since these check valves were subsequently sold, drawing 2998-738 shall be deleted and associated documentation updated accordingly. No technical specifications have been affected and there are no unreviewed safety questions.

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?

FUEL POOL PURIFICATION SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT

ABSTRACT: The existing mechanical seals utilized in the subject pumps are being replaced by a cartridge type mechanical seal. The change will reduce the subject pumps' downtime required for seal replacement and is considered a maintenance enhancement. Also addressed is a clarification of the seal material specified for the subject pumps in the Unit 2 FSAR. This change does not affect any technical specification and there are no unreviewed safety questions.

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 X No ___ 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?

CONDENSATE RECOVERY SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT

ABSTRACT: The existing mechanical seals utilized in the subject pumps are being replaced by a cartridge type mechanical seal. The change will reduce the subject pumps' downtime required for seal replacement and is considered a maintenance enhancement. This change does not affect any technical specifications and there are no unreviewed safety questions.

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?

TURBINE GLAND SEAL SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT

ABSTRACT: The existing mechanical seals utilized in the subject pumps are being replaced by a cartridge type mechanical seal. The change will reduce the subject pumps' downtime required for seal replacement and is considered a maintenance enhancement. This change does not affect any technical specifications and there are no unreviewed safety questions.

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?

DOCUMENTATION CORRECTIONS FOR PS-29-4, 4-1, 4-2

ABSTRACT: Revise the St Lucie Unit 2 Instrument List, and the Total Equipment Data Base to reflect as-built conditions for pressure switches PS-29-4, PS-29-1, and PS-29-4-2. Also revise EMDRAC 2998-13526.

1) The model number for pressure switch PS-29-4 appears incorrectly in the St Lucie Unit 2 Instrument List. The following change shall be made:

   Model number B464 B-XNF shall be listed as B464BXNF.
   (See Attachment 4 , Sheet 1)

2) The model numbers for pressure switches PS-29-4, PS-29-4-1, and PS-29-4-2 appear incorrectly in the Total Equipment Data Base. The following change shall be made:

   The listing of the model number for each pressure switch shall change from B464-B-XNF to B464BXNF (Reference Ashcroft Bulletin 110, dated 4/80). (See Attachment 4 , Sheet 1.)

3) The set points for pressure switches PS-29-4, PS-29-4-1, and PS-29-4-2 appear incorrectly in the St Lucie Unit 2 Instrument List. The following changes shall be made:

Tag Number	Incorrect Set Point Listing	Correct Set Point Listing
PS-29-4	HI ANN 6.5" WC	HI INT 5.8" WC
PS-29-4-1	HI INT 5.8" WC	LO ANN 1.0" WC
PS-29-4-2	LO ANN 1.0" WC	HI HI ANN 6.5" WC

   (See Attachment 4 , Sheet 1)

4) The mounting location for pressure switches PS-29-4-1 and PS-29-4-2 shall change from IR 10-1B to IR 10-1A. (See Attachment 4, Sheet 1)

5) The information "QTY-2" and "TAG-PS-29-4-2" shall be deleted from EMDRAC Drawing #2998-13526. (See Attachment 4, Sheet 2)

No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not 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?
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?

TURBINE GANTRY CRANE SEPARATION REQUIREMENTS

INTRODUCTION

This PCM Supplement provides restrictions on the clear distance to be normally maintained between the Units 1 and 2 turbine gantry cranes and loading combination restrictions under which the cranes may be operated regardless of separation.

DESCRIPTION

The St Lucie Units 1 and 2 turbine gantry cranes share a common runway. In order to prevent overstressing of the Turbine Building structures, it is necessary to either maintain a specified distance between the cranes or restrict the loads allowed on each crane.

This PCM Supplement reiterates the separation requirement originally provided by Supplement 0 of this PCM. Where this separation is maintained, the cranes may be loaded to design capacity simultaneously.

This PCM Supplement further provides load tables indicating maximum lifting capacities of each crane in conjunction with various loads on the other crane, when normal separation between the cranes cannot be maintained. The load tables envelop all possible crane loading conditions and locations for each building bay.

The allowable loads do not distinguish between loading on the main and auxiliary hooks, but represent the total load on both hooks.

