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

April 12, 2000

L-2000-086
10 CFR 50.59

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

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

Pursuant to 10 CFR 50.59 (b)(2), the enclosed report contains a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each which were made on Unit 1 during the period of January 7, 1998 to October 19, 1999. This submittal correlates with the information included in Amendment 17 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

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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 1
Docket No. 50-335
L-2000-086
Enclosure

ST. LUCIE UNIT 1
DOCKET NUMBER 50-335
CHANGES, TESTS AND EXPERIMENTS
MADE AS ALLOWED BY 10 CFR 50.59
FOR THE PERIOD OF
JANUARY 7, 1998 THROUGH OCTOBER 19, 1999

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 January 7, 1998 through October 15, 1999. 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. 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, which evaluated the specific change(s). Each of these safety evaluations concluded that the change does not represent an unreviewed safety question or requires 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 95098

REVISION 3

RCP 'A' VAPOR SEAL LEAK-OFF LINES

Summary:

St Lucie Action Report (STAR) 950323 documented a problem associated with the Unit 1 & 2 reactor coolant pumps (RCP) "A" vapor seal leak-off lines. Accumulation of boric acid crystals on the top of the RCP seals had been attributed to the lack of proper drainage from the vapor seal assemblies. This condition caused the concern that degradation of the seal flange bolting might be initiated by the presence of boric acid. The vapor seal leak-off lines are 3/4" Quality Group D, non-seismic lines that connect to the vapor seal drain cavity via a flanged connection at the pump. These leak-off lines are located on the downstream side of the RCP vapor seals and are used to route vapor seal leakage to the Reactor Drain Tank.

While Unit 1 was in a forced outage, the 1A1 and 1A2 RCP vapor seal leakoff lines were temporarily rerouted to a local floor drain in lieu of the reactor drain tank. This work was evaluated under JPN-PSL-SEMS-95-015 and was performed under Jumper/Lifted Lead JLL 1-95-037. The normal system configuration is a closed system (not open to atmosphere) that drains to the reactor drain tank. The configuration documented by this PC/M is an open system that drains to the containment sump via open floor drains. The radiological and operational effects of this change have been determined to be insignificant due to the relatively small amount of leakage (approximately 0.3 gallons per hour per seal).

Revision 0 of this Engineering Package was issued to document the change provided by the above evaluation and allow closure of the JLL.

Revision 1 allows for similar modification of the 1B1 RCP vapor seal leakoff line. STAR 951360 documented leakage of 20-30 drops per minute at the seal, indicative of improper drainage through the leakoff line.

Revision 2 provided justification to support the existing "temporary" routing of the drain lines to a floor drain for Cycle 14 and documented various editorial and administrative corrections.

Revision 3 provides justification to qualify the existing routing of the vapor seal leak-off drain lines for the 1A1, 1A2 and 1B1 RCPs for permanent use, and to modify the 1B2 RCP vapor seal leak-off line, permanently routing it to a local floor drain and capping the unused section going to the Reactor Drain Tank.

PLANT CHANGE/MODIFICATION 96081

REVISION 4

REPLACEMENT OF FIRE PENETRATION SEAL 050-S-2 WITH TYPE M-9 FIRE PENETRATION SEAL

Summary:

Revision 4 of this Engineering Package (EP) provides the engineering analysis and design information to grout the portion of concrete masonry block wall #86 (BW86) common with the Laundry and Stairwell RA-7 at RAB elevation 19.5'. The grouting of the wall is required in order to maintain a three-hour fire rated barrier between fire zones 55w(C) and 36(O). As a result of the disposition of Condition Report 98-0689, it was determined that BW86 was hollow and required to be grouted for compliance with the Unit 1 UFSAR Section 3.11.2 of Appendix 9.5A.

Revision 3 of this EP provided the engineering analysis and design information to install three-hour fire barrier seals at the gaps at the top of concrete block walls where the cap is greater than one inch. Specifically, this revision addressed block walls ceiling joints for BW 123, 124 and 125. The design is similar to the design generated by revision 2 in that the same materials are used as the penetration seal materials. This design pertains to all gaps greater than 1" and no larger than 3.125". The major difference in the larger gap design is the addition of a steel backing plate on one side of the wall to assure there is sufficient strength to withstand a hose stream test after a three-hour fire. On the other side of the wall the thickness of the sealant material has been increased to assure the adhesion to the concrete will securely hold the Durablanket in place. This design meets or exceeds the requirements set forth in PSL-FPER-96-009, Revision 1, for sealing joints at the top of concrete block walls.

Revision 2 of this EP provided the engineering analysis and design information to install seals in the gaps at the top of Safety Related, seismically supported, concrete block walls between two fire areas. These seal characteristics are similar to and are bounded by the type M-7 seals. Since these walls are Safety Related and seismically supported they will be temporarily supported, where necessary, to the level of their permanent supports at all times while the seals are being installed. A ceiling joint above a concrete block wall is considered the same as a penetration seal because it uses the same seal material and it is required to make the wall a three-hour barrier as well as possibly providing an air barrier. Although the design of the ceiling joints look different from the design of the M-7 seal, their philosophy and technical requirements, such as being a fire barrier and an air barrier are the same as a fire penetration seal.

Revision 1 of this EP provided engineering analysis and design information necessary to replace or add fire seals, that are described on various design drawings with a type M-7 seal in the fire barriers. Note that the M-9 type seal in Revision 0, was renamed to M-7 in Revision 1, to agree with the nomenclature of the vendor, Promatec. These seals have been identified as needing modification or initial installation. As noted on RAB Penetration Schedule, Drawing 8770-B-054, all presently installed seals listed in this EP are type M-3 which are depicted on Penetration Barrier Fire Stop Assembly Detail M-3, Drawing No. 8770-10752. Some seals also have a boot type design as shown on Penetration Barrier Fire Stop Assembly Detail M-6, Drawing No. 8770-10754.

Revision 0 of this EP provided the engineering analysis and design information necessary to replace fire penetration seal 050-S-2 that is described on design drawing 8770-G-594, Sheet 12 with a type M-9 fire penetration seal, without the 88B high temperature seal.

PLANT CHANGE/MODIFICATION 97038

REVISION 2

FIRE ZONE 57B ('B' INVERTER ROOM) THERMO-LAG WALL AND CEILING REPLACEMENT

Summary:

This Engineering Package replaces the Thermo-Lag walls and ceiling of the St. Lucie Unit 1 "B" Inverter Room (Fire Zone 57B).

The NRC provided notification that Thermo-Lag fire barrier materials had failed to meet its intended barrier ratings during fire endurance testing via Generic Letter 92-08, Bulletin 92-01 and Bulletin 92-01, Supplement 1. In FPL letter L-92-273, St. Lucie committed to compensatory measures (i.e., roving fire watch and verification of fire detector operability) until the Thermo-Lag fire barrier issues were resolved.

PSL utilizes Thermo-Lag 330-1 firewalls as fire barriers between certain adjacent fire areas. These walls are constructed out of steel frames with Thermo-Lag 330-1 panels bolted to the frames. The Thermo-Lag firewalls were originally designed as three-hour rated fire barriers (as defined) or meeting the requirements of ASTM E-119. As a fire barrier, Thermo-Lag walls are required to: 1) minimize the temperature increase on the unexposed side of the barrier; 2) prevent the passage of flames through the unexposed side; and 3) prevent projection of water beyond the unexposed side due to a hose stream test following constant exposure to a fire for a duration of three hours.

A specific evaluation of the Thermo-Lag walls and ceiling of the "B" Inverter Room (Fire Zone 57B) was performed and documented in PSL-FPER-97-001 to obtain additional details on the walls and ceiling to determine accessibility for modifications. In lieu of upgrading the existing Thermo-Lag wall panels and ceiling, the evaluation provided an alternative of removing the Thermo-Lag barriers and replacing them with a sheet metal and ceramic fiber barrier. Based on accessibility and experience from the PSL Unit 2 modifications, the alternative barrier configuration was chosen. This configuration is supported by both analytical and test documentation.

This EP provides the required modifications to implement the alternative fire barrier consisting of sheet metal and ceramic fiber as identified by the FPER evaluation and documents the basis for the acceptability of the sheet metal and ceramic fiber barrier not meeting all acceptance criteria for a 3-hour fire rating in compliance with ASTM E-119.

Revision 1 added a HOLD POINT for accepting the new Inverter Room Fire Barrier as "operable" until the new design for the walls and ceiling had been shown to meet the requirements of PSL-FPER-97001 by fire testing in accordance with the test requirements of ASTM E-119. This requirement was being added to provide additional assurance/validation of the new fire barrier design.

Revision 2 added the results from the fire test for the walls and ceiling of the Inverter Room, and removed the HOLD POINT from this engineering package. The test results showed the walls and ceiling of the Inverter Room met the requirements of ASTM E119-98, `Fire Tests of Building Construction and Materials', for a fire endurance rating of 3 hours. Based on the test results, the Inverter Room Fire Barrier is operable and compensatory measures of Administrative Procedure 1800022, 'Fire Protection Plan" are not required.

