Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

November 16, 2000

L-2000-243
10 CFR 50.59

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

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 and summary of the safety evaluations for changes, tests, and experiments which were approved for Unit 2 during the period of December 9, 1998 through May 16, 2000. This submittal correlates with the information included in Amendment 13 of the Updated Final Safety Analysis Report submitted under separate cover.

Please contact us should there be any questions regarding this information.

Very truly yours,

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/spt

Enclosure

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

IE47

St. Lucie Unit 2
Docket No. 50-389
L-2000-243
Enclosure

ST. LUCIE UNIT 2
DOCKET NUMBER 50-389
CHANGES, TESTS AND EXPERIMENTS
MADE AS ALLOWED BY 10 CFR 50.59
FOR THE PERIOD OF
DECEMBER 9, 1998 THROUGH MAY 16, 2000

INTRODUCTION

This report is submitted in accordance with 10 CFR 50.59 (b), which requires that:

i) changes in the facility as described in the SAR;
ii) changes in procedures as described in the SAR; and
iii) tests and experiments not described in the SAR

which are conducted without prior Commission approval be reported to the Commission in accordance with 10 CFR 50.59(b)(2) and 50.71(e)(4). This report is intended to meet this requirement for the period of December 9, 1998 through May 16, 2000. Note that, where practical, summaries from more recent 10 CFR 50.59 evaluations have also been included in this report.

This report is divided into three (3) sections: the first, changes to the facility as described in the Updated Final Safety Analysis Report (UFSAR) performed by a Plant Change/Modification (PC/M); the second, changes to the facility or procedures as described in the UFSAR not performed by a PC/M and tests and experiments not described in the UFSAR; and the third, a summary of any fuel reload safety evaluations.

Each of the documents summarized in Sections 1, 2 and 3 includes a 10 CFR 50.59 safety evaluation that evaluated the specific change(s). Each of these safety evaluations concluded that the change does not represent an unreviewed safety question nor require a change to the plant technical specifications; therefore, prior NRC approval was not required for implementation.

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SECTION 1

PLANT CHANGE / MODIFICATIONS

PLANT CHANGE/MODIFICATION 96151

EDG 2A/2B WOODWARD ELECTRICAL GOVERNOR REPLACEMENT

Summary:

This modification affects the Emergency Diesel Generators (EDGs) for St. Lucie Unit 2. The EDGs provide backup Class IE 4160V ac power in the event of a loss of offsite power to those components and systems necessary for a safe shutdown of the plant and accident mitigation. The modification affects the control circuits of the EDGs.

The purpose of this modification was to replace the existing Woodward model 2301 electrical load controllers and associated components, which provide control of the EDG speed during operation, with Woodward model 2301A electronic load controllers and digital reference units. The existing Woodward model 2301 electrical load controllers are obsolete and spare parts are not readily available. Also, the existing system is susceptible to electronic "noise" and requires periodic electronics adjustment, which results in replacement of modules to bring the system performance to within acceptable criteria. These alignments have resulted in lost production for Unit 1, which has a similar EDG load controller as Unit 2. In addition, the existing design requires use of the hydraulic governor and speed changer (Bodine motor) on the engine governor actuators for emergency full-speed start; this speed changer motor is responsible for a majority of engine governor failures.

The existing Woodward model 2301 electrical load controllers only operate in a narrow band around 900 rpm; EDG control during acceleration and at idle speeds is done via the hydraulic governor. The Woodward model 2301A electronic load controllers and digital reference units are capable of controlling the EDGs during acceleration and at idle and full speeds. In order to take advantage of increased capabilities of the model 2301A load controllers over the model 2301 load controllers, the EDG control system is to be modified to permit the model 2301A load controller to control EDG operation at all speeds. This also requires modification of the hydraulic governor control logic circuits. Installation of the Woodward model 2301A electronic load controllers will also require installation of a second magnetic speed pickup at the engine to provide an EDG speed input signal to the load controller, which is currently derived from the generator output for the model 2301 load controller.

Per the original design, EDG speed is manually adjusted from a governor up/down switch in the Control Room, which controls a motor operated potentiometer to vary an input signal to the

electrical load controller. Upon occurrence of a SIAS or bus undervoltage condition, the motor operated potentiometer is automatically switched out of the circuit and a fixed resistance is connected in its place. The fixed resistance is set to provide an input to the electrical load controller resulting in EDG operation at approximately 60 Hz (900 rpm). During this time, manual control from the Control Room is blocked. The existing motor operated potentiometers and fixed resistors are to be replaced with new Digital Reference Units, which will provide improved and more precise control functions and, due to the lack of moving parts, will also require less maintenance. Manual control of EDG speed using the Digital Reference Unit when operating in parallel with the offsite source will not be different from manual control with the motor operated potentiometer. Operation of the red/green/amber EDG governor setting indicating lights will be somewhat different than the present operation due to the characteristics of the Digital Reference Unit and will require additional operator training.

Revision 1 of this modification provides clarification to functional verification, plant restrictions, and post-modification testing.

PLANT CHANGE/MODIFICATION 98021

RCP SEAL CHANGE TO N-9000 SEALS

Summary:

This modification includes the engineering and design necessary to provide justification for replacement of the existing Unit 2 Reactor Coolant Pump SU mechanical seals with N-9000 seals.

The existing Unit 2 RCP multistage mechanical seal cartridges have two component stationary and rotating face assemblies and rotating springs; these seals have some undesirable responses to pressure and temperature transients which limit seal life. Under Station Blackout (SBO) conditions seal failures can occur as a result of overheating of the seal's elastomeric components and a high rate of two-phase flow across the seal faces. The potential for a seal to pop open and result in unmitigated seal leakage has been identified by the NRC as a serious concern - especially during station blackout conditions when makeup provisions are affected.

10CFR 50.63, entitled "Loss of All Alternating Current Power" (Station Blackout), requires that each light-water-cooled nuclear plant be able to withstand and recover from a station blackout of a specified duration. Failures of the Reactor Coolant Pump seals that would result in a loss of coolant accident are of concern. Particular attention is focused on RCP seal behavior during an SBO when seal injection and component cooling water flow are lost and the seal is exposed to full reactor coolant temperatures.

Replacing the SU seals with N-9000 seals will increase service life and provide a reliable seal during normal and abnormal plant operation. N-9000 seals have a proven reliability record in other Nuclear Power plants (e.g., Millstone Unit 2, Waterford 3, and Maine Yankee) and have a projected life of 50,000 hours. The replacement N-9000 seals have been designed and rigorously tested to demonstrate their ability to meet Station Blackout coping time with low leakage rates.

10CFR 50, Appendix R Safe Shutdown Analysis assumes loss of system function during fires in specified zones. The N-9000 seals require that the controlled bleed-off (CBO) flow be interrupted during loss of seal cooling for the elastomers to cope with the full reactor coolant temperature. There are two separate CBO paths; normal path is to the VCT via V2532 and the backup path is to the Quench Tank via V2507. Pneumatic valve V2507 will be changed from normally locked open, fail open to normally closed, fail closed. RTGB switch HS-2507 will be changed to provide open contacts in each power/control leg to

prevent misoperation during a fire. An emergency light will be installed in the mechanical penetration room to allow manual closing of valve V2532. This will provide isolation of the controlled bleed-off flow should the automatic containment isolation valves fail to close due to a Control Room fire.

The Reactor Coolant System is Quality Group A, Seismic, designed in accordance with ASME Section III, Class 1, 1973 Edition, Summer 1973 Addenda to 1971 Code. The replacement Reactor Coolant Pump N-9000 seals will be designed in accordance with ASME Section III, 1983 Edition with Summer 1984 Addenda, Class 1 requirements. Design of the N-9000 seals to a later ASME Section III Code year and Edition is acceptable under the rules of ASME Section XI, Article IWA 7000.

Revision 1 of this modification adds more details for post-modification testing, replaces Attachments 5, 9, 12 and 13, makes minor text corrections and identifies plant affected operating and emergency operating procedures.

PLANT CHANGE/MODIFICATION 98033

REPLACEMENT OF ISOLATION VALVES FOR CONTAINMENT PRESSURE TRANSMITTERS

Summary:

This modification provides details for replacement of the existing normally open, fail closed valves (SE-07-5A, B, C, D) with normally open, fail open valves such that power loss will not prevent the containment pressure transmitters from sensing their inputs.

In the current configuration, power loss due to a single breaker or fuse will cause two valves to fail closed and prevent the pressure transmitters from sensing their inputs. Loss of two channels of containment pressure would prevent actuation of the protective system when in a two-out-of-three configuration. High Containment Pressure is used as input for Reactor Trip, SIAS, CIAS, CSAS, and MSIS per Technical Specifications 3/4.3.1, 3/4.3.2, and 3.3.3.6.

Condition Report 98-0512 identified that the electrical configuration of isolation valves (SE-07-5A. B, C, D) for operation of the containment pressure transmitters (PT-07-2A, B, C, D) does not meet single failure criteria in accordance with IEEE 279-1971 and General Design Criteria when in a two-out-of-three configuration. Because the containment pressure inputs have been placed in bypass at different times in the past, it was determined that this condition (in conjunction with a single failure) could have prevented the fulfillment of a safety function and was consequently reported under 10 CFR 5.73(a)(2)(v) in LER 50-389/98-003-0.

PLANT CHANGE/MODIFICATION 98037

DELETE BYPASS FUNCTION FOR LOW CCW FLOW TO THE RCP RPS TRIP

Summary:

The purpose of this modification is to delete the control board switch (CS-206-1, 2, 3 and 4) function, which is used to bypass the (low CCW flow to the RCP) RPS trip.