SAFETY ANALYSIS

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

This PCM Supplement imposes load and proximity restrictions on the Turbine Building gantry cranes. The Turbine Building is a Non-Seismic Category I structure and contains no safety-related equipment. Therefore, this PCM Supplement does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

The load and proximity restrictions provided ensure that the cranes are operated without exceeding the design capacity of the turbine building structure. Therefore, this PCM Supplement does not create the possibility of an accident or malfunction of a different type than previously evaluated in the FSAR. This PCM Supplement does not involve any change to the St Lucie Unit 1 or 2 Technical Specifications.

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

HYPOCHLORITE CELL FLUSH SYSTEM

ABSTRACT

This Engineering Design Package (EDP) modifies the existing portable hypochlorite cell flush cart. The cart will be permanently mounted on a concrete pad and all piping and electrical connections will be made permanent.

This EDP is classified as non-safety related since it is a modification to a non-safety related system. The safety evaluation has shown that this EDP does not constitute any unreviewed safety question, nor does it require a Technical Specification change. Therefore, prior NRC approval is not required for implementation of this EDP.

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 Sodium Hypochlorite System is a chemical treatment system—for the water in the Circulating Water and Intake Cooling Water Systems and does not perform any safety related function. Accordingly, all components of the system are classified non-safety class, Quality Group D and non-Class 1E.

Chlorine, in the form of Sodium Hypochlorite is used to control biological fouling in the Circulating Water System by use of a Hypochlorite Generating System, serving both St Lucie Units 1 and 2.

This 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 Hypochlorite Cell Flush System equipment and associated piping and power supply 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 report is not created. The components involved in this modification have no safety related function. No changes have been made to the operational design of the hypochlorite 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. Failure of this system will be identified by instrumentation used to detect effulent chlorine content. This system can be restored to its operable status prior to unacceptable levels of slime accumulation.

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 necessary.

INTAKE CANAL DREDGING AND SLOPE RESTORATION

ABSTRACT

Several areas of the intake canal have been subjected to continuous erosion and sedimentation. Recent inspections of the areas indicate that the deterioration is due to various factors contributing to different extents; among these probable causes are canal currents, tidal action, and rainfall.

This PC/M provides restoration of canal embankments, the installation of new protection against erosion, and the removal of sedimentation east of the A1A bridge.

This PC/M does not involve an unreviewed safety question and has no effect on plant safety. The intake canal is not considered safety-related and the work to be done will be restoring the canal to its original design profile.

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 may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This PC/M restores Intake Canal embankments and does not involve an unreviewed safety question for the following reasons:

i The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since none of the work will be performed in the proximity of safety-related equipment; most of the work will be done outside the plant security fence. Failure of this system will not prevent the safe and orderly shutdown of the plant. The Emergency Cooling Water System, through Big Mud Creek, can provide adequate makeup and has been considered.

ii There is no possibility for an accident or malfunction of a different type than any evaluated previously since the Intake Canal is non-safety related, and this modification cannot affect any safety-related system. This modification will, in fact, increase the reliability and decrease the probability of an accident by restoring the canal to its original configuration.

iii This modification does not change the margin of safety as defined in the basis for any Technical Specification. This PC/M may be performed in any plant mode of operation.

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

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

BLOWDOWN BUILDING RADIATION MONITORING SYSTEM

ABSTRACT

This engineering package provides for the replacement of the effluent gaseous portion of the presently installed Nuclear Measurement Corporation (NMC) Blowdown Building Ventilation Airborne Radiation Monitoring System with a spare General Atomics (GA) Technologies Airborne Radiation Monitor.

The sensitivity levels of the GA Airborne Radiation monitor will be equal to or greater than the presently installed system and the outputs will be duplicated.

The Blowdown Building Airborne Radiation monitor will be used to record and annunciate the gross airborne trends in the Blowdown Building Ventilation System and the amount of radioactive releases to the atmosphere. Although this system provides no safety related function and the monitor will be physically located inside the Steam Generator Blowdown building, this PC/M is classified as Quality Related, since the monitors are used to assess the Blowdown Building's contribution of airborne radiation to the total airborne radiation effluent at the site.