PLANT CHANGE/MODIFICATION 97040

REVISION 2

THERMO-LAG BARRIER UPDATES TO 'A' CABLE LOFT AND STAIRWELL ENCLOSURES

Summary:

Thermo-Lag is utilized as fire barriers between certain plant areas to meet Appendix R safe shutdown requirements. The Thermo-Lag walls were originally designed, tested and stipulated as 3-hour rated barriers meeting the requirements of ASTM-E-119. The Nuclear Regulatory Commission Generic Letter 92-08, Bulletin 92-01 and Bulletin 92-01, Supplement 1 provided notification that Thermo-Lag fire barrier materials failed to meet fire barrier ratings during retest. In FPL letter L-92-273, St. Lucie committed to compensatory measures (i.e., roving fire watch and verification of fire detector operability) until the Thermo-Lag issues are resolved.

Revision 0 of this PC/M provided an evaluation of the Thermo-Lag stairwell barriers as documented in evaluation PSL-FPER-98-002. It concluded that PSL Unit 1 Thermo-Lag walls do not meet 3-hour fire barrier temperature requirements, however it demonstrated that the barriers are adequate to withstand the fire hazard associated with their respective fire areas for a 3 hour duration. This evaluation covered the walls and ceilings for the stairwell enclosures separating Fire Areas "E" and "C" on the 19.5 elevation from Fire Area "O" on the -.5' elevation. The evaluation also documented the need to structurally enhance some of the barrier panels to preclude direct flame and hose stream impingement.

Revision 1 provided for the Thermo-Lag barrier upgrades to the 'A' Cable Loft Area east and west walls on the 28.67' elevation in the Reactor Auxiliary Building of Unit 1 at the St. Lucie facility. Revision 1 also provided for the installation of approximately 4'-0" of 1-1/2" conduit. This conduit will penetrate the west wall of the "A" Cable Loft Area, and it is provided for maintenance purposes to permit routing temporary cables into the room without requiring the door to be open which would then require a Fire Breach Permit.

Revision 2 provides for the Thermo-Lag barrier upgrades to the 'A' Cable Loft Area south wall from the 19.5' elevation to the 43' elevation and the southeast wall from 19.5' to the 43' elevation in the Reactor Auxiliary Building (RAB) of Unit 1 at the St. Lucie facility. Revision 2 also provides for the installation of approximately 8 linear feet of Thermo-Lag protection on cable tray C- 17. This revision completes the Thermo-Lag wall upgrade program for Unit 1.

Revision 2 also releases a "Hold Point" provided that the Thermo-Lag walls continue to be maintained in the Fire Breach Permit Log in accordance with AP 0010434, "Fire Protection Guidelines", and compensatory measures required by AP 1800022, "Fire Protection Program" continue in effect until directed by Engineering.

PLANT CHANGE/MODIFICATION 97041

REVISION 1

THERMO-LAG BARRIER UPDATES FOR CONDUITS, CONDUIT SUPPORTS, AND INTERFERENCES

Summary:

This PC/M provides the engineering and design information for the modifications required to upgrade Thermo-Lag fire barriers associated with conduits, conduit supports, and intervening or interfering commodities for Unit 1, and to address and document the required changes to the FSAR.

NRC provided notification that Thermo-Lag fire barrier materials had failed to meet their intended barrier ratings during fire endurance testing via Generic Letter 92-08, Bulletin 92-01, and Bulletin 92-01, Supplement 1. In FPL letter L-92-273, St. Lucie committed to compensatory measures (i.e., roving fire watch and verification of fire detector operability) until the Thermo-Lag fire barrier issues were resolved. The NRC was notified of the PSL schedule to update the Unit 1 Thermo-Lag deficiencies in FPL letter L-97-19, that was based on a meeting with the NRC on March 4, 1997.

This PC/M implements the required modifications to upgrade 294 feet of conduit that is presently protected by Thermo-Lag fire barriers, including their conduit supports, and intervening/ interfering commodities for PSL Unit 1. In addition, revision 0 of this PC/M provides for installing Thermo-Lag protection on 8 feet for conduit 11601B, in the Cable Spread Room, (one-hour rating) and 6 feet for conduit 11696, in the AWST in the northeast corner of the RAB on the -0.5' elevation, (three hour rating), including all conduit supports, and intervening or interfering commodities. These two conduits are being protected in lieu of upgrading existing Thermo-Lag on conduits 11002B-3" and 11552-4", respectively. All of the conduits to be upgraded are located in the Cable Spread Room, which has detection and a Halon Suppression System; therefore, one-hour fire barriers are acceptable. The installation of Thermo-Lag on conduit 11696 is required in Fire Area/Fire Zone 0/27 (AWST Area), which does not have detection and suppression, and therefore requires a three hour barrier. All Unit 1 conduits with Thermo-Lag fire barriers that require upgrading to one hour are identified in PC/M 97034. The addition of Thermo-Lag to conduit 11696 (three-hour barrier protection) is provided by revision 0 of this PC/M in lieu of upgrading Thermo-Lag protection on conduit 11552-4"(SA).

Revision 0 to this PC/M 97041 provided for the addition of Thermo-Lag 3-hour barrier protection to conduit 11696-4" ("IC" ICW Pump Power Circuit). Cable and Raceway Schedule, drawing

8770-B-328, was also modified to reflect actual fire zone routing for this cable.

Revision 1 to this PC/M provides the direction and authorization to protect three conduits in the Charging Pump Access Hallway, Fire Area/Zone N/36A and in the 1C Charging Pump Cubicle, Fire Area/Zone N/38, with Thermo-Lag to provide a one hour Appendix "R" fire barrier.

PLANT CHANGE/MODIFICATION 97061

REVISION 1

INSTALLATION OF FIRE DAMPER AS A 3-HOUR FIRE BARRIER AND ADDITION OF FIRE AREA 'D'

Summary:

In order to reduce the reliance on Thermo-Lag 330-1 fire barrier systems for protection of essential circuits in St. Lucie Unit 1, PC/M 97034 relocated the control cables and remote contactors for the Pressurizer Power Operated Relief Valves (PORV V1402 & V1404) to different fire areas or zones. By relocating the affected cables, Thermo-Lag protection is no longer required for control cables associated with these valves. However, PORV V1402 and V1404 are high-low pressure interface valves and must therefore be protected from spurious operation due to a fire. A circuit failure analysis performed by PC/M 97034 determined that separation of the 1B Switchgear Room and the 1B Electrical Penetration Room was required to eliminate the potential for spurious operation of V1404. This spurious operation could occur by postulating multiple electrical failures in the 1B Electrical Switchgear Room and in the 1B Electrical Penetration Room. Previously, both of these rooms formed part of a large fire area (Fire Area "C"). To eliminate the concern for spurious actuation of V1404, Revision 1 of PC/M 97061 created a new Fire Area/Fire Zone, D/78, bounded by the walls, floor and ceiling which previously defined Fire Area/Fire Zone, C/78. Now, with Fire Area/Fire Zone C/78 being changed to D/78, Fire Area "C" encompasses the following Fire Zones:

C-43	Control Building Personnel, El. 19.5'
C-56	1B Switchgear Room, El. 43'
C-44	Radio Chemistry Lab, El. 195
C-57B	Static Inverter Room, El. 43'
C-54	Laundry and Decontamination Area, El. 19.5'
C-58	1B Battery Room, El. 43'
C-55W	Main Hallway, El. 19.5', Cable Loft Area, El. 28.67'

Revision 0 of this PC/M installed a new fire damper in the west wall of this room as part of qualifying that wall as a 3-hour fire barrier. Revision 0 also added conduit and wiring for the damper limit switch to the local alarm panel for zone 12B, which is located in the same room, as well as analyzing the fire barriers to assure they would qualify as three-hour Appendix R barriers. The boundaries (walls, ceiling, floor) of the new fire area that were analyzed in Revision 0 will now be used to create the new Fire Area D in Revision 1. Fire hazards associated with this new fire area are being analyzed to determine if a fire originating in this fire area will be confined to this new fire area. PC/M 97061, Revision 1, also reanalyzes and revises, as

1B Electrical Penetration Room as a separate fire area.

A HOLD POINT was included in Revision 0, requiring Engineering to review the new fire damper's seismic analysis report prior to installation of the fire damper. Engineering reviewed and approved the new fire damper's seismic analysis report and the HOLD POINT was released.

Two HOLD POINTS were included in Revision 1 of this PC/M. Prior to the completion of this package that makes the 1B Electrical Penetration Room its own Fire Area 'D', the following tasks must be completed.

1.) PC/M 96081, Revision 3, must complete the restoration of the fire barrier seals at the top of concrete block walls 123, 124 and 125 reported in CR 98-1161. This task assures these walls are acceptable Appendix R three hour fire barriers.

2.) In addition to the two pipe penetrations being repaired, the south wall must be completed to repair many small defects found during a walk down and reported in CR 98-1283. This work will return these fire stops to their original condition as acceptable three-hour fire stops and the south wall will be a three-hour fire barrier.

PLANT CHANGE/MODIFICATION 98020

REVISION 0

ABANDONMENT OF DRUMMING STATION PROCESS RADIATION MONITORING EQUIPMENT

Summary:

The purpose of this PC/M is to abandon the Drumming Station Process Radiation Monitoring equipment including:

- Drumming station process radiation monitoring equipment, drumming station air sampling monitor, isokinetic nozzle and process tubing.
- Instruments RE-26-48, RE-26-50, RIS-26-48 and RIS-26-50.