During a review of UFSAR section 7.2.1.1.5 it was discovered that bypass features are provided for each of the RPS inputs for the trip on low CCW flow to the RCPs. This bypass feature is provided by placing four key operated switches (located within RTGB 206) to the bypass position. Contrary to the UFSAR, multiple keys are available to allow bypassing the trip function without adequate indication of the bypass condition (Reference: CR 98-0241). This EP resolves the concern of multiple keys and inadequate indication.

This Engineering Package provides the engineering necessary to justify the modifications and document the changes. The PC/M allows for the bypassing of multiple (CCW flow to the RCP) channels when the RPS trips are not required by Technical Specifications. Channel bypasses are to be accomplished using multiple keys at the RPS panel where positive indication is available to the control room operators.

The annunciation (L-14) and sequence of event logic is being changed as follows. The control switch (CS-206-1, 2, 3 and 4) is no longer required to be in the "Normal" position as a precondition for a low flow alarm (timer counting down). This change is being made to eliminate any concern of inadvertently bypassing a low flow alarm condition (during normal conditions) and to allow for testing the timing relay (80XA, B, C and D).

PLANT CHANGE/MODIFICATION 98038

PRESSURIZER HEATER REPLACEMENT

Summary:

Eight pressurizer heaters were identified as requiring replacement during the Unit 2 Cycle 11 outage. While the replacement of the heaters is considered a standard maintenance activity not requiring a PCM, several jumpers have been installed as documented in two TSA's and two previous PCMs, which will require removal. This PCM revises the plant documentation to reflect the original pressurizer heater configuration, provides for abandonment of a stuck pressurizer heater in the event it is necessary during this maintenance effort and incorporates the conclusions of Safety Evaluation JPN-PSL-SEMP-95-004 Revision 4 into the pressurizer vendor manual.

PLANT CHANGE/MODIFICATION 98041

REPLACE TRAVELING SCREENS & TRASH RACKS

Summary:

The Engineering Package includes the engineering and design necessary to provide justification for replacement of the Unit 2 Intake Screening System (Traveling Water and Stationary Screens) with upgraded corrosion resistant material.

The replacement Unit 2 Intake Screening System has been designed, fabricated and tested to be an improved equivalent with stainless steel and fiberglass materials to increase service life and provide a reliable system during normal and abnormal plant operation.

The traveling screen motors are being changed from 7.5/3.8 Hp to 10/5 Hp for larger capacity during times of increased loading (jelly fish intrusion). This increase in motor size requires the replacement of the circuit breakers and motor thermal overload heaters and changeout of the motor power feeder cables to larger size cables. Much of the existing exposed conduits are to be replaced with noncorrosive UV-resistant thick-walled PVC conduit.

Cast Iron Valves V21381, V21382, V21383 & V21384 will be replaced with stainless steel valves to afford better seawater corrosion resistance.

Additionally, the exposed sections of the Unit 2 trash pipe will be removed. Since it serves as a drain for the hypochlorite system, the hypochlorite drain piping will be rerouted to by-pass the removed sections of trash piping.

PLANT CHANGE/MODIFICATION 98075

CONTROL ROOM EMERGENCY AIR CLEAN-UP SYSTEM (CREACS) HVAC DUCT INSPECTION/REPLACEMENT/REWORK/REPAIR

Summary:

This PCM provides technical direction to assist in the inspection and replacement/rework/repair of potential corrosion degradation identified in the Unit 2 Control Room Emergency Air Cleanup System (CREACS). The scope of this PCM includes the CREACS Outside Air Intake (OAI) inlet lines up to the common header.

Several instances of degraded/corroded ducts have been identified in the CREACS. CR 96-1840 identified a through wall leak on the bottom of the duct between valves FCV-25-15 and FCV-25-17. CR 98-0866 identifies that the CREACS duct is not externally insulated as required by drawing 2998-G-873, and the presence of internal insulation has not been confirmed. CR 98-1620 identified degraded duct in the Control Room Kitchen, with through wall holes on several duct sections. The degradation of the duct appears to be due to water condensing out of the humid outside air upon being exposed to the cooler duct within the Control Room Envelope. From the initial inspection of the duct in the Control Room Kitchen, it appears that this section of duct has internal insulation. Once water is condensed, it could be absorbed by this insulation, and the moisture-laden insulation would remain against the ductwork, promoting corrosion. The internal insulation on degraded sections should be removed, and if required for acoustic abatement or thermal insulation, replaced with external insulation. In order to "breach" the duct for the inspection and replacement/rework/repair, while maintaining the Control Room Envelope, compensatory actions are required. This PCM provides the compensatory actions, such as installation of a Blanking Plate to extend the Control Room Envelope to the inlet of the CREACS OAI inlet and isolation of one of the inlet valves.

During preparation of this PCM, several drawing errors were identified and additional information was located to support duct replacement/rework/repair. The valve numbering as depicted on drawing 2998-G-879, sheet 2 is not in conformance with the other related drawings with regard to location of valves FCV-25-14 and FCV-25-16. Additionally, drawing 2998-G-873 does not identify valves FCV-25-14 and FCV-25-16 and will be revised to add proper tagging. Drawing 2998-G-873 is also revised to show the correct drawing reference for the missile shield hoods. The vendor document, "Duct Brochure and Construction Details HVAC Work" is added to the Document Control System, and provides details for access doors. The as-built configuration for the wall

penetration/hanger on the North OAI inlet will be added as a detail.

Revision 1 of this modification clarifies the basis for the CREACS inlet duct wall thickness acceptance criteria, and clarifies the closure methodology for the internal insulation inspections.

The Engineering Package (EP) includes the modification of the Control Room Envelope by installation of a Blanking Plate to extend the Control Room Envelope to the CREACS OAI inlet, on a temporary basis, in accordance with Technical Specifications requirements.

PLANT CHANGE/MODIFICATION 98117

CONTROL ROOM ACOUSTIC PANELS IN PLENUM

Summary:

The Operations Department has been experiencing uncomfortable noise levels in the Unit 2 Control Room. In order to reduce the ambient noise levels, some measures (e.g., installing additional carpeting, relocating the RPS panel fans, etc.) have been implemented. Although these modifications have alleviated some of the noise, they have not reduced the noise to a more comfortable level.

A study was conducted by a Human Factors Engineering Consultant to more accurately determine the source(s) of the noise and recommend potential solutions. Based on the results of the study, it was determined that the average ambient noise level in the Unit 2 Control Room can be further reduced. Therefore, it has been determined that the installation of sound absorption material in the control room plenum would further reduce the ambient noise level.

The proposed modification will install acoustical panels and blankets above the suspended ceiling of the Control Room. The acoustical material consists of rigid fiberglass panels and fiberglass quilted blankets, which will be seismically fastened to the exposed ceiling and walls areas of the control room plenum (i.e., above the reflective ceiling). The fiberglass blankets will be mainly used as noise absorption baffles suspended above the luminous louver panels ("egg crate") ceiling.

Revision 1: Consistent with the recommendations of Human Factors Engineering Consultant, to reduce the noise levels originating from the back of RTGB cabinets, appropriate fiberglass acoustic panels will be installed vertically above the front face of RTGB cabinets between the top of cabinets and the reflective ceiling. The fiberglass panels will be seismically supported and will have sufficient clearance with the top of cabinets to preclude any potential seismic interaction.

PLANT CHANGE/MODIFICATION 99068

FIRE PROTECTION SYSTEM HOSE STATION ADDITIONS/RELOCATIONS/MODIFICATIONS

Summary:

Revision 0 of this Engineering Package (EP) installed three hot tap branch connections in the manual Fire Protection System suppression water piping that will be used to supply the new hose stations at five locations in the plant. Hose stations are located throughout the plant so that all areas, with some exceptions, are within reach of a fog nozzle when attached to a hose that is not more than 100 feet long. Hot taps or pressure taps is a method of making a connection to existing piping while the existing system is under pressure.

Revision 1 adds five new hose stations, along with all associated piping, that are required to provided manual Fire Protection suppression water spray coverage in accordance with the UFSARs. The five new areas that require coverage were determined in the Engineering Disposition to CR 98-0180-1. Unit 1 requires an additional hose station to cover the area east of the Control Room, which includes the CCW Surge Tank Room and the Electrical Equipment Fan Room. Both units require two similar additional hose stations in the Radiation Control Area (RCA); For Unit 2, one hose station will be located at the northeast corner of the H and V Equipment Room and the other hose station on the outside wall of the Containment Purge Air Room. These hose stations will have 100 feet of hose on each reel.

PLANT CHANGE/MODIFICATION 99116

SHIELD BUILIDING SECONDARY BELLOWS REPLACEMENT FOR COMPONENT COOLING WATER PENETRATIONS P-15 THROUGH P-24

Summary:

This modification provides design details to replace shield building secondary bellows and apply antisweat insulation to the CCW process piping between the RAB and RCB for penetrations P-15 through P-24.

The formation of condensation inside the bellows assemblies has resulted in standing water and corrosion of the carbon steel components of the penetrations and the associated Component Cooling Water piping. Access to the process piping and internal area of the penetrations will require removal of the secondary bellows from the CCW penetration assemblies.

Shield Building secondary bellows for PSL-2 were designed in accordance with ASME Section III. Class 3. The replacement bellows will be designed in accordance with the Expansion Joint Manufacturer's Association Standard and original PSL Unit 2 design criteria and manufactured under Pathways ASME Section VIII program.