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 PC/M, this PC/M 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 PC/M.

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 probability of occurence as the consequences of an accident or malfunction of equipment previously evaluated in the Safety Analysis Report is not increased by this PC/M. These modifications to the blowdown building radiation monitoring system will duplicate the outputs of the replaced system and reuse the existing isokinetic probe and output recorder.

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

a. The new equipment mounting will be seismically analyzed for additional loading in accordance with St Lucie Design Criteria Manual, Section I.

b. The new radiation monitor will be located in the blowdown building, which is considered to be a mild environment.

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 and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required."

SIMULATOR TRAINING FACILITY PIPING TIE-INS

ABSTRACT

This engineering package is being issued to cover the addition of the St. Lucie Simulator Training Facility fire protection, service water and sanitary piping tie-ins. No aspect of this project will add to, modify or otherwise affect any plant safety related system. Fire system modifications associated with this project that tie into the plant fire loop up to the first isolation valve shall be classified as "Quality Related" QA/QC required. The remainder of the fire protection, all service water and all sanitary piping shall be Non-Nuclear Safety Related.

The addition of the Simulator Training Facility fire protection, service water and sanitary tie-in lines do not pose any unreviewed safety questions nor involve any changes to plant Technical Specifications.

SAFETY EVALUATION

The St. Lucie plant fire protection loop is defined as a Quality Related system. Those portions of this modification that tie into the fire loop up to and including the first isolation valve have been designated as Quality Related and conform with the requirements of the original fire loop. The remainder of the fire protection piping added by this modification has been designated Non-Nuclear Safety Related Quality Group D. Those portions of the modification providing service water and sanitary piping and tie-ins are classified as Non-Nuclear Safety Related. These components tie into the existing plant service water and sanitary systems which are also classified as Non-Nuclear Safety Related Quality Group D.

A failure mode anlaysis was performed on the Non-Nuclear Safety Related portions of the modification. Based on this analysis, failure of the service water and sanitary piping or components and those portions of the fire main downstream of the first isolation valve will not inhibit the operation of any safety related equipment or components. These materials are located remote from any safety related equipment or components and as such cannot fall on a hit such components. Failure of the service water line will cause loss of service water to the simulator building. Failure of the sanitary line will inhibit the use of the Simulator Building sanitary system. Failure of the downstream fire main piping will not inhibit the functional capabilities of the fire loop since the post indicator valve, located upstream of these portions of the system provides adequate isolation capabilities to ensure functional integrity of the fire loop.

Those portions of the modification providing fire protection piping tie-ins to the first isolation valve can affect the functional capabilities of the fire loop and therefore can affect fire protection capabilities for Safety Related equipment and components. As addressed in the Design Analysis, these portions of the modification have been designed and construction requirements have been specified to comply with the necessary Quality Related requirements. Since the equipment affected by this modification is not considered by the FSAR in determining the probability of accidents or possible types of accidents or in the evaluation of the consequences of accidents, it can be concluded that the probability of occurrence of accidents previously addressed in the FSAR is unchanged and the possibility of new accidents not considered in the FSAR has not been created. Therefore, the potential failure mode of this system and degree of protection provided to nuclear safety related equipment remains unchanged.

Based on this information, it can be demonstrated that an unreviewed safety question as defined by 10CFR50.59 does not exist since the consequences of all analyzed accidents remains unchanged. Additionally, with respect to Nuclear Safety, no new accidents or malfunctions are introduced as a result of this modification. Finally, the margin of safety as defined in the Technical Specifications has not been reduced nor have changes to the Technical Specifications been required.

In conclusion, this modification is acceptable from the standpoint of nuclear safety since it does not involve an unreviewed safety question nor require changes to the Technical Specifications. Thus implementation of this modification does not require prior NRC approval.

STEAM GENERATOR BLOWDOWN TREATMENT FACILITY SYSTEM PUMPS MECHANICAL SEAL REPLACEMENT

ABSTRACT: The existing mechanical seals utilized in the subject pumps are being replaced by cartridge type mechanical seal. The change will reduce pumps' downtime required for seal replacement and is considered a maintenance enhancement. This change does not affect any technical specifications and there are no unreviewed safety questions.

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?