The drumming station process radiation monitoring system was designed to measure particulate and gas radioactivity of the exhaust air (effluent) exiting the drumming area. A representative sample was pumped from an isokinetic nozzle mounted in the drumming area exhaust duct and was routed from the drumming area to the drumming station air monitor. Particulate and gas radioactivity levels were monitored at the drumming station air monitor and the signals were retransmitted radiation monitor cabinet A located in the main control room.

The drumming station process radiation monitoring equipment is no longer required. Waste solidification operations are no longer performed as originally stated in the UFSAR. The drumming station area is now considered a normal RAB area. Plant stack effluent monitors and area radiation monitoring in the RAB along with HP Surveys of this area provide adequate coverage for this area.

PLANT CHANGE/MODIFICATION 98030

REVISION 0

CHARGING PUMP 1A CONDUIT REROUTES AND ICW INTAKE E-LIGHTS

Summary:

During the 1997 - 1998 Engineering effort to reverify and reissue the Safe Shutdown Analysis (SSA) for St. Lucie Unit 1, it was determined that the relocation of one 1A Charging Pump cable and protection of the power cable for the 1C Intake Cooling Water (ICW) Pump would eliminate the addition or upgrade of approximately 150 feet of Thermo-Lag fire barriers. This package provides the design and justification to implement the options determined from the engineering review.

The control cable for PS-2224X, Charging Pump 1A suction pressure, is currently routed in conduits along the east side of the ceiling of the charging pump area common access hallway. Cable failure can disable Charging Pump 1A. To ensure the availability of at least one charging pump during fires in this area, Fire Area N - Fire Zone 36A, Thermo-Lag fire barriers rated for at least 1-hour protection would have to be installed on affected conduits and a raceway pullbox. By relocating the cable inside of the 1A Charging Pump cubicle (routed through an embedded conduit), the need for fire barriers for this cable is eliminated.

This PCM also adds 8-hour emergency battery backed lights (E-lights), as required by 10CFR50, Appendix R Section III.J, at the ICW intake structure for manual actions associated with using the 1C ICW pump in place of the 1A ICW pump.

PLANT CHANGE/MODIFICATION 98043

REVISION 0

ULTRAFINE MEDIA FOR CVCS PURIFICATION FILTERS 1A AND 1B

Summary:

This PC/M allows the use of various size micro-filter media in one or both Unit 1 CVCS Purification Filters 1A and 1B, allows the 1A Purification Filter to be normally bypassed, eliminates use of orifices within the 1B CVCS Purification filter and allows an increase in purification flow while on shutdown cooling.

The use of smaller particulate rated (higher efficiency) filter elements in the CVCS Purification Filters will have no adverse impact on plant safety or operation. The Engineering Package includes the engineering and design necessary to provide justification for modification, document changes in the FSAR and provide administrative controls for the various grades of filter media and to note the 1A CVCS Purification Filter is typically placed in bypass as operational experience indicates there are waste disposal benefits for initial particulate removal by the Purification Ion Exchangers.

Increasing purification flow while on Shutdown Cooling provides a more rapid decrease in RCS non-soluble particulate inventory. Increasing the flow from 132 gpm to 140 gpm will allow the reduction of radiolytic particulates during the initial stages of a refueling outage for a reduction in accumulated dose to station maintenance and operating personnel.

PLANT CHANGE/MODIFICATION 98044

REVISION 0

APPENDIX R SAFE SHUTDOWN ANALYSIS

Summary:

Condition Report 97-2288 identified the need to perform a validation effort of the St. Lucie Unit 1 and 2 Appendix R Safe Shutdown Analysis (SSA). This need was identified as the original Unit 1 SSA included a limited discussion of assumptions and overall methodology. In addition, a lack of detail existed in the disposition of cable failures in affected fire areas and made the translation of the SSA identified manual actions into operating procedures difficult.

This PC/M includes the results of the validated SSA essential cable dispositions for each Unit 1 Fire Area/Fire Zone. The validated SSA along with a description of the methodology and assumptions has been documented in Engineering Evaluation PSL-ENG-SEMS-98-035. As documented in that evaluation, an EP will be utilized to formally issue the SSA (drawing 8770-B-048) for as-building and will include the appropriate 10 CFR 50.59 evaluation to support changes to the associated plant operating procedures.

In addition to the SSA essential cable dispositions, this EP includes the necessary revisions to the Unit 1 Appendix R Essential Equipment List (EEL - drawing 8770-B-049) and the necessary revisions to the on-line cable and raceway system (CARS) database that resulted from the SSA validation effort.

PLANT CHANGE/MODIFICATION 98046

REVISION 1

TELEPHONE SYSTEM UPGRADE

Summary:

The changes involved in this modification include: 1) Installation of a new NORTEL Meridian PBX telephone system, Option 81C, in the SSB telecommunication room, to replace the existing System 85 AT&T PBX telephone system equipment located in the existing NSB telecommunication room. 2) Replacement of the existing System 85 AT&T remote module in the D-13 telecommunication room with a NORTEL Meridian remote module, to be located in the new NSB telecommunication room. 3) Installation of a new 120/208V PP-154 in the new NSB telecommunication room to provide power to the plant telephone system equipment through a series power line conditioner. 4) Rerouting the power supply cables and conduit (Cable 15006B) from existing 480V PP-135, circuit 5, in the North Security Building, to new 120/208V PP-154 in the new NSB telecommunication room. Existing PP-154 in the existing NSB telecommunication room will be removed as part of this modification. The air conditioning loads on existing PP-154 are being rendered obsolete by the installation of the new NSB chiller unit, which is being installed under a separate contract. 5) Providing power from existing 120/208V PP-153 in the existing NSB telecommunication room to the backup air conditioner for the new NSB telecommunication room. The backup A/C unit will be located on the south side of the NSB and will be automatically started on a high temperature in the new NSB telecommunication room. 6) Removal of the normally energized contactor from 480 volt PP-135.

The plant telephone system upgrade is being implemented in order to standardize telephone system equipment throughout the FPL system and for reasons of Year 2000 compatibility.

This EP was revised to incorporate FRG comments regarding safety classification, configuration, and EDG loading.

PLANT CHANGE/MODIFICATION 98088

REVISION 0

EVALUATION OF AFW PUMPS DISCHARGE PIPING FOR INCREASED PRESSURE AND TEMPERATURE

Summary:

Condition Report 96-2972 was generated to document the actual pressure being higher than the design pressure in the Auxiliary Feedwater (AFW) Pumps 1A and 1B discharge piping to the main feedwater header connections lines 20"-BF-14 and 20"-BF-19 and in the recirculation piping to the isolation valve (V09104) prior to the Condensate Storage Tank (CST). CR 96-2063 was generated to document the operating temperature being higher than the design temperature in all three AFV Pumps 1A, 1B and 1C discharge piping to the main feedwater header connections lines 20"-BF-14 and 20"-BF-19. This PC/M provides the evaluation for the higher pressure and temperature in the subject piping system and the design details for pipe support modifications necessary to satisfy the higher loads associated with the increased temperature.

PLANT CHANGE/MODIFICATION 99005

REVISION 0

INTAKE SCREENING SYSTEM UPGRADE

Summary:

The Engineering Package includes the engineering and design necessary to provide justification for replacement of the Unit 1 Intake Screening System (Traveling Water and Stationary Screens) with upgraded corrosion resistant material.

The replacement Unit 1 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 were replaced with stainless steel valves to afford better seawater corrosion resistance.

Additionally, the exposed sections of the Unit 1 trash pipe were removed and the remaining buried sections capped. Degraded concrete saddle type supports used to support the trash pipe were removed, as required.

The screen rotation indication system was updated from the current chain driven switching mechanism to a proximity switch mechanism similar to that on Unit 2. This involves the removal of four time delay relays from RTGB 102 and installation of four local time delay circuits at the traveling screen bays. Addition of a non-safety 120 VAC power source was required for these relay circuits. The proximity switch installation required the fabrication of stainless steel support hardware and additional PVC conduit and non-metallic junction boxes.

PLANT CHANGE/MODIFICATION 99010

REVISION 2

APPENDIX 'R' CABLE REROUTES

Summary:

As a part of the preparation for the 1998 NRC FPFI (Fire Protection Functional Inspection) audit, a revalidation of the existing SSA (Safe Shutdown Analysis) was initiated. The effort resulted in identification of a number of shortcomings of the original SSA or items where the original assumptions were no longer applicable in today's licensing environment. Specifically, the original FPL position that multiple hot shorts (resulting in a spurious energization of equipment) are not a credible cable failure mode for not "high-low pressure" interface equipment has been rejected in recent NRC practice. The reanalysis effort identified a number of situations where multiple cable hot shorts could have resulted in an inadvertent positioning of valves which could lead to a loss of RCS inventory beyond make-up capability of available equipment, therefore, resulting in a fire induced LOCA. Further, assumptions concerning the need for containment fan coolers to allow in containment manual actions required to establish shutdown cooling were found to be not acceptable. This necessitates rerouting of the fan cables to assure that conditions in containment are acceptable for entry. A number of other items overlooked in the original SSA were also addressed in this EP.