PLANT CHANGE/MODIFICATION 99117

PARTIAL REPLACEMENT OF ECCS SUCTION LINES 24" CS-2 & CS-3

Summary:

PC/M 99117 provides design details and justification for the replacement of two sections (approximately 110 feet each) of 24" Emergency Core Cooling System (ECCS) piping (lines 1-24"-CS-2 and 1-24"-CS-3) located within the concrete tunnel connecting the Refueling Water Tank (RWT) and the Reactor Auxiliary Building (RAB). This replacement is a "like-for-like" replacement in that the configuration and layout of the piping will remain in its present state. However, the current Type 304 stainless steel piping will be replaced with dual certified Type 316/316L stainless steel piping to increase the corrosion resistance properties of the lines. Due to the similarities in physical characteristics of the two materials, no additional supports will be required. The existing supports will be dismantled to facilitate installation, however, these supports will be reconstructed to the as-designed strength or better. The operation of all ECCS systems due to this modification will be unchanged.

Two new high point vents will be added to these lines (one per line) for operational considerations. These vents will be seismically restrained with two new supports. The two existing drain valves will be replaced in their original configuration. All new valves installed (both the two vents and the two drains) will be Anchor/Darling valves. This represents a change in valve vendor for the existing drains, which are currently Rockwell valves. However, no change in function or capability will occur due to this modification.

Two 4-inch mechanical closure branch connections (one in each train) will be incorporated in the vertical pipe section to afford installation of FME devices and/or water stops should MV-07-1A or 1B leak past the seat.

Additionally, the piping replacement detailed in this EP requires the modification of portions of Instrument Air (IA) header 2"-IA-8 due to physical interference with field welding machinery. This line will be permanently rerouted and have an isolation and drain valve installed between the RAB and the Component Cooling Water Building (CCWB), the Fuel Handling Building (FHB), the Diesel Generator Building (DGB), and the Primary Water Storage Tank. This isolation valve will allow work on the air header supplying the referenced areas without requiring isolation and loss of air to a large section of the RAB. The drain valve will allow depressurization and blowdown of the header from the RAB for the branch connections.

The ECCS supplies borated water to the reactor core during design basis events to ensure the capability to shut down the reactor and mitigate the consequences of accidents that could result in offsite exposure as defined in 10CFR100. ECCS piping lines 1-24"-CS-2 and 1-24"-CS-3 connect the RWT to the suction of ECCS safety components such as the High Pressure Safety Injection (HPSI) pumps, Low Pressure Safety Injection (LPSI) pumps, and the Containment Spray (CS) pumps. These two 24" lines are located immediately downstream of valves I-MV-07-1A and I-MV-07-1B. The lines enter a concrete piping tunnel prior to entering the RAB, which limits any natural air circulation around the pipes. Due to the proximity of Unit 2 to the ocean, the environment within the tunnel is salt laden resulting in high rates of chloride induced stress corrosion cracking of the outside diameter of this ECCS piping. Therefore, these two piping runs are to be replaced with a more corrosive resistant material.

Revision 1 incorporates changes to the Unreviewed Safety Question Determination, includes mechanical closure branch connection description and calculation reference, adds branch connection mechanical closure drawing, revises rigging attachment drawings, revises spool drawings, and provides other minor word document enhancements.

PLANT CHANGE/MODIFICATION 99125

FUEL HANDLING SYSTEMS UPGRADES

Summary:

The purpose of this modification is to upgrade the Fuel Handling System to increase the speed of refueling operations and to automate the refueling operations to minimize the likelihood of errors during refueling. The upgrade will also address the issue of equipment obsolescence and component degradation due to aging.

Refueling Machine upgrades consist of the following:

- Removable Console-Standard 6' Dual Bay
- Bridge, Trolley and Hoist motors and drives replaced
- Replacement Encoders/Positioning & Boundary System
- Motion Alarm
- Main Hoist Load Weighing System
- Software
- Automatic, Semi-Automatic and Manual Controls
- RFM to Transfer Control (Start/Stop) Cabling

Fuel Transfer System upgrades consist of the following:

- Reactor Building Console
- Fuel Handling Building Console
- Winch Motor and Gearbox replaced
- Laptop Computer for troubleshooting
- Winch Encoder
- Software
- Hydraulic Power Units (HPU) in RCB and FHB

Spent Fuel Handling Machine upgrades consist of the following:

- Bridge, Trolley, Hoist Motors and Motor Drives are replaced
- Hoist Gearbox
- Hoist Speed-Increase
- Operator Console/Trolley Junction Box
- Control Panel - Existing Enclosure
- Load Weighing System
- PLC for Interlocks - No Motion Control
- Software

This modification provides the design outputs necessary to justify the modifications and document the changes. The PC/M revises the UFSAR to reflect the design changes. Documentation will include revised drawings, new drawings, new vendor manuals, installation instructions, and Post-Modification Test instructions. This EP installs 1E, Safety Related circuit breakers and penetration modules.

Revision 1 of this PC/M was issued to document the following changes:

1. The Safety Classification is changed to Safety Related to reflect the installation of Class 1E, Safety Related circuit breakers and penetration modules
2. Provide the latest setpoint information. This change is to reflect changes in the numbering of the selector switch positions and their associated setpoint values.
3. Change cable number from 20185G to 20185C in the design description section to match the CCL and Implementation instructions.
4. Change revision number for new VTM from "4" to "NEW" to reflect a new issue.
5. Correct typos in Implementation instruction 19 and delete duplicated instruction 20.
6. Add "Seismic Qualification" to Design Change Checklist, Section 8.19, and hold points to address 1E breakers.
7. Hold point added for Rockbestos cable Doc Pac revisions.
8. Attachments 21, 25, 26, 27 Revised to include vendor signatures and associated calculation spreadsheets.

Revision 2 of this PC/M was issued to document the following changes:

1. Implements hoist speed restrictions imposed by PSL-2FJF-00-002 and remove the associated hold point.
2. Provides installation and test instructions for the new fiber-optic feedthrough module for Penetration D9. The EQ doc pac, 2998-A-451-15.1 will not be revised. However, penetration D9-10 will be removed from the EQ List.
3. Address Failure Mode and Effects Analysis (FMEA) issues and remove HOLD points.
4. Incorporate comments resulting from the Inter-Discipline Review (IDR) by the Chief Engineers.
5. Incorporate the addition of the "Test Weight" position on the SFHM Selector Switch.
6. Add HOLD point for RFM hoist motor drive "Current Limit" setting.

7. Added General Installation Requirements addressing bolt lengths and hole "slotting" to Implementation Instruction 43 based on "lessons learned" input received.
8. Revised Implementation instructions 28 and 31 to reflect changes made in accordance with CRN 99125-8953.
9. Added instrument-scaling information for the Transfer Machine load weighing system.

PLANT CHANGE/MODIFICATION 99158

AFW - CHEMICAL INJECTION SYSTEM SAFETY CLASS BOUNDARY CHANGE

Summary:

This modification provides design details for adding class boundary check valves between the Safety Related Auxiliary Feedwater System and the Not Nuclear Safety Chemical Injection System. This modification has been developed to bring about corrective actions identified in CR 98-1107 and CR 98-1593. CR 98-1593 identified that a non-manual class boundary did not exist between the two referenced systems when the manual isolation valve was opened to inject chemicals into AFW in Mode 3. Chemistry has stated that they need to inject chemicals into the Auxiliary Feedwater System in Mode 3 to support startup and shutdown of the plant. These CRs identified that the AFW System was in an operable, but degraded condition, with compensatory actions required.

This Engineering Package also includes modifications to the piping interface between the Chemical Injection System and the Main & Auxiliary Feedwater Systems. These piping modifications are designed to resolve a valve inaccessibility problem, and are as follows:

1. Two new 3/8" manual isolation valves, V09728 and V09729, will be installed in the Steam Trestle, one near each associated MFIV. These valves will be Not Nuclear Safety (NNS), and provide accessible isolation between the Chemical Injection System and the Main Feedwater supply headers. The existing isolation valves, V09290 and V09310, are located in the overhead of the Turbine Generator Building mezzanine level, and are inaccessible without using a ladder or scaffold.

2. Two new tubing runs will be installed, using existing tube track, between the Steam Trestle and the Turbine Building mezzanine level to support the remote location of the new isolation valves.

3. Existing 1" piping and valves between Chemical Injection supply lines 3/8-CF-1 and 3/8-CF-2 and isolation valves V09290 and V09310 will be reworked with tubing and 3/8" valves. The components to be replaced include vent valves V09288, V09289, V09308 and V09309 and check valves V09287 and V09350. These changes are being made in lieu of adding new pipe supports, and will simplify implementation of this modification.

SECTION 2

SAFETY EVALUATIONS

SAFETY EVALUATION PSL-ENG-SENP-94-029
REVISION 3

SHUTDOWN OPERATIONS CIRTERIA FOR REDUCED INVENTORY AND DRAINING THE REACTOR COOLANT SYSTEM

Summary:

The purpose of this evaluation is to identify the plant changes required to safely operate St. Lucie Unit 2 during modes 5 and 6 with the reactor coolant system (RCS) partially drained of water and with a full core of irradiated fuel present in the reactor vessel.

Following reactor shutdown (k_{eff} <1.0) irradiated fuel continues to produce substantial quantities of heat due to the decay of fission products, primarily through emission of gamma (γ) rays. Most of this decay heat is deposited in the reactor coolant and is subsequently removed from the RCS by the shutdown cooling heat exchangers. The rate of production of decay heat decreases as the time after shutdown increases. The water inventory present in the RCS to absorb decay heat will fluctuate during mode 5 and 6 activities but it will always be less than the normal operating level. During this evolution, the RCS will be drained to approximately the mid-plane of the hot leg piping; this condition is referred to as "mid-loop."