This Engineering Package provides the design details for:

a. rerouting of cables for the reactor head gas vent system RHGVS) valves V1445, V1446 and V1449 (cables 11256C, 11255C and 11256D) in the Reactor Auxiliary Building (RAB). These new cable routes, in dedicated conduits, will assure strict compliance with requirements of Appendix R by eliminating the potential for fire induced hot shorts in RAB. Isolation switches were also provided to assure removal of power to the valves for an alternate shutdown scenario.

b. rerouting of containment fan coolers 1HVS-1C and -1D power circuits (cables 10309A and 10310A) out of the Cable Spreading Room. This will assure availability of the containment fan coolers, required to allow containment entry necessary to manually align shutdown cooling valves, for an alternate shutdown scenario.

b. HSCP room cooling fans circuit (cable 10517E) were rerouted out of the cable spreading room, which will assure it's availability for the alternate shutdown.

d. PORVs cables in electrical penetration rooms were rerouted in dedicated conduits from the isolation switches to their respective penetrations to minimize exposure to any other energized cables within penetration wireways.

e. PT-08-1B cable (10603Q) which provides signal for Steam Generator 1B pressure indication on the HSCP, was rerouted out of cable spreading room.

f. Isolation fuses were provided for five(5) circuits originating from 125 V dc PP-119 in Fire Area C. These fuses will assure that no fire created faults in other fire areas can result in a loss of the PP-119.

Revision 0 of the EP provided electrical and civil design for item a.) - conduits 11256F and 15022L on RAB el. 43' and item e.) - conduit 15022E on RAB el. 19.5'.

Revision 1 (general revision) incorporated editorial comments requested by FRG.

Revision 2 provided all the design details to complete the modifications outlined above. Revision 2 is also incorporating SSA/CARS changes due to disposition of CRs 98-1609 and 98-1609 Supplement 1.

PLANT CHANGE/MODIFICATION 99011

REVISION 0

CABLE SPREAD ROOM THERMO-LAG WALL CIRCUIT MODS

Summary:

During the 1998 NRC Fire Protection Function Inspection (FPFI) of Unit 1, a concern was expressed about the qualification of the Thermo-Lag wall separating the Cable Spread Room and the 'B' Switchgear Room on elevation 43.0' at column RA3 between columns RAI and RAJ. Engineering concluded that the appropriate course of action was to replace the Thermo-Lag wall with an alternate qualified wall. This work is to be performed through PCM 99029.

The wall separates two rooms on elevation 43.0' that represent two different Appendix R fire areas. Unit 1 Cable Spread Room is Fire Area/Fire Zone B/57 and shutdown train 'B'. The 'B' Switchgear Room is Fire Area/Fire Zone C/56 and shutdown train 'A'. In addition to separating the two rooms, the Thermo-Lag wall is extended into the Cable Spread Room to enclose the cable tray riser area at column RAI-RA3. This section of wall around the cable tray riser will be deleted when the separating wall is replaced by PCM 99029.

Cables essential for safe shutdown of 'B' train are routed through the cable tray risers at column RA3 and RAI and protected by the existing wall enclosure. To maintain required safe shutdown functions, essential cables currently routed through the cable tray risers are being rerouted on elevation 19.50' of the RAB to floor penetrations directly in the 'B' Switchgear room by this PCM. The affected cables are as follows:

Cable 10059E - RY-26-8OB1 in the 'B' Electrical Penetration Room to RY-26-8OB2,
10098A - PT-1100Y (Penetration #B5) to Isolation Panel 1B,
10120C - V-1405 limit switch contacts (Penetration D1) to MCC 1B5, and
16054 - 125-volt dc power from Bus 1B to Bus 119

New floor penetrations were added near 125-volt dc Bus 119 for cable 16054 and near Isolation Panel 1B for cable 10098A. Installation of conduits, supports, floor penetrations and new cables may be performed in any mode of plant operation with prior approval. Disconnecting existing cables and re-terminating new cables requires a unit outage due to the sensitive nature of the components and functions provided by the affected cables.

PLANT CHANGE/MODIFICATION 99012

REVISION 1

THERMO-LAG REMOVAL FROM FIRE DAMPER ASSEMBLY 25-123 STANDOFFS

Summary:

This PC/M provides details for removal of Thermo-Lag from the existing Fire Damper Assembly 25-123 standoff. Since Thermo-Lag has been shown to be a combustible in Generic Letter 92-82, PSL is reducing the use of Thermo-Lag from as many applications as possible. Fire dampers are allowed to pass heat through their louvers during a fire but the heat can not cause the ignition of a combustible or damage to safe shutdown components on the cold side of the fire damper. Since the Thermo-Lag could ignite due to the heat transferred to it by the fire, its presence in the vicinity of the fire damper voids the fire damper's qualifications.

One function of a fire damper is to assure that when a fire exists on one side of the damper it will not spread to the other side of the damper. Since the fire damper is constructed of metal the heat from the fire will certainly pass through the fire damper curtain and the standoff to the metal ducting on the other side of the damper. This is acceptable providing there are no combustibles to ignite and no safe shutdown components to be damaged in the area where the heat will be concentrated. When Thermo-Lag was shown to be a combustible in Generic Letter 92-82, plans were formulated to remove all Thermo-Lag from fire damper assemblies. Fire Damper Assembly 25-121, which had a Thermo-Lag standoff, was recently modified via PC/M 98091 to remove the Thermo-Lag and replace it with a sheet metal standoff. This PC/M removed the Thermo-Lag from the 14 gauge sheet metal standoff in Fire Damper Assembly 25-123, which is the last fire damper in Unit 1 that employed the use of Thermo-Lag as part of its standoff material. The 14-gauge sheet metal was left in place to act as the standoffs.

Revision 1 requires that the Battery Room exhaust fans operate at all times and the supply fans are started as soon as possible after the supply duct has been put back in service.

PLANT CHANGE/MODIFICATION 99014

REVISION 0

RCP MECHANICAL SEAL REPLACEMENT WITH N-9000 SEAL

Summary:

This Engineering Package includes the design information necessary for replacement of the existing Unit 1 Reactor Coolant Pump SU mechanical seals with N-9000 seals. The seal modification has already been implemented on Unit 2.

Replacing the SU seals with N-9000 seals will increase service life and provide reliable seal operation 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, Maine Yankee) and have a projected life of 50,000 hours (5.7 years). The replacement N-9000 seals have been designed and rigorously tested to demonstrate their ability to meet station blackout (SBO) coping time with low leakage rates.

10CFR50, 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; the normal path is to the VCT via V2198 and the backup path is to the Quench Tank via V2507. Pneumatic valve V2507 was changed from normally locked open, fail open to normally closed, fail closed. RTGB switch HS-2507 was changed to provide open contacts in each power/control leg to prevent spurious operation during a fire. An emergency light was installed in the mechanical penetration room to allow manual closing of valve V2198. This will ensure isolation of the CBO flow to mitigate the postulated failure of the automatic containment isolation valves to close due to an Appendix R fire.

Due to the design of the N-9000 seals and the expected CBO flows resulting from a postulated single failed seal stage, the high CBO flow alarm set-point was lowered to ensure that this condition will be readily apparent to the operators.

PLANT CHANGE/MODIFICATION 99018

REVISION 0

INSTALLATION OF BYPASS LINE ACROSS THE RCS SIDE VALVE SEAT FOR V3651, V3652 & V3481

Summary:

This engineering package provides for the modification of the St. Lucie Unit 1 Shutdown Cooling (SDC) isolation valves V3651, V3652 and V3481. The SDC valves are located inside containment on the hot leg suction lines to the 1A and 1B Low Pressure Safety Injection (LPSI) pumps. The modification consists of installing a 1/2" bypass line across the Reactor Coolant System (RCS) side valve seat. The bypass line for V3651 and V3652 will contain two 1/2" gate valves to allow for line installation with water pressure in the SDC valves. The bypass line for V3481 will not contain manual isolation valves since the bypass line will be installed with the valve depressurized and drained. A process called Electrical Discharge Machining (EDM) will be used to bore the holes in the SDC valves to minimize entry of foreign materials.

This modification will increase motor operator design margin during a postulated pressure locking scenario and satisfy the requirements of the NRC Generic Letter 95-07 as identified in Engineering Evaluation PSL-ENG-SEMS-99-022. This evaluation also determined that the remaining SDC suction valve, V3480, did not need modifying due to differences in the valve wedge design.

Installing the bypass line across the upstream or Reactor Coolant System (RCS) side of the valve disk allows venting of high pressure fluid from the bonnet, and thereby prevents the potential of a pressure locking condition. Pressure locking is a phenomena that can occur in flexible wedge gate valves when fluid becomes pressurized within the valve bonnet and the actuator is not capable of overcoming the additional thrust required. This modification is endorsed by NUREG-1275 Vol. 9, and has been performed within the nuclear industry to address the pressure locking issue (i.e., Calvert Cliffs).

PLANT CHANGE/MODIFICATION 99025

REVISION 0

AFW CHEMICAL INJECTION CLASS BOUNDARY CHANGE

Summary:

PC/M 99025 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 PC/M 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. These CRs identified that the AFW System was in an operable, but degraded condition, with compensatory actions required.

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.

The Engineering Package (EP) includes the engineering and design necessary to provide justification for adding four class boundary check valves to the Auxiliary Feedwater System.

Additionally, two 3/8" valves were identified in the NNS Chemical Injection System, upstream of the new check valves, which have not been incorporated into design drawings, and were added to design drawings by this PC/M.

PLANT CHANGE/MODIFICATION 99029

REVISION 0

CABLE S CABLE SPREAD ROOM TO "B" SWITCHGEAR ROOM THERMO-LAG WALL REPLACEMENT

Summary:

This Engineering Package replaced the Thermo-Lag wall between the St. Lucie Unit 1 Cable Spread Room (Fire Area/Zone B/57) and the "B" Switchgear Room (Fire Area/Zone C/56).

The NRC provided notification that Thermo-Lag fire barrier materials had failed to meet its intended barrier ratings during fire endurance testing via Generic Letter 92-08, Bulletin 92-01 and Bulletin 92-01, Supplement 1. In FPL letter L-92-273, St. Lucie committed to compensatory measures (i.e., roving fire watch and verification of fire detector operability) until the Thermo-Lag fire barrier issues were resolved.

PSL utilizes Thermo-Lag 330-1 fire walls as fire barriers between certain adjacent fire areas. These walls are constructed out of steel frames with Thermo-Lag 330-1 panels bolted to the frames. The Thermo-Lag fire walls were originally designed as three-hour rated fire barriers (as defined) or meeting the requirements of ASTM E-119. As a fire barrier, Thermo-Lag walls are required to: 1) minimize the temperature increase on the unexposed side of the barrier; 2) prevent the passage of flames through the unexposed side; and 3) prevent projection of water beyond the unexposed side due to a hose stream test following constant exposure to a fire for a duration of three hours.

In lieu of upgrading the existing Thermo-Lag wall panels, a FPL fire protection evaluation provides the alternative of removing the Thermo-Lag barriers and replacing them with a sheet metal and ceramic fiber barrier. Based on experience from the PSL Unit 1, Inverter Room modifications, the alternative barrier configuration was chosen. This configuration is supported by test documentation as a 3-hour fire rated assembly.

This EP provides the required modifications to implement the alternative fire barrier consisting of sheet metal and ceramic fiber and documents the basis for the acceptability of the sheet metal and ceramic fiber barrier as a 3-hour fire rating in compliance with ASTM E-119.

In addition to the replacement of the fire barrier between the cable spread room (CSR) and the "B" Switchgear room, this PC/M permanently removed the Thermo-Lag from the structural steel framing around the cable riser in the southeast corner of the CSR.

PLANT CHANGE/MODIFICATION 99100

REVISION 2

PRESSURIZER INSTRUMENT NOZZLE REPLACEMENT

Summary:

This Engineering Package provides for the modification of St. Lucie Unit 1 pressurizer instrument nozzles (steam space). The nozzles are designated on pressurizer drawing 8770-17 as two 1" pressure tap nozzles (top), and two 1" level nozzles (top). The nozzles serve RCS and pressurizer level instruments (LT-1103, LT-1110X, LT-1110Y, LT-1117, LT-1117-1 and the tygon tubing), and pressure transmitters (PT-1100X, PT-1100Y, PT-1102A, PT-1102B, PT-1102C, PT-1102D, PT-1103, PT-1104, PT-1107, PT-1108). The instrument nozzles are made of a heat of material that is known to be susceptible to primary water stress corrosion cracking (PWSCC). The purpose of these replacements is to minimize the possibility of a future nozzle failure. To replace the nozzles, a modification of the original partial penetration weld/nozzle design will be required to reduce the susceptibility to PWSCC, to reduce residual stress and for ALARA consideration. The modification moves the partial penetration weld joint to the pressurizer outside surface where previously the joint was on the inside surface. The material of the nozzles will be changed from alloy 600 to alloy 690. Exposure to personnel during installation and welding of the nozzles were reduced by both the modified design and by automation of the welding processes. The replacement nozzle design employed in this PCM has previously been installed on Unit 2 for the pressurizer liquid space instrument nozzles and RCS hot leg instrument nozzles.

The pressurizer instrument nozzles provide taps for sensor inputs to monitor the level and pressure of reactor coolant in the pressurizer. Replacement of the pressurizer instrument nozzles requires the temporary removal of the associated condensate pots and connected piping during implementation to provide access to the nozzles. The condensate pots and instrument piping were reinstalled at the same location per the original design requirements. The affected level instruments are required for safe shutdown, post-accident monitoring and the Pressurizer Level Control. The affected pressure instruments are required for RPS, ESFAS, DSS, safe shutdown, SDC interlocks, post-accident monitoring, the Reactor Coolant Pressure Control System and the Overpressure Mitigation System. Several of the level and pressure instruments are also listed in the Essential Equipment. Level instruments (LT-1117, LT-1117-1 and the tygon tube level indicator), which are used to monitor RCS level during refueling will have alternate reference lines temporarily installed to allow replacement of the associated pressurizer nozzle without affecting their operation.

The location of the pressurizer spray line upstream of valve V1253 (at centerline elevation 73') creates an interference for the work on Nozzles B and C. A portion of spool 1-3"-RC-109G (ASME Class 1, Seismic Class 1), associated with the pressurizer spray line, was removed and either reinstalled or replaced as detailed on drawing ENG-99100-018. In addition, associated seismic restraint SPS-617 and Pressurizer Tailpipe restraint RC-005-12B will also be removed and reinstalled as delineated in the Implementation Instructions/Specifications.

Revision 1 incorporated an evaluation of the requirements of Technical Specifications 3.4.2 during implementation of this PC/M, and added a separate temporary instrument reference line for the tygon tubing to increase its independence from the installed LT-1117 and LT-1117-1.

Revision 2 clarifies the evaluation of the requirements of Technical Specification 3.4.2 during implementation of this PC/M.

PLANT CHANGE/MODIFICATION 99101

REVISION 0

<u>TRIP MAIN FEEDWATER PUMP & HEATER DRAIN PUMPS ON MSIS</u>

Summary:

The purpose of this PC/M is to add an MSIS trip signal to the Main Feedwater Pumps and Heater Drain Pumps control circuits for rapid feedwater isolation. The design relies on the availability of the spare relay contacts and the spare terminal block terminals in the ESFAS cabinets shown on design documents. All other aspects of the design have been verified by walk-down. The ESFAS relays are located on a "drop down" cabinet door, and accessing this area is considered a high risk evolution during power operation of the unit.

This Engineering Package (EP) provides the engineering necessary to justify the modifications and document the changes. The PC/M revises the FSAR and allows for rapid Main Feedwater isolation on receipt of a MSIS signal.

SECTION 2

SAFETY EVALUATIONS

SAFETY EVALUATION SEMS-90-052
REVISION 1

GENERIC USE OF SEALANT INJECTION

Summary:

The purpose of this evaluation is to evaluate temporary repairs of gasket and packing leaks on safety related and quality related systems (i.e., sensitive systems such as feedwater) or components through the use of sealant injection. The affected components shall be replaced or permanently repaired in accordance with the time constraints delineated in the CR disposition or other approved Engineering output. The use of this method to repair the subject components will have no impact on plant safety or operation.

Revision 1 updated the evaluation to reflect changes in vendors, sealant materials, and to address recent regulatory issues (i.e., IN 97-74, SER 5-97, and Part 9900) pertinent to leak sealing.

SAFETY EVALUATION SEMP-93-033
REVISION 7

STEAM GENERATOR REPLACEMENT REPORT

Summary:

This safety evaluation addresses the physical implementation activities (rigging and handling, heavy loads, etc.), of temporary and permanent changes to equipment and components conducted before, during, and following the steam generator replacement outage (SGRO). In addition, this evaluation considers subsequent plant operation with various permanent plant changes to be implemented as a result of the SGRP.
The following SGRP activities are evaluated in this safety evaluation:

- Heavy load evaluations for the rigging/handling of, and transport/haul routes for the steam generators
- Modifications of pipe rupture restraints from their as-designed configurations
- Modifications of the steam generator manway platforms
- Modifications to the RSG insulation
- Modifications of the RSG blowdown piping
- Erection and utilization of the OSG Interim Storage Facility
- Modifications to the steam generator supports
- Steam Generator replacement
- Contingency replacement of cold leg elbows
- Reactor Containment Shield Building (RCSB) concrete construction hatch
- Containment preparations
- 1A Main Stream Trestle Structural Modifications
- Containment Vessel Construction Hatch
- Containment Tool Room removal

The first item covers various rigging/handling and transport activities evaluated under 10 CFR 50.59 because of their potential effect on both Unit 1 and Unit 2 during the SGRP. The remaining items, which change the plant configuration from that existing prior to the SGRO, are either permanent plant modifications or activities which were not screened for 10 CFR 50.59 applicability, and therefore are evaluated under 10 CFR 50.59.