In the event of a loss of shutdown cooling while at a reduced inventory or mid-loop condition, the deposited γ-ray energy would heat the RCS inventory to saturation and begin to boil-off the remaining coolant. If this boiling condition persists, reactor fuel will be uncovered by the loss of inventory. Substantial quantities of steam will be evolved during any inventory boil-off.

Calculations have shown that the boil-off of RCS inventory at low pressures is a relatively slow evolution that requires two hours or more to reduce the RCS water level below the top of the active fuel. However, if the steam generated by the boil-off process is not effectively vented from the system, the pressure within the reactor vessel upper plenum may increase, depressing the reactor vessel water level such that the active fuel is exposed. This scenario, which also requires the presence of an opening in the RCS cold leg, leads to a more rapid core uncovery than does the low pressure boil-off scenario.

To preclude rapid core uncovery following a loss of shutdown cooling, the criteria for draining the RCS after shutdown will now be constrained by both the time to core uncovery and by vent path area requirements. As a result, an important requirement

from this safety evaluation is that a vent pathway connecting the fluid in the RCS hot leg and the containment atmosphere must exist when the RCS coolant inventory level is below the top of the hot leg piping. Evaluation of this condition effectively bounds other scenarios initiated at higher RCS levels.

Revision 1 of this evaluation updated the discussion to reflect use of more bounding decay heat generation curves for St. Lucie and included information on RCS hot side vent path requirements.

Revision 2 of this evaluation discussed that the removal of a primary side steam generator manway can provide a vent path that may be credited toward the vent area requirements under certain conditions. This revision discussed the constraints and limitations associated with use of this vent pathway.

Revision 3 of this evaluation removes portions of the text and appendix that deal with credit for heat removal from the steam generators during reduced inventory conditions. Revision 3 revises and updates the presentation of available vent path areas, boil-off rates and required wait times for entering reduced inventory conditions after reactor shutdown.

SAFETY EVALUATION PSL-ENG-SENP-94-043
REVISION 1

TEMPORARTY REMOVAL OF THE ICW PUMP MISSILE SHIELD

Summary:

This safety evaluation demonstrates the acceptability of plant operation while a section of the intake cooling water (ICW) pump missile shield roof is removed temporarily to perform maintenance activities on an out of service pump. The function of the missile shield is to protect the ICW pumps during a hurricane/tornado from missiles.

This safety evaluation documents the design intent of the ICW pump missile shield with respect to maintenance access. It concludes that removal of missile shield roof sections for maintenance during plant operation is consistent with the original design intent of shields. Furthermore, the risk of tornado missiles is negligible for the short period of time the roof section is not in place. As an additional precaution, the missile shield roof sections are re-installed in the event of a threatening hurricane when the risk of damage from tornadoes is the greatest.

Revision 1 was issued to clarify the acceptability of removing the intake structure ventilation system missile shielding as necessary in order to support ICW pump maintenance.

SAFETY EVALUATION PSL-ENG-SENS-97-006
REVISION 2

PERFORMANCE OF FULL CORE REFUELING OFFLOADS

Summary:

Revision 2 of this Safety evaluation documents the constraints on plant operation during the performance of full core fuel offloads that are necessary to assure conformance with the requirements of an NRC Safety Evaluation Report (SER) issued to support Amendment 101 to the St. Lucie Unit 2 license. This SER, dated May 6, 1999, was issued to license an increase in the spent fuel pool storage capacity at Unit 2. Section 2.2.2 of this SER requires that FPL:

> ... perform outage-specific calculations to demonstrate that the SFP [spent fuel pool] bulk temperature will not exceed the St. Lucie design basis temperature of 150ºF given the operation of one SFPCS [spent fuel pool cooling system] pump and one heat exchanger.

This Safety Evaluation utilizes calculation results and information from PC/M 99-125 to provide specific constraints applicable to the end of Cycle 11 (EOC11) refueling operations. These constraints assure that the maximum SFP bulk water temperature will remain less than 150ºF at all times during the refueling evolution.

This safety evaluation was originally prepared to justify the routine performance of full core fuel offloads. With the condition imposed by the SER issuing License Amendment 101, each full core offload requires a cycle-specific calculation documenting that the maximum bulk water temperature remains below 150ºF. Accordingly, a single set of guidance applicable to all full core fuel offloads can no longer be provided which would be most efficient for cycle-specific refueling operations. Thus, Revision 2 of this safety evaluation represents a major rewrite of the text presented in Revision 1 and has been prepared to document requirements applicable to EOC11. Future full core fuel offloads at St. Lucie Unit 2 may require an additional revision to this evaluation.

Revision 1 of this safety evaluation was prepared to correct an arithmetic value and to specifically state that the electrical bus swapping evolution may be performed during refueling activities without invalidating the conclusions of this safety analysis.

SAFETY EVALUATION PSL-ENG-SEMS-97-013
REVISION 2

IN-SITU HYDROSTATIC TESTING OF STEAM GENERATOR TUBE FLAWS

Summary:

Steam generator tube integrity performance criteria are stated in NEI 97-06. These criteria are consistent with and bound other criteria that have been applied previously in earlier versions of NEI 97-06, Draft Regulatory Guide 1.121 and factors of safety implicit in the ASME Code.

The structural performance criterion is:

> Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a margin of 3.0 against burst under normal steady state full power operation and a margin of 1.4 against burst under the limiting design basis accident concurrent with a safe shutdown earthquake.

The accident-induced leakage performance criterion is:

> The primary to secondary accident induced leakage rate for the limiting design basis accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. For St. Lucie Unit 2, leakage is not to exceed 1 gpm through all steam generators, and 0.5 gpm through any one steam generator.

In situ testing is performed on tubes containing indications of degradation based on tube inspection by eddy current testing. Eddy current test results may conservatively overestimate the severity of flaws due to eddy current tests' inability to detect micro-ligaments in cracks that provide reinforcement.

In situ pressure and in situ leak testing are methods to demonstrate that tubes meet the performance criteria stated above. The results of in situ pressure testing allow the determination of the structural and leakage integrity of steam generator tubes for the purpose of Condition Monitoring. This testing may also be used in conjunction with other inspection results to predict tube integrity for Operational Assessment. The FPL Steam Generator Integrity Program requires that in situ testing be evaluated when indications are detected that may challenge performance criteria and in cases of known

primary-to-secondary steam generator tube leakage.

Revision 1 of this evaluation removed the run time correction factor added to the test pressures to justify a longer run time since Cycle 11 operation time is scheduled and designed to be less than Cycle 10. Additional details were also provided for collection of leak rate testing. A change in the plant restrictions was also made to account for in situ testing being performed with fuel in the reactor vessel (or during fuel movement) as opposed to being defueled.

Revision 2 is a general revision of this safety evaluation. The Revision 2 evaluation also updated industry references, incorporated the FPL Steam Generator Program requirements, NEI 97-06 and revised the target test pressure tables to account for the planned longer Cycle 11 operating period.

SAFETY EVALUATION PSL-ENG-SECS-97-014
REVISION 2

SAFETY EVAL FOR SPEC-C-035: INSTALLATION OF TYGON TUBING FOR VENTING & DRAINING IN-SERVICE EQUIPMENT

Summary:

Nuclear Engineering Specification SPEC-C-035, entitled "Installation of Tygon Tubing for Venting and Draining In-Service Equipment," has been developed to provide generic installation instructions and securing details for vent and drain rigs for Safety Related, Quality Related, and Non Nuclear Safety equipment. Plant personnel are permitted to install vent and drain rigs for in-service equipment, as long as the requirements of the specification are satisfied. If the Implementers cannot install a vent or drain configuration in accordance with the provisions of the specification, or should the Implementers require a change to any provisions of the specification, a "Request for Specification Clarification or Change" sheet must be submitted to Engineering for review and approval. This request is reviewed by Engineering to ensure that the proposed configuration is acceptable with regard to applicable design criteria, the requirements of the specification, and this safety evaluation.

This safety evaluation provides the basis for the acceptability of using the specification for the installation of the venting and draining rigs, in lieu of the current practice which requires that the applicable system be declared Out of Service (in the absence of an engineering evaluation, specification, or Operations procedure) before the vent or drain rig can be installed. It also demonstrates that vent and drain rigs installed in accordance with the specification meet all technical and licensing requirements for St. Lucie Units 1 and 2.

This safety evaluation concludes that the use of the specification meets all technical and licensing requirements and has no adverse impact on plant operations. It is also concluded that the use of the specification does not compromise the safety and licensing bases for St. Lucie Units 1 and 2.

The specification provides installation instructions for vent and drain rigs for equipment which may be Safety Related, Quality Related, or Not Nuclear Safety. However, the quick disconnect couplings, tygon tubing, and associated components and hardware will not be used as pressure retaining components.

SAFETY EVALUATION PSL-ENG-SENS-97-036
REVISION 1

INTAKE STRUCTURE VENTILATION SYSTEM OPERATION

Summary:

The intake structure ventilation system consists of exhaust fans HVE-41A and B. The fans are designed to maintain room temperature within design limits and to operate as a support system for the intake cooling water (ICW) pumps which are required by the Technical Specifications in Modes 1, 2, 3 and 4. There is no Technical Specification Limiting Condition of Operation (LCO) for the fans.