SAFETY EVALUATION SEMP-94-083
REVISION 1

REDUCED PRESSURIZER HEATER CAPACITY

Summary:

The purpose of this safety evaluation is to permit operation with a reduced pressurizer heater capacity of 1375 kW. This would permit up to 10 pressurizer heaters, a total of 125 kW, to be removed from service, as described below:

2 proportional pressurizer heaters (25 kw), and;
8 backup pressurizer heaters (100 kw), with a maximum of 2 heaters in any one backup heater bank.

Each time a pressurizer heater is retired (i.e. heater leads de-terminated), a Minor Engineering Package (MEP or MEP revision) will be issued to as-build the change on the affected documents (control wiring diagram, TEDB, and power distribution data sheets).

The pressurizer heaters have a total installed capacity of 1500 kw, 300 kw of which are on two proportional heater banks (P-1 and P-2), and 1200 kw are on six backup heater hanks (B-1 to B-6). Because of the design margin of the pressurizer heaters, safe plant shutdown and the results of postulated events in the FSAR safety analyses are not adversely affected by operation with reduced heater capacity.

Revision 1 removed the requirement to revise Table 1 of this safety evaluation due to changes in the availability of pressurizer heaters. The maximum number of heaters which may be out of service was unchanged. In addition, the SE was updated following replacement of the Unit 1 heaters per PC/M 97009. The changes evaluated in this revision do not invalidate the safety analysis conclusions.

SAFETY EVALUATION SECP-96-059
REVISION 3

<u>ST. LUCIE COUNTY - S. HUTCHINSON ISLAND WASTE WATER TREATMENT FACILITY AND WASTE WATER COLLECTION/RECLAIMED WATER DISTRIBUTION SYSTEMS</u>

Summary:

To provide treatment for the increasing quantity of domestic wastewater generated on Hutchinson Island, St. Lucie County has constructed a new Wastewater Treatment Plant (WWTP) on Hutchinson Island. This facility is located approximately 2 miles south of the St. Lucie Nuclear Power Plant. This facility will serve developments from the Martin County line north to the power plant. Reclaimed water from the WWTP will be used for irrigation of properties on Hutchinson Island. During periods of high flow and/or rainy weather, excess reclaimed water will be discharged through an outfall to the St. Lucie Plant discharge canal.

Three piping lines run from the WWTP to the St. Lucie Plant - an 8-inch force main for wastewater collection, an 8-inch reclaimed water line and an 18-inch reclaimed water outfall line. The pipes are installed under the intake and discharge canals via subaqueous crossings, utilizing directional boring.

Potentially hazardous materials are used and stored at the WWTP.

This evaluation addresses the activities related to the construction of the wastewater collection/reclaimed water systems to be performed on FPL property (subaqueous crossing of pipes under the intake and discharge canals, and construction of the reclaimed water outfall line along the discharge canal dikes). This evaluation also addresses the discharge of excess reclaimed water at an outfall point at the St. Lucie discharge canal (including consideration of PSL plant operations, quality of discharged water, and environmental concerns). Consideration of the storage of hazardous materials at the WWTP, and transport of these materials to the WWTP, are also addressed in this evaluation.

The construction operations addressed in this evaluation are limited to areas on the east side of State Road A1A, and subterranean crossings of A1A for routing of tie-ins. These areas are in the FPL Owner Controlled Area, but are not included in the Plant Protected Area. The use and storage of potentially hazardous materials at the WWTP, and the delivery of these materials to the WWTP have been evaluated for control room habitability.

Revision 3 of this evaluation documents the acceptability of the abandoned directional bore hole beneath the intake canal, addresses the use and storage of potentially hazardous materials

at the WWTP, the delivery of these materials to the WWTP, follow-up issues related to the operation of the WWTP, and includes FSAR changes related to the location of the reclaimed excess reclaimed water outfall line.

SAFETY EVALUATION SECS-97-014
REVISION 2

SPECIFICATION SPEC-C-035, INSTALLATION OF TYGON TUBING FOR VENTING AND DRAINING INSERVICE 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 Implementor cannot install a vent or drain configuration in accordance with the provisions of the specification, or should the Implementor require a change to any provisions of the specification, a "Request for Specification Clarification or Change" 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 & 2.

Revision 1 addresses effects on this safety evaluation due to changes made to specification SPEC-C-035, Rev. 1 to incorporate permanent changes requested per specification clarification sheet request numbers 297-002, C97-003, C97-008, and C98-023 and removes the limitation for double isolation valves.

Revision 2 incorporates Rev. 2 of Specification SPEC-C-035, which was revised (a) to clarify the recommendations for reuse of CPVC and PVC threaded couplings and king fittings and (b) to permit tygon tubing to be used for control of leaks at vent or drain valves for in-service systems.

SAFETY EVALUATION SEES-97-015
REVISION 0

UFSAR REVISION DUE TO REMOVAL OF FIRE BARRIERS FOR THE NUCLEAR INSTRUMENTATION AMPLIFIERS

Summary:

This safety evaluation provides the basis for an UFSAR update required due to the removal of the barriers for the Nuclear Instrumentation Amplifiers (wide range detectors) located in the nuclear instrumentation tunnel. These "fire" barriers are specifically mentioned in the UFSAR even though there are neither Appendix A nor Appendix R (sections 2.4 and 2.5 of UFSAR Chapter 9.5A) requirements for such. The original purpose for these "barriers" was to provide shielding for the Nuclear Instrumentation amplifiers from the heat generated by the RCP motors cabling located in the Nuclear Instrumentation tunnel.

SAFETY EVALUATION 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 were clarified via this evaluation.

This evaluation discusses use of equipment in the SGBTF that is no longer needed since other available equipment perform similar functions. Therefore, no adverse affect is rendered on the system

SAFETY EVALUATION SEMS-97-066
REVISION 1

10CFR50.59 SAFETY EVALUATION FOR ADDITION OF HYDROGEN PEROXIDE TO THE RCS DURING SDC

Summary:

This evaluation addresses the addition of hydrogen peroxide to the Reactor Coolant System during Shutdown Cooling (SDC) to oxygenate the reactor coolant and facilitate crud burst/removal at PSL Units 1 and 2. To comply with chemistry limits for the RCS in the Technical Specifications and FSAR, the RCS temperature (Tave) shall be less or equal to 200F and the RCS hydrogen concentration shall be less than 5 cc/kg prior to addition of hydrogen peroxide.

This evaluation demonstrates that addition of hydrogen peroxide to the RCS during SDC will not be detrimental to plant systems.

Revision 1 of this Safety Evaluation changed the requirement for the RCPs to be in-service during hydrogen peroxide additions, to optional. Technical Specification 3.1.1.3 can be met by running at least one RCP or one LPSI pump. Securing the RCPs prior to hydrogen peroxide addition reduces the volume of reactor coolant to be treated and hence, has the potential to reduce clean-up time, thus shortening this outage window. The effects resulting from this practice should be evaluated and shutdown chemistry operations modified accordingly for future shutdowns.

SAFETY EVALUATION SEMS-97-092
REVISION 1

UFSAR COMBUSTIBLE LOADING UPDATE FOR UNIT 1

Summary:

This safety evaluation addresses the combustible loading increase in various fire zones due to Thermo-Lag barrier material and other miscellaneous materials. The evaluation was required because according to NRC Information Notice 92-82, Thermo-Lag is now considered a combustible material, and also, over the years the Operations and Maintenance departments have requested the storage of combustible material in these fire zones.

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

Revision 1 to this safety evaluation has been prepared to update the combustible loadings in Fire Area GG, Fire Zone 12 due to the addition of a modular water treatment plant.

SAFETY EVALUATION SEMS-97-102
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR ON-LINE MAIN STEAM SAFETY VALVE SETPOINT VERIFICATION TESTING

Summary:

Periodic Inservice Testing of the Main Steam Safety Valves (MSSV's) for lift setpoint verification is required by plant Technical Specification (TS) 3.7.1.1. This evaluation addresses the acceptability of performing MSSV setpoint verification testing while in Modes 1-3. On-line testing of the MSSV's is expected to save 8-24 hours of outage critical path time.

The proposed on-line testing requires Safety Valve Test (SVT) equipment to measure the hydraulically assisted stem load required to overcome the closing spring force and initiate valve opening. MSSV's will be tested (i.e., lifted) one valve at a time. The test equipment consists of several components (load frame, accelerometer, load cell, transducers, cables, hydraulic pump, controls and recording devices). The SVT software detects lift and terminates the lift by eliminating the additional lifting force provided by the hydraulic lift cylinder. As the energy release during the test is negligible, perturbations in secondary and primary system parameters will be small and bounded by normal operating transients.

During the proposed test, the SVT equipment will not prevent the MSSV from operating to provide overpressure protection in accordance with ASME B&PV Code, Section III requirements. Accordingly, the MSSV under test is considered OPERABLE per TS requirements.

SAFETY EVALUATION SEMS-97-104
REVISION 1

CONDENSATE STORAGE TANK VOLUME DESIGN BASIS REVIEW

Summary:

As discussed in an NRC letter from Robert M. Gallo to T. F. Plunkett dated March 25, 1997, the NRC performed a design inspection of the St. Lucie Unit 1 Auxiliary Feedwater System. During preparation for that inspection, it was identified that the calculations supporting the condensate storage tank (CST) volume requirements had not been updated as a result of stretch power and that the design basis for the use of the dedicated Unit 1 condensate volume stored in each Units' CST in accordance with Technical Specifications was not well defined.

This evaluation defines the safety related design bases for the condensate volume dedicated for Unit 1 which is stored in the Unit 1 and Unit 2 Condensate Storage Tanks and the capacity for the Unit 1 Atmospheric Dump Valves (ADV), which are integral in the determination of required condensate volume. Plant documentation (FSAR, Design Basis Documents, Operating Procedures and setpoints) concerning Unit 1 condensate requirements and ADV capacity were reviewed against the defined criteria and updated as necessary.

Revision 1 combines the condensate requirements for decay and RCP heat removal into a single curve in accordance with CR 98-0998. It also provides documentation to demonstrate design basis compliance with the inclusion of S/G secondary side water sensible heat removal. This update was initiated to incorporate NRC audit comments.

SAFETY EVALUATION SEMS-97-105
REVISION 0

OPERABILITY REQUIREMENTS FOR EDG BUILDING AIR INTAKE PLENUM MAINTENANCE ACTIVITIES

Summary:

This evaluation demonstrates that plant operation with one of the four air intakes completely blocked on an EDG building bay is acceptable during maintenance activities on the air intakes. Requirements for maintenance activities are developed to ensure the operability of the EDG air intake plenum.

This evaluation demonstrates that blocking one of four air intakes on an EDG Building bay (i.e., one of the two screen areas within one air intake plenum) during maintenance activities is acceptable.

This evaluation demonstrates that plant operation with one of the four air intakes on an EDG Building bay completely blocked is acceptable. The evaluation was developed to allow temporarily blocking one intake during maintenance activities and is not intended to be a permanent condition. While developed to allow replacement of the bird screens on the front of the intake louvers, this safety evaluation has been written to generally allow maintenance activities on the air intake plenums as long as certain minimum requirements are met.

SAFETY EVALUATION SEIS-97-108
REVISION 0

REACTOR PROTECTION SYSTEM - LOGIC MATRIX TEST STATUS LIGHTS

Summary:

The design of the Reactor Protection System includes built-in test features and status indication lights, which facilitate periodic surveillance testing of the actuation logic circuits. The lights are also used to provide the status of each stage of the actuation logic when surveillance testing is not being performed on the system. Section 7.2.1.6d of the Unit 1 FSAR, and Section 7.2.1.1.9.5 of the Unit 2 FSAR, describe the periodic testing of the RPS Trip Path circuits and Reactor Trip Switchgear Breakers. These sections contain the following statement:

> "Proper operation of all coils and contacts is verified by lights on a trip status panel."

At the present time, a status indication light for the UV trip coil of TCB-6 is inoperative. At various times in the past other RTSG breaker trip status lights have also been inoperative for a short period of time. For example, CR 96-2944 describes a condition where two RTSG Breaker Shunt Trip Coil status light circuits were inoperative. In this previous case the status lights were repaired on a expedited basis, and were returned to service before the grace period expired on the required RPS surveillance. At that time, contingency plans were developed which included preparation of a draft safety evaluation to justify use of test equipment in lieu of the inoperable status lights. The TCB-6 UV trip coil status light is also being repaired on an expedited basis. However, the repair effort may extend beyond the allowed grace period for the RPS logic matrix functional test due to availability of spare parts. Therefore, it may be desirable to perform the periodic logic matrix test, which is required by the Technical Specifications, by using a Digital Volt Meter in lieu of the inoperative status light. This evaluation documents the design and licensing requirements for the Reactor Protection System status indication lights, and provides justification for a change to the FSAR that will allow use of test equipment in lieu of an inoperative light to satisfy a surveillance requirement.

SAFETY EVALUATION SEIS-98-002
REVISION 0

ALTERNATE NIS CONTROL CHANNEL ARRANGEMENT

Summary:

The St. Lucie Unit 1 Nuclear Instrumentation System (NIS) normally operates with two linear power range control channels. The detector #10 for Control Channel 2 (CC2) was replaced during the Fall 1997 refueling outage and could not be fully calibrated until a 25% power level was reached. Therefore, it was to be unavailable for use following startup physics testing until calibration was completed.

This safety evaluation provides an assessment of an alternative control channel arrangement for the NIS. This involves connecting the detector signal from linear power range control channel 1 to both linear power range control channel (CC1 & CC2) outputs. This provides a power level signal to the 1B Low Power Feedwater Regulating System which normally receives its input from control channel 2 until channel 2 can be calibrated.

The use of the alternative arrangement involves changes to the reactor regulating system, the low power feedwater system, the power ratio calculator and recorders JR-010 & 012.

SAFETY EVALUATION SEFJ-98-003
REVISION 2

OPERATION OF ST. LUCIE 1 WITH REMOVAL OF A SINGLE RCS HOT LEG RTD INPUT TO RPS

Summary:

This safety evaluation assesses the effects on St. Lucie Unit 1 safety analyses of taking a hot leg Resistance Temperature Detector (RTD) out of service, and not feeding its input to the Reactor Protection System (RPS). RTD TE-1122HC feeding into RPS channel 'C' was found to perform inaccurately during the operation of St. Lucie Unit 1 Cycle 15 (January 29, 1998).

RTD TE-1122HC was identified to provide erratic hot leg temperature indications. Each of the two reactor coolant system (RCS) hot legs have 4 RTDs. with one RTD from each hot leg feeding into one RPS channel. Removal of the RTD TE-1122HC from service will result in RPS channel 'C' getting hot leg temperature reading from only one RCS hot leg (RTD TE-1112HC). Although temperature indication from one RCS hot leg is deleted from channel 'C', channel 'C' will continue to remain operable as justified in this evaluation.

The impact on plant safety is evaluated here assuming the removal of one RTD input from RPS channel 'C'. This evaluation however, does not address the actual modifications to be made to remove the RTD TE-1122HC input to the RPS.

Revision 1 of this evaluation provided additional justification for the operability of RPS channel 'C' with one hot leg RTD input removed. The conclusions of the safety evaluation remain unchanged.

Revision 2 of this evaluation clarifies that the analysis and conclusions made for removal of TE-1122HC inputs from the RPS are applicable to any single RCS hot leg RTD input.

SAFETY EVALUATION SEIS-98-006
REVISION

ESFAS - REPLACEMENT OF 24 VOLT POWER SUPPLY

Summary:

The Engineered Safety Features Actuation System (ESFAS) includes two pairs of auctioneered 24-volt power supplies in each actuation logic train. One auctioneered pair supplies power to the normally energized (i.e., fail safe) actuation relays, and the other auctioneered pair supplies the normally de-energized actuation relays. An auctioneered power supply pair is used to prevent spurious actuation of the normally energized actuation relays in the event one power supply fails.

During the ESFAS Monthly Functional Test, it was determined that the output voltage of both individual power supplies in the S2/M2 auctioneered pair were abnormally low. This auctioneered pair supplies the normally energized actuation relays in Train A. The circuit is designed to facilitate replacement of a failed power supply without causing or preventing an actuation signal. However, since both power supplies are suspect, removal of either could cause spurious actuation. The purpose of this Safety Evaluation is to justify temporary installation of a power supply in parallel with the failed auctioneered pair. The temporary power supply maintains the supply voltage to the associated actuation relays during the replacement of the failed power supplies.

SAFETY EVALUATION SEFJ-98-007
REVISION 1

FSAR UPDATE FOR INCORPORATING SMALL BREAK LOCA REANALYSIS WITH ASYMMETRIC HPSI FLOWS

Summary:

Condition Report 97-1437 addressed the operability of St. Lucie Units 1 and 2 with respect to the HPSI flow asymmetry concern identified at Millstone Unit 2. The operability of St. Lucie Units 1 and 2 was justified based on the available analysis margin and the margin in the actual pump performance. For Unit 2, the analysis HPSI flows conservatively accounted for the loop asymmetries. The small break LOCA (SBLOCA) analysis for Unit 1 however was required to be redone with the assumptions of the existing analysis but including the effects of flow asymmetry. This analysis by Siemens Power Corporation (SPC) is complete and meets all the 10 CFR 50.46 acceptance criteria. The PCT in the revised analysis increased by 149F to 1953F.

This safety evaluation, updates the St. Lucie Unit 1 UFSAR to incorporate the revised SBLOCA analysis performed by SPC for the effects of safety injection loop asymmetries. The loop asymmetries have been determined to result in asymmetric high pressure safety injection (HPSI) flows in the cold leg injection lines (CR 97-1437).

The SBLOCA peak cladding temperature (PCT) increased from 1804F to 1953F and the limiting break size shifted from 0.10 sq. ft.to 0.05 sq. ft. The loop asymmetries were determined to produce a change of 1% in the minimum 3-loop flow. This change was sufficient to impact significantly the results of break sizes smaller than the current limiting break size (0. 1 sq. ft.). The smaller break sizes lead to a slower reactor coolant system depressurization and can potentially delay the PCT turnaround time substantially with reduced HPSI flow.

Revision 1 corrects typographical errors and includes statements related to the calculated cladding oxidation values. The conclusions of the evaluation remain unchanged.