The Unit 2 UFSAR describes the fans as Safety Related components which are designed to maintain the temperature of the ICW pump room between 80 - 120°F. The correctly stated design function of the fans is to limit room temperature to less than 120°F. The UFSAR description also implies that fan operation is required in all modes of plant operation. In fact, there are certain plant conditions where the fans may not be required to operate (e.g., during periods of cool weather and/or with only one ICW pump in service). NRC Generic Letter 91-18 describes support system operability and recognizes that a licensee may modify the support function by the use of the 50.59 process and UFSAR change. This evaluation revises the UFSAR to address the above items and also clarifies a statement regarding manual and automatic system operation.

Revision 1 of this evaluation supports an additional UFSAR change in that the ICW exhaust fans are not provided with and are not required to have two thermostats to automatically start (stop) the fans at room temperature increasing to (falling to) 80°F as stated in the UFSAR.

SAFETY EVALUATION PSL-ENG-SENS-97-052
REVISION 0

DISCONTINUED USE OF RADIATION MONITORS IN THE STEAM GENERATOR BLOWDOWN TREATMENT FACILITY

Summary:

This safety evaluation provides the basis and justification for an UFSAR update required due to the non-use of the automatic monitoring and control features of three radiation monitors (RE-45-1, RE-45-2, RE-45-3) located in the Steam Generator Blowdown Treatment Facility (SGBTF) and downstream of the Monitor Storage Tanks (MST). These monitors and control functions are specifically discussed in the UFSAR.

In addition, a few editorial inconsistencies were noted in the UFSAR chapter descriptions that will be clarified via this revision.

This evaluation evaluates equipment in the SGBTF that is no longer needed since other available equipment perform similar functions.

SAFETY EVALUATION PSL-ENG-SEMS-97-070
REVISION 1

USFAR COMBUSTIBLE LOADING UPDATE

Summary:

This safety evaluation addresses the combustible loading increase in various fire zones due to Thermo-Lag fire barrier material and other miscellaneous materials. This Safety Evaluation is required to update the UFSAR because: 1) NRC Information Notice 92-82 stated that Thermo-Lag was a combustible material, and 2) over the years Operations and Maintenance have requested to store various combustible materials in these fire zones.

This evaluation concludes that adequate protection is provided to assure the continued availability of redundant safe shutdown equipment and components with the additional combustible loading in these fire zones.

Revision 1 of this evaluation makes minor editorial changes, adds Class A combustibles in the form of piping insulation in Fire Area/Fire Zone K/14, the Reactor Containment, and adds piping insulation and miscellaneous Class A combustibles such as tools, personnel clothing, etc. used during normal work activities in Fire Area/Fire Zone J/24, the Pipe Tunnel.

SAFETY EVALUATION PSL-ENG-SENS-98-033
REVISION 1

RECONFIGURATION TEST FOR THE SPENT FUEL POOL

Summary:

FPL submitted to the NRC a license amendment request to reconfigure the Unit 2 spent fuel pool to increase the pool storage capacity. The current St. Lucie Unit 2 spent fuel storage rack modules have a total of 1584 cell locations, of which 1076 are available for fuel storage. License amendment 101 was approved by the NRC on May 6, 1999 and it permits the storage of 1360 fuel assemblies in these racks. At present, certain cells in each Unit 2 storage rack module contain mechanical blocking devices that prevent the placement of a fuel assembly in these locations. Two different blocking device designs are utilized at St. Lucie Unit 2; Region I utilizes blocking bars welded to the walls of some L-inserts, while the Region II design relies on a friction fit with the storage cell walls. To reconfigure the fuel pool to increase the storage capacity in accordance with the requirements of the revised Technical Specification, more than four hundred storage cell blocking devices must be removed. Removal of blocking devices and the repositioning of irradiated fuel required approval of license amendment 101 by the NRC.

Removal of blocking devices uses new processes and requires the manipulation of tools that are infrequently used. For this reason and others, such as the large number of blocking devices that will ultimately require removal, the differences in design between the welded and non-welded components and the potential for significant personnel radiation exposure, it is prudent to perform a limited scale blocking device removal test prior to undertaking the main blocking device removal campaign. The purpose of this safety evaluation was to define the conditions under which a small-scale test of Region I and II blocking device removal techniques may be performed.

Revision 1 to this evaluation updates text to note the NRC has approved license amendment 101. Revision 1 includes minor updates to the text to reflect more precisely how the reconfiguration test will actually be performed. Additionally, Revision 1 modifies or deletes some of the required actions because they are no longer required to safely perform the reconfiguration test.

SAFETY EVALUATION PSL-ENG-SENS-98-062
REVISION 0

1998 UFSAR REVIEW FINDINGS REQUIRING CHANGES OR CLARIFICATIONS TO THE FSARS IN ACCORDANCE W/50.59

Summary:

The purpose of this safety evaluation was to provide a method to update, correct, or add clarifications to the UFSAR following an UFSAR review project on selected systems. This safety evaluation contains a description of the change, its impact on plant safety, and the determination that the change does not involve an unreviewed safety question. User comments were issued for those comments, corrections, updates, or clarifications, which could be classified as administrative or editorial in nature. The subject matter of these changes involves the I&C and electrical fields as far as subject matter is concerned. Safety Evaluation PSL-ENG-SENS-98-075 addresses the other subject fields. This safety evaluation meets the requirements of 50.59 for making changes to the plant as identified in the UFSAR.

SAFETY EVALUATION PSL-ENG-SENS-98-075
REVISION 0

1998 UFSAR REVIEW FINDINGS REQUIRING CHANGES OR CLARIFICATION TO FSARS IN ACCORDANCE W/10CFR50.59 FOR I&C ISSUES

Summary:

The purpose of this safety evaluation was to provide a method to update, correct, or add clarifications to the UFSAR following an UFSAR review project on selected systems. This safety evaluation contains a description of the change, its impact on plant safety, and the determination that the change does not involve an unreviewed safety question. User comments were issued for those comments, corrections, updates, or clarifications, which could be classified as administrative or editorial in nature. The subject matter of these changes involves the I&C and electrical fields as far as subject matter is concerned. Safety Evaluation, PSL-ENG-SENS-98-062, will addresses the other subject fields. This safety evaluation meets the requirements of 50.59 for making changes to the plant as identified in the UFSAR.

SAFETY EVALUATION PSL-ENG-SENS-98-094
REVISION 0

LETDOWN RADIATION MONITOR - UFSAR CHANGES

Summary:

In the 1996 UFSAR Consistency Review, a discrepancy between Unit 2 UFSAR Section 9.3.4.2 and existing plant procedures was identified. Section 9.3.4.2 and Section 11.5.2.2.2 describe the function and use of the letdown radiation monitor, however, plant operating procedures have this monitor isolated with no instructions to align it for use.

The letdown radiation monitor, also referred to as the CVCS Process Monitor, was originally installed to alert plant operators to an increase in reactor coolant system (RCS) radioactivity primarily to indicate a potential fuel failure(s). However, this monitor is not being used due to its installed location. In contrast to the same monitor in Unit 1, which is installed in a shielded area on the 19.5 foot elevation, the Unit 2 monitor is located on the 43 foot elevation and is unshielded. The pipe routing to the Unit 2 monitor poses an ALARA problem since it may contain high radiation fluid in an unshielded area. This monitor does not perform any functions related to safe plant shutdown or functions that are required to mitigate the consequences of an accident. The option of relocating the monitor to a more suitable location has not been pursued since other means exist to monitor RCS activity. Therefore, this evaluation will provide the Justification to revise the UFSAR to acknowledge the isolation and abandonment of the letdown radiation monitor.

SAFETY EVALUATION PSL-ENG-SEMS-98-119
REVISION 1

OPERATION WITH MOTOR-OPERATED VALVE V3523 OPEN

Summary:

Condition Report 98-2096 identified a degraded condition of V3523, a containment isolation valve on the Hot Leg Injection line of the High Pressure Safety Injection System in St. Lucie Unit 2. The Hot Leg Injection piping is used for long term cooling for an accident (per Section 6.3.2.2.3 of the UFSAR). The High Pressure Safety Injection (HPSI) pumps are manually re-aligned for simultaneous Hot and Cold Leg Injection. This ensures flushing and ultimate sub-cooling of the core coolant independent of break location.

The change proposed by this safety evaluation is to "lock open to its throttled position" valve V3523, and use the second, redundant valve in the line, V3551, as both the containment isolation valve and the Hot Leg isolation valve. The change places V3523 in its "safe position" for Alternate Hot Leg and Cold Leg Injection. In addition, the safety function of V3523 for containment isolation and Hot Leg isolation is transferred to V3551. Since the quality and level of protection of the original design basis has been maintained, this change has no effect on plant operation or safety.

The proposed change requires a change to plant procedures which normally electrically key lock close the valves. This change has no effect on power operation and does not place any restriction on plant operation. In addition, valve V3551 was de-energized to provide single failure protection under specific plant scenarios. Therefore, the re-energization of valve V3551 requires that adequate direction be provided to plant Operating personnel prior to the need for simultaneous Cold Leg-Hot Leg Injection. The proposed change is an adequate measure until the next outage of sufficient duration at which time the valve (V3523) will be repaired.

Revision 1 addresses comments from the Facility Review Group related to documentation of the acceptability of manual actions.

SAFETY EVALUATION PSL-ENG-FPER-99-006
REVISION 0

FIRE PROTECTION EVALUATION TO DETERMINE FIRE RATING FOR UNFILLED CMUs OF VARIOUS CONFIGURATIONS

Summary:

The purpose of this safety evaluation was to demonstrate that single wythe hollow CMU wall assemblies (nominal 12" or greater in thickness) or multi-wythe hollow CMU wall assemblies (nominal 8" or greater in thickness) can be implemented without prior NRC approval in 3-hour fire resistance rated wall assemblies. The basis for this determination were documented in this evaluation. This evaluation was necessary to justify the existing CMU wall assemblies which are 12" thick or greater are acceptable for the "as installed" plant conditions for 3-hour fire resistance rating. In addition, it was necessary to update the UFSAR by providing the minimum thickness concrete masonry wall assemblies shall have in order to meet the 3-hour fire resistance rating.

SAFETY EVALUATION PSL-ENG-SEFJ-99-012
REVISION 0

SAFETY ANALYSIS REQUIREMENT FOR PWR CALIB DURING POWER ASCENSION DUE TO POWER DEPENDENT PWR MEASUREMENT UNCERTAINTY

Summary:

The purpose of this evaluation was to address the issue related to the increased secondary calorimetric uncertainties at lower power levels. The scope of work includes the assessment of plant specific calorimetric power measurement uncertainties as compared to the values used in the St. Lucie Units 1 and 2 analyses, and provide guidelines for power calibration requirements to be incorporated into the appropriate plant procedures.

ABB-Combustion Engineering (ABB-CE) had previously issued Info Bulletin 94-01 regarding a potential 10 CFR 21 issue related to the secondary calorimetric power measurement (SCPM) uncertainties, which may exceed 2% of rated power at reduced power levels. The increased uncertainties are associated with the feedwater flow transmitter drift and the calibration temperature effects. Conservative power calibration requirements were imposed at that time to satisfy safety analysis acceptance criteria.

SAFETY EVALUATION PSL-ENG-SEMS-99-027
REVISION 0

HYDRANTS, HOSE HOUSES & HOSE STATIONS - COMPARISON OF THE FSAR VS. FIRE FIGHTING STRATEGIES

Summary:

This Safety Evaluation compares the assumptions made in the St. Lucie Unit 1 and 2 Final Safety Analysis Report for the use of fire protection suppression equipment (i.e., fire hydrants, hose houses, and hose stations) to the Fire Fighting Strategies per AP 1/2-1800023, evaluates the differences and identifies changes.

SAFETY EVALUATION PSL-ENG-SEES-99-031
REVISION 0

OPERATION OF UNIT 2 DIESEL GENERATOR WITHOUT KW RECORDER

Summary:

During several surveillances of the Unit 1 and 2 Diesel Generators, failure of the diesel's watt recorder has caused Operations to cease testing and declare the Emergency Diesel Generator (EDG) inoperable until the attendant instrumentation has been returned to service. In addition several Condition Reports have been issued to request that Engineering address operability of the diesel without the use of the watt recorder.

Operation of the Unit 1 EDG without use of the watt recorder was previously evaluated under Safety Evaluation PSL-ENG-SEES-99-016. The purpose of this Safety Evaluation was to determine the acceptability of operating the Unit 2 Diesel Generator in the same manner and to provide operator guidance in the use of diesel current and other instrumentation for determining diesel load during surveillances and emergency load management conditions.

This evaluation concludes that the Unit 2 EDGs may continue to satisfy their safety functions and may be monitored for proper performance without the use of the EDG watt recorder(s) provided procedures are revised using the guidance in this evaluation. In conclusion, operation of the Unit 2 EDGs without use of the watt recorders does not pose an operability question nor does it reduce the margin of safety as defined in the basis for any Technical Specification or the UFSAR.

SAFETY EVALUATION PSL-ENG-SEMS-99-032
REVISION 0

EVALUATION OF THE REACTOR CAVITY, REACTOR SUPPORT AND CONTAINMENT COOLING SYSTEMS

Summary:

Condition Report 98-1556 identified apparent discrepancies related to the reactor cavity and reactor support cooling systems. The discrepancies were associated with the operational requirements of the system, the UFSAR descriptions of the systems, and maintenance of the reactor support and reactor cavity cooling systems. Condition Report 98-1556 and Supplement 1 to Condition Report 98-1556 addressed the identified conditions, and determined there were no operability concerns associated with the apparent discrepancies.

While the general philosophy of PC/M 86013 was correct, the supporting justification was incomplete. The supporting justification relied on monitoring fan operation and ensuring a minimum combination of fans was running. The monitoring requirements did not address the future performance of the operating fans.

Therefore, the reactor cavity and reactor support cooling systems continue to operate as originally designed, with some temperature indications out of service. The original design never specified the minimum number of operable channels required to properly monitor the reactor supports.

This evaluation develops and justifies the operational and monitoring requirements for the reactor cavity and reactor support cooling systems. This evaluation addresses the minimum redundancy required for monitoring the reactor cavity and reactor supports. This evaluation specifies the compensatory measures required if temperature indication is not maintained (alternate flow monitoring). Finally, this evaluation supports any changes to procedures ONP-1-2000030 and ONP-2-2000030 to ensure consistency between St. Lucie Unit 1 and St. Lucie Unit 2 in response to a loss of Reactor Containment Building (RCB) cooling fans.

An additional issue was raised in Condition Report 98-1556 involving a lack of consistency between the safety and seismic classifications of various flow switches and temperature elements that are part of the containment cooling, reactor cavity cooling and reactor support cooling systems. This evaluation provides justification for safety and seismic classification changes as needed to address this issue.

SAFETY EVALUATION PSL-ENG-SEMS-99-034
REVISION 0

UPDATE OF FLOW CAPABILITIES OF STEAM DUMP AND BYPASS VALVES PCV-8801 THROUGH PCV-8805

Summary:

This evaluation addresses the flow capabilities of the steam dump and bypass valves (PCV-8801 through PCV-8805) on Units 1 and 2 and provides for revision of the UFSAR and DBD. The UFSAR and DBD currently state that the steam dump and bypass valves are capable of a load rejection of 45%. Calculation PSL-2FSM-95-015 has determined that the steam dump and bypass valves are capable of a load rejection of 29% (stretch power). The effective stroke length of the valves was reduced by previous actuator modifications, which resulted in a lower flow coefficient (C_v). Based on the C_v change and using the Stretch Power steam flow rates, the percentage of full flow through the valves is decreased.

SAFETY EVALUATION PSL-ENG-SEMS-99-039
REVISIONS 0 & 1

USE OF PROCUREMENT ENGINEERING'S GENERIC EVALUATIONS BY THE NUCLEAR PROCUREMENT & LOGISTICS DEPARTMENT

Summary:

This safety evaluation demonstrates that the practice of the Pre-Screen Reviewer(s) from the Nuclear Procurement & Logistic Department (known as Nuclear Materials Management Department on site) to independently use Procurement Engineering's (PE) approved Generic Evaluations for stock code item dispositions is acceptable. Currently, the Pre-Screen Reviewer routes Stock Codes to Procurement Engineering and asks for authorization from PE to use a certain Generic Evaluation if he/she thinks it is applicable. If the Procurement Engineer concurs that the Generic Evaluation is applicable, a statement is made as such via a single signature electronic PE Evaluation. The stock code is then routed back to the Pre-Screen Reviewer to apply the Generic Evaluation.

SAFETY EVALUATION PSL-ENG-SENS-99-040
REVISION 0

CHANGE TO UPDATE STEEL CONTAIMENT VESSEL ANALYSIS RESULTS TO REFLECT THE LATEST ANALYSIS BY CBI

Summary:

The purpose of this safety evaluation is to provide a method to update, correct, or add clarifications to the Unit 2 UFSAR section 3.8 following an UFSAR review project on the Unit 2 Steel Containment Vessel. This safety evaluation contains a description of the changes, their impact on plant safety, and the determination that the change does not involve an unreviewed safety question.

SAFETY EVALUATION PSL-ENG-SENS-99-059
REVISION 0

1999 UFSAR REVIEW FINDINGS REQUIRING CHANGES OR CLARIFICATIONS TO THE FSARs IN ACCORDANCE W/50.59

Summary:

The purpose of this safety evaluation is to provide a method to update, correct, or add clarifications to the UFSAR following the 1999 UFSAR Review Project reviews of selected risk significant systems. This safety evaluation contains a description of each change, its impact on plant safety, and the determination that the change does not involve an unreviewed safety question. The subject matter of these changes involve the following:

- Unit 1 Safety Injection Pipe Whip Restraints
- Unit 1 Containment Spray Suction Design Pressure
- Unit 1 Component Cooling Water FMEA Correction
- Unit 2 Component Cooling Water Pump Suction and Discharge Isolation Valve Positions
- Unit 1 Chemical and Volume Control Heat Exchanger Performance Monitoring; Boric Acid requirements
- Unit 2 Chemical and Volume Control Boric Acid requirements and pump control, Boron concentration; relief valve setpoint; Boronometer vessel design temperature; pneumatic valve V2523

SAFETY EVALUATION PSL-ENG-SECS-99-060
REVISION 0

UNITS 1 & 2 SITE STORM WATER DRAINAGE SYSTEM REDUCTION IN LEVEL OF DETAIL WITHIN FSAR

Summary:

During review of the Unit 1 and Unit 2 UFSARs it was questioned whether the site storm drainage flow calculations have been updated relative to recent paving of various areas. While updating the calculation it was found that within each UFSAR, details pertaining to the design of the site storm drainage system provide a level of detail above and beyond that required by Regulatory Guide 1.70. Specifically, Section 3.4.4 of the Unit 1 UFSAR and Section 3.4.1 provide the methodology and precise parameters used in determination of the storm water flow and subsequent sizing of the drain. As the site is developed the precise parameters may change (i.e., unpaved sections may have different runoff coefficients, etc.) and therefore the UFSAR would need to be updated. This Safety Evaluation provides the basis for the acceptability to remove this level of detail from each UFSAR with documentation of the storm water drainage design within a unique calculation. It also demonstrates that the required changes to each UFSAR meets all technical and licensing requirements for St. Lucie Units 1 & 2 and remains consistent with the requirements of Regulatory Guide 1.70.

Both Unit 1 and Unit 2 UFSAR state that the site storm water drainage system is designed to preclude flooding of Safety Related structures, systems and components under maximum precipitation conditions (PMP or PMH), however total flooding of the drain lines will not cause water to backup into areas which would jeopardize the required function of a Safety Related system. This evaluation concludes that this statement remains valid and that the level of detail quoted within each UFSAR is not required.

SAFETY EVALUATION PSL-ENG-SEMS-99-064
REVISION 1

REVIEW OF UNIT 2 SDC SYSTEM OPERATION

Summary:

Shutdown Cooling (SDC) System operation is initiated at a point in the plant cooldown when the RCS conditions drop below SDC design conditions. The system continues to remove decay and sensible heat until the RCS is reduced below a nominal temperature of 135°F. The Unit 2 SDC System is described in UFSAR Section 5.4.7. Condition Report 98-1749 was issued to address the inability to maintain Unit 2 RCS temperature during mid-loop operation due to a degraded SDC system. The cause of the degraded Unit 2 SDC system performance was determined to be excessive seat leakage through the heat exchanger bypass valves. The valves had been degraded by cavitation of the seats during throttled flow operation. This evaluation performs a review of the Unit 2 SDC system design and operation including single train operation and SDC failure scenarios. Recommendations are provided for procedural changes and operational limits that should minimize the cause and impact of heat exchanger bypass valve leakage as well as other SDC design considerations. Partially drained and mid-loop operation is reviewed to establish RCS level and SDC flow limits to ensure the reliability of SDC.

Similar to the evaluation of the Unit 1 SDC system, adverse Unit 2 SDC suction conditions caused by flashing at system high points were determined to be limiting at reduced RCS levels. LPSI Pump NPSH and vortexing limits also apply at specific operating conditions. Operating limits designed to preclude loss of SDC due to credible failures in consideration of the suction restrictions have been defined and are included herein. The most important information resulting from this review of SDC system performance is the difficulty in recovering SDC at mid-loop conditions once RCS temperature exceeds 190°F. If SDC A Train is lost at a pre-fuel shuffle mid-loop condition, there is limited time for the operators to respond (approximately seven minutes for A Train recovery). This time may be insufficient for the operators to diagnose and correct the initiating problem and then re-establish A Train SDC at a flow rate adequate to maintain RCS temperature. B Train SDC can be recovered at saturation conditions in the RCS. Failure to expeditiously re-establish SDC will result in containment evacuation due to core boiling. Maintaining the RCS at saturation conditions at reduced inventory precludes restoration of SDC A Train in any reasonable time. It is expected that B Train, if available, could be re-started to restore the RCS temperature. If hot mid-loop operation is to be employed, it is recommended that the time spent in this condition

be minimized to reduce the overall risk of SDC loss.

Unit 2 UFSAR changes were provided by this Safety Evaluation to align the Unit 2 UFSAR description of SDC operation with system design characteristics. Procedural guidance and limits are also provided for operating the SDC system in such a manner that will reduce flow control and bypass valve seat damage due to significant throttling and establish limits for SDC operation in a mid-loop configuration.

A number of issues that were raised as a result of the Unit 2 SDC event during the SL2-11 outage are being addressed by this Safety Evaluation.

Revision 1 to this Safety Evaluation was issued to add information and applicable UFSAR changes associated with the thermal performance of SDC in the emergency shutdown scenario for stretch power. Associated DBD changes were also included. Alternate means of defeating the SDC suction valve auto-close interlock were incorporated to avoid outage critical path impact due to post maintenance testing of control circuit modifications.

SAFETY EVALUATION PSL-ENG-SEIS-99-067
REVISION 0

REMOVAL OF A FAILED RCS COLD LEG TEMPERATURE INPUT SIGNAL FROM THE QSPDS RCS SATURATION MARGIN CALCULATION

Summary:

The Qualified Safety Parameter Display System (QSPDS) was designed in response to the post TMI NUREGs to provide detection of inadequate core cooling (ICC). The three main ICC display functions provided by QSPDS are saturation margin monitoring (SMM), reactor vessel level monitoring and core exit thermocouple (CET) monitoring.

This safety evaluation involves the temporary removal of a failed RCS RTD input signal from the saturation margin calculation. The loop 2A1 cold leg temperature RTD, TE-1112CB, has exhibited erratic behavior that has affected the validity of the calculated RCS saturation margin. It is desirable to remove the bad input signal to ensure the validity of the RCS SMM calculation, which will then be based on the maximum of the two remaining RTD input signals.

SAFETY EVALUATION PSL-ENG-SEES-99-068
REVISION 1

CROSSTIE OF THE SPENT FUEL POOL PUMPS POWER SUPPLY

Summary:

During plant refueling outages, electrical system maintenance is performed which requires the de-energization of either 480V motor control center (MCC) 2A8 or 2B8. These MCCs power the spent fuel pool pumps, among other loads. The UFSAR states that only one fuel pool pump is needed to be operating to maintain adequate cooling for the spent fuel pool with a full reactor core discharge. In order to protect against an active failure during full fuel core offloads, it is desired to have power available to and operate both spent fuel pool pumps when either 480V MCC 2A8 or 2B8 is out of service. This can be done by installing a temporary power feed from the MCC remaining in service to the pump normally powered by the MCC to be removed from service.

Implementation of the cross-tie consists of the following steps:

- installing a spare 100 amp circuit breaker in a spare compartment of the MCC remaining in service,
- disconnecting the line side of the spent fuel pool pump circuit breaker in the MCC to be removed from service,
- routing a temporary cable between the MCCs from the load side of the spare circuit breaker to the line side of the spent fuel pool pump circuit breaker.

In order to meet the requirements for electrical train separation and isolation, the temporary jumper must meet the following requirements:

- The cable used for the jumper must meet the flame test requirements of IEEE 383-1974.
- Electrical isolation from safety related power sources must be provided by the use of properly coordinated protective devices (circuit breakers).
- The jumper cable must not be routed in any raceways or conduits that contain safety related cables and must not be supported from any seismic supports or safety related equipment.

SAFETY EVALUATION PSL-ENG-SENS-00-002
REVISION 0

REPLACEMENT OF FIRE WATER PIPE PENETRATING THE RAB

Summary:

Condition Report CR 99-2579 identified a 2-3 gpm through wall leak in the 6" fire water header. This leak was located near the flange on line 6-FP-108 at the penetration of an exterior wall to the Reactor Auxiliary Building (RAB) at the -0.5 floor elevation. This leak has currently been terminated via a temporary repair. Replacement of this section of pipe would require extraction of the existing pipe leaving an opening in the RAB wall. The presence of this opening with Unit 2 at power was evaluated in the safety evaluation.

SAFETY EVALUATION PSL-ENG-SENS-00-009
REVISION 1

TEMPORARY STRUCTURE MODIFICATION FOR MSIV HCV-08-1A ACTUATOR

Summary:

The valve actuator on Unit 2 MSIV HCV-08-1A was removed and refurbished during the Unit 2 Cycle 12 refueling outage. This activity required removal of some steam trestle steel grating and/or modification to accommodate rigging. The steel grating provides protection against external missile impacts for the steam trestle. In addition, seismic structure and support members were temporarily affected, including pinning of spring hangers for the associated main steam line (a water solid environment will be established to provide lubrication in order to stroke the valve). With the valve actuator removed, normal refueling operations were conducted. Additionally, direct paths from the RCS to the containment environment via manways and access holes were in operational condition during the time the valve actuator was removed. As such, HCV-08-1A in the closed position provides a containment penetration barrier during "Core Alterations."

The purpose of this safety evaluation was to demonstrate that the temporary activities associated with the modifications and attachment of rigging, rigging support members, rigging loads, steel grating removal/reinstallation and support members/pinning of spring hangers did not adversely impact HCV-08-1A as a containment penetration system barrier during Cycle 12 refueling operations. HCV-08-1A, with its actuator removed and with the valve plug in the closed position, is capable of performing its intended function as a containment penetration barrier during refueling operations (Tech. Spec. 3.9.4). This temporary configuration has been evaluated and it has been determined to be acceptable for refueling operating conditions.

Revision 1 to this safety evaluation provides justification to relax the original recommendation to suspend core alterations during the actual lifting/removal and subsequent reinstallation of the actuator in the vicinity directly overhead of the valve. The change to relax this recommendation however is based on conformance to Technical Specification 3.9.4 which requires that if the containment building penetration requirements are not satisfied, immediately suspend all operations involving core alterations or movement of irradiated fuel in the containment building. Likewise, if direct paths from the containment environment via S/G manways and access holes are secured, no requirements are imposed since specification 3.9.4 is fully met for this activity.

SAFETY EVALUATION PSL-ENG-SENS-00-013
REVISION 1

USE OF PRC-01 RESIN IN THE CVCS LETDOWN STREAM TO REMOVE CO-58 CONTAMINATES

Summary:

This Safety Evaluation has been prepared to examine the use of an additional resin material in the chemical and volume control system (CVCS) ion exchangers during the plant cooldown and refueling evolution at St. Lucie Unit 2. The material to be utilized during the conditions specified in this safety evaluation is an acrylic substrate weak acid resin identified as PRC-01 that will increase the decontamination factor (DF) of the CVCS cleanup system for sub-micron sized particulates of Cobalt 58. In the presence of a neutron flux, Cobalt (Co) 58 is produced through the absorption of a neutron by Nickel (Ni) 58 in an (n, p) reaction. Nickel is a constituent element in structural materials used to fabricate portions of the St. Lucie reactor coolant system (RCS).

During power operation at St. Lucie a corrosion layer forms on interior surfaces of the RCS. A portion of this corrosion layer is comprised of a stable inner oxide that serves to passivate the surface and limit future base metal corrosion. In addition to this region of relatively stable oxide, an outer transient corrosion film also forms during plant operation. This film consists of a non-stoichiometric mixture of nickel ferrites, nickel oxides and base metal nickel. The transient film also contains significant quantities of Cobalt-58 that results from the Nickel (n, p) reaction described above. The transient nature of this corrosion layer is due to changes in the operating environment that effect the release and deposition of corrosion products. Coolant-borne corrosion products are also deposited on and released from fuel surfaces. The stability and behavior of these corrosion products change significantly when the RCS is opened during an outage and then re-closed for the following cycle's operation. To control the timing and the rate of release of these corrosion products, most PWRs (including St. Lucie Unit 2) employ actions to decompose nickel and cobalt compounds and increase their solubility prior to opening the RCS. Usually, these actions are incorporated into plant shutdown procedures. If these actions are successful in establishing the correct chemical form of corrosion products the CVCS cleanup system can remove the corrosion products through filtration and ion exchange.

Sometimes the actions taken to dilute corrosion products are less than fully successful; thus, some inventory of corrosion products may remain as insoluble particulate matter. Additionally, some corrosion products formed during, power operation, such as nickel oxide, are not soluble in the acid reducing conditions normally established just after shutdown. As discussed in this evaluation, particulates are not efficiently removed by resins in the current mixed bed ion exchanger and, depending on particle size, may not be removed by installed CVCS purification filters. The PRC-01 resin proposed for use has a demonstrated high affinity for corrosion products in the particulate form. Use of this resin material will assist in reducing the occupational dose incurred during refueling outages.

This evaluation concludes that, subject to some constraints, PRC-01 resin may be effectively and acceptably applied to refueling cleanup of the RCS. This evaluation does not authorize the use of PRC-01 in other safety-related applications or to clean CVCS effluent during power operations at St. Lucie Unit 2.

Revision 1 to this Safety Evaluation has been prepared to make minor modifications to Plant Restrictions, to add and more precisely paraphrase UFSAR text and to include an additional required action related to calibration of the CVCS ion exchanger bypass valve.

SAFETY EVALUATION PSL-ENG-SEMS-00-016
REVISION 0

USE OF 1/8" PERFORATED STRAINER MESH IN ICW STRAINERS SS-21-1A/B

Summary:

This evaluation reviews the acceptability of using 1/8" perforated strainer mesh material within the Unit 2 ICW Strainers serving the CCW Heat Exchangers (SS-21-1A, 1B). The originally specified 1/4" perforated 316 SS strainer mesh for use within strainers SS-21-1A and B is not available for replacement of the currently degraded strainer elements. Installation of 1/8" perforated strainer mesh is proposed as it is available, is used within the comparable Unit 1 strainers, and is suitable for use in the Unit 2 strainers.

SAFETY EVALUATION PSL-ENG-SEFJ-00-016
REVISIONS 0

UFSAR CHANGE FOR CHAPTER 10 AFW EVALUATION ANALYSES

Summary:

The purpose of this evaluation is to update the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR) to incorporate a revised conservative best estimate reactor coolant pump (RCP) heat value into the Chapter 10 auxiliary feedwater (AFW) system evaluation analyses. The proposed changes affect the UFSAR Section 10.4.9A. The affected analyses are the Loss of Main Feedwater (LOMF) with offsite power available and the Feedwater Line Break (FLB) with offsite power available. The effect of increased RCP heat on these events with loss of offsite power is insignificant since the RCPs are tripped immediately after the reactor trip, and the RCP heat prior to the reactor trip has no significant impact on these analyses.

The best estimate RCP heat was revised in 1994 as part of the ground rules verification process and the impact of the increased heat was evaluated as being acceptable from analysis consequences viewpoint as documented in the body of this evaluation. The UFSAR, however, was not changed to reflect this revised RCP heat. This inconsistency in the UFSAR was identified during the recent UFSAR review work.

SAFETY EVALUATION PSL-ENG-SEMS-00-017
REVISION 0

UPDATE OF SDC SUCTION LINES MODERATE ENERGY LINE BREAK ANALYSIS

Summary:

This evaluation reviews the acceptability of updating the moderate energy line break analysis for the Shutdown Cooling (SDC) suction line leak within the A or B ECCS compartment to use V3481 in lieu of V3664 and V3651 in lieu of V3665 for mitigating the leak path.

CR 00-0790 identified that the design basis calculation for differential pressure across V3664 and V3665 and the calculation for actuator capability did not correctly address design basis requirements. Review of the V3664/V3665 differential pressure design basis requirements concluded that the analysis within UFSAR Appendix 3.6F.2.2.1 (a) implies V3664 and V3665 are capable of remote manual-closure to isolate postulated moderate energy line breaks within the A or B ECCS compartment. Actuator modifications to fully qualify V3664 and V3665 for remote manual closure against the design basis pressure drop is technically feasible but difficult to achieve with a high level of confidence.

SAFETY EVALUATION PSL-ENG-FPER-00-019
REVISION 1

EVALUATION OF VARIOUS ITEMS FOR APPENDIX R SEPARATION IN UNIT 2 CONTAINMENT

Summary:

The purpose of this evaluation was to address the requirements of 10 CFR 50.59 for UFSAR changes related to the separation of certain redundant safe shutdown components in the Unit 2 containment. This evaluation also addresses the licensing issues of such a change in relation to the operating license condition for fire protection. The evaluation considered the need for a license amendment and/or the need to submit the items to the commission as deviations to Appendix R and determined that a license amendment or deviation submittal is not required.

Certain redundant safe shutdown components were determined from engineering reviews and/or walk down to be separated by less horizontal distance than that required by Appendix R or accepted by NRC SER. This evaluation includes fire hazards analyses that evaluate the physical separation and/or fire protection features for certain items at a specific location in the Unit 2 containment. The fire hazards analyses ascertain and establish that adequate separation between redundant systems/components is provided at the areas of concern. The items evaluated include separation between:

- Conduit 25018Y (circuit for Pressure Transmitter PT-1108) and Pressure Transmitter PT-1107
- Circuits for Auxiliary spray/charging isolation valves (I-SE-021, 2, & 4) and PORV/PORV block valve (V1474 & V1476)
- Pressurizer Level Instrument Racks (IR-51-1 & IR-36-12A)
- Pressurizer Pressure Instrument Racks (IR-53-15 & IR-53-16) and
- Conduits 25011A, 23058B and 23059B for Nuclear Instrumentation RE-001-A2, RE-26-80A, and RE-26-80B, respectively

SECTION 3

RELOAD SAFETY EVALUATIONS

RELOAD SAFETY EVALUATIONS 99163

ST. LUCIE UNIT 2 CYCLE 12 RELOAD

Summary:

This Engineering Package provides the reload core design of St. Lucie Unit 2 Cycle 12 developed by Florida Power & Light Company (FPL) and Asea Brown Boveri-Combustion Engineering (ABB-CE). The Cycle 12 core is designed for a cycle length of 12,890 EFPH, based on a nominal Cycle 11 length of 11,583 EFPH. The Cycle 12 reload design supports an additional end-of-cycle coastdown length of ~360 EFPH at full power with a maximum reduction in primary coolant inlet temperature to 535°F. Cycle 11 is expected to achieve an end-of-cycle exposure of ~11,500 EFPH.

The primary design change to the core for Cycle 12 is the replacement of 77 irradiated fuel assemblies (72 Region L assemblies and 5 Region K assembly) with 76 fresh Region P fuel assemblies and 1 irradiated Region K fuel assembly currently residing in the spent fuel pool. The fuel in the Cycle 12 core is arranged in a low leakage pattern similar to the design of the Cycle 11 core. The mechanical design of Region P fuel is essentially the same as that of Region N fuel, and consists of "value-added" fuel pellets and "guardian grid" design, first introduced in Cycle 11. Changes to Region P fuel include first time use of enriched axial blankets for U02 rods.

The implementation instructions provided in this EP, for core reconfiguration from Cycle 11 to Cycle 12, support a full core off-load only.

The safety analysis of this design was performed by ABB-CE and by FPL using NRC approved methodology. The core design and the generation of physics inputs to safety are performed by FPL using the Westinghouse physics methodology. The NRC has previously approved the use of Westinghouse physics methods in combination with ABB-CE safety analysis methods for St. Lucie Unit 2.

Cycle 12 reload is the first reload for the implementation of the Reload Process Improvement (RPI) designed to provide flexibility in the core design. This approach uses a checklist format to assess cycle-specific core designs for compliance with the existing safety analysis. Technical Specifications changes and the safety analyses supporting the implementation of RPI have been approved by the NRC for Cycle 12.

The analyses for Cycle 12 supports a change to the Core Operating Limits Report (COLR) MTC limit from -30 pcm/ºF to -32 pcm/°F, and a change to the COLR Shutdown Margin Limit for Modes 1, 2, 3, and 4 from 5000 pcm to 4500 pcm. All analyses in support of this EP support a maximum steam generator tube plugging level of 1250 tubes per steam generator with a maximum asymmetry of 400 tubes.