SAFETY EVALUATION SENS-98-011
REVISION 1

ONSITE STORAGE OF RADIOACTIVE MATERIALS IN A REMOTE RCA

Summary:

The purpose of this safety evaluation is to identify the conditions under which a separate radiation controlled area (RCA) for the storage of contaminated materials may be developed within the FPL owner controlled area of the St. Lucie site but situated non-contiguous with the existing RCA that encompasses the Unit 1 and 2 reactor containment, auxiliary and fuel handling buildings. Certain byproduct material (as defined by 10CFR 20.1003) typically used during refueling outages is currently stored in the existing radiation controlled area. It is desired to retain this material for future use but, while not in use, storage of this material requires sufficient space such that its presence is judged to interfere with the normal operational and maintenance functions occurring inside the existing radiation controlled area.

The proposed remote RCA will be used to store equipment necessary for outage work on-site. Its location and design will ensure that the dose to members of the public remains within acceptable limits.

Revision 1 to this evaluation includes Facility Review Group (FRG) comments related to the discussion of specific gravity for containers to be stored in the remote RCA. No changes to the UFSAR change package prepared for Revision 0 of this evaluation were required by Revision 1.

Even though this evaluation was prepared and approved, it has not been implemented in the field.

SAFETY EVALUATION SEMS-98-012
REVISION 0

ADDITION OF DIMETHYLAMINE TO THE SECONDARY SYSTEM

Summary:

This safety evaluation addressed the addition of dimethylamine (DMA) to the secondary system during all modes of operation for St. Lucie Units 1 and 2. The addition of DMA will be in conjunction with the current hydrazine and ammonia chemistry program, not as a replacement. This evaluation demonstrates that addition of DMA to the secondary system is not detrimental to plant operations and systems.

St. Lucie secondary chemistry currently operates on all volatile treatment (AVT) and controls the secondary system pH with hydrazine, and ammonia formed from hydrazine decomposition. This typically results in a feedwater pH between 9.3 and 9.6. In order to reduce the transport of iron to the steam generators, the at temperature pH (pHt) has to be raised throughout the balance of plant systems by increasing the concentration of ammonia or adding an additional amine that will raise the pHt of these systems without increasing the amine loading of the SG demineralizers. Several alternatives were analyzed and DMA was selected due to its more advantageous characteristics. DMA has also been shown to remove corrosion product buildup from the water flow venturies and assist in the removal of sludge from the SGs at other operating plants.

SAFETY EVALUATION SEMS-98-016
REVISION 0

TEMPORARY LEAK REPAIR ENCLOSURE ON MV-09-7 BYPASS LINE ELBOW

Summary:

Condition Report 98-0372 concerns operational leakage discovered in the socket weld joint for the downstream elbow of the MV-09-7 bypass line. This elbow is adjacent to valve V09188 and is shown on Drawings 8770-G-080 Sheet 3 and 8770-G-125 Sheet BF-M-6. This evaluation addresses the temporary clamping of the steam leak with a leak repair enclosure to encapsulate the elbow and a clamp to absorb the axial loading adjacent to the affected socket weld.

The leak is located in the socket weld of the downstream bypass line elbow. The leak is characterized as a feedwater leak creating a visible jet of steam about 2 foot long creating a moist area on the adjacent lagging and some plywood below. The leak is considered minor in nature (less than 5 gallons per hour based on observation of the wetted area below the leak) but needs to be stopped to minimize further steam cutting of the MV-09-7 bypass line elbow weld.

This evaluation demonstrates the acceptability of the addition of a temporary leak repair on the MV-09-7 Bypass Line Elbow. The leak repair enclosure being installed via this safety evaluation is considered a non-Code repair to stop the leakage to prevent steam cutting of the MV-09-7 bypass line elbow weld.

SAFETY EVALUATION SEMS-98-017
REVISION 0

BIOCIDE TREATMENT OF PSL CLOSED COOLING WATER SYSTEMS

Summary:

This safety evaluation addresses the addition of biocides on an 'as needed' basis as determined by sampling and analysis of the various closed cooling water systems at St. Lucie Units 1 and 2. Microbiological contamination of closed cooling water systems has been identified as a problem by many of the U.S. nuclear utilities. Testing at St. Lucie plant and Turkey Point plant has identified the presence of microbiological organisms in several of the closed cooling water systems. At St. Lucie the following systems are included for this evaluation: Component Cooling Water (CCW), Turbine Cooling Water (TCW), and Steam Generator Blowdown Cooling Water (SGBDCW). Some utilities have experienced equipment degradation and/or failure as a result of material corrosion that was caused by microbiological activity. Biocide treatment is a standard practice in other industries. Although there are numerous biocides on the market, only two non-oxidizing biocides will be evaluated in this safety evaluation. They have both been evaluated and successfully used at other nuclear utilities. These two biocides are Glutaraldehyde and Isothiazolin.

Both of these biocides produce a rapid kill of both aerobic and anaerobic microbes and each has a short effective active response time, typically less than 48 hours. After that time the biocide is no longer present in the system. Two biocides are required to develop an effective biological contaminant control program. A single biocide can provide significant induction in biological contaminant; however, some bacteria develop a resistance to a single biocide. The second biocide becomes necessary to eliminate the resistant bacteria. The effectiveness of the biocide selected is also dependent on the ability of the biocide to contact the microbiological organism. To assist the biocide in penetrating and removing biomass the chemical industry also recommends the use of a biodispersant in conjunction with the biocide application to ensure that the systems are effectively cleaned at the first application. This evaluation also addresses the use of biodispersants.

SAFETY EVALUATION SEIS-98-021
REVISION 0

RSPT SIGNAL RESTORER EVALUATION

Summary:

Two non-Class 1E systems of Control Element Assembly (CEA) position indication are provided for display CEA position on the main control panel. The systems are the Pulse Counting CEA Position Indication System and the Reed Switch CEA Position Indication System. The Reed Switch CEA Position Indication System obtains its signals from a Reed Switch Position Transmitters (RSPT) which consists of a network of resistors connected in series. As the CEA moves up and down, a magnet moves with it causing the reed switches to change state and the resistance value to change proportionately. With voltage applied to the RSPT, it acts as a stepwise potentiometer that provides an output voltage proportional to CEA position.

The open circuit or high resistance of the RSPT resistor bank or a stuck reed switch will cause the Reed Switch Position Indication to be inoperable for the applicable CEA which requires a Limiting Condition for Operation in accordance with Technical Specification. A RSPT Signal Restorer Device is available which electronically compensates for these two failure mechanisms.

The purpose of this safety evaluation is to document the acceptability of utilizing the RSPT Signal Restorer to reinstate partial or full operability of a failed RSPT and to establish plant restrictions with device installed.

SAFETY EVALUATION SEMS-98-024
REVISION 0

CONTROL ROOM PAINTING - 10CFR50.59 EVALUATION

Summary:

REA SLN-97-091-90 requests various upgrades to the St. Lucie Control Rooms and the simulator. One of these upgrades involves repainting of the control panels. This evaluation addresses the impact of the painting activities and the final installed paint on plant operation and nuclear safety. Relevant restrictions were identified to insure that the criteria in the FSAR and Technical Specifications are met.

SAFETY EVALUATION SENS-98-028
REVISION 0

DUMMY FUEL ASSEMBLY USE - UFSAR CHANGES

Summary:

The UFSAR review process identified a discrepancy related to the use of dummy fuel assemblies to check the alignment of spent fuel storage racks and fuel handling systems (Masterdat item no.198). Plant procedures do not check the alignment with dummy fuel assemblies as specified in the UFSAR. Instead, funnels are provided to guide the fuel assemblies into the spent fuel pool rack locations.

Safety evaluation JPN-PSL-SENS-94-025 was issued to provide UFSAR corrections for Unit 1 since dummy fuel assemblies were only used to test the fuel handling systems during the manufacturing and post-installation periods. Subsequently, the Unit 2 UFSAR was revised with safety evaluation JPN-PSL-SENS-95-021, Rev. 1, for similar inconsistencies related to the use of dummy assemblies during periodic testing of the spent fuel storage racks and fuel handling equipment.

The present evaluation justifies and provides the necessary Unit 1 and Unit 2 UFSAR change packages to address the alignment checks of spent fuel pool storage racks. In addition, the Unit 1 text is being written in a similar manner to Unit 2 to add a discussion about the use of a test weight for fuel handling crane testing prior to lifting the fuel assemblies.

An editorial change was also addressed in both UFSAR change packages related to the liquid penetrant testing which was a one time test performed during the preoperational testing period for both units.

SAFETY EVALUATION SENS-98-032
REVISION 1

UFSAR CHANGES TO THE RCS LEAK DETECTION DESCRIPTION

Summary:

Section 5.2.4.5 of the Unit 1 UFSAR, describes the instrumentation provided for reactor coolant system leak detection. CR 96-2572 identified that the table referenced in this section, Table 5.2-11, contained inconsistencies between the information presented and actual instrument data. This evaluation addresses the change to the Unit 1 UFSAR description of the RCS leak detection instrumentation contained in Table 5.2-11. The changes apply specifically to the quench tank and SIT level and pressure characteristics, and associated monitoring instrumentation.

The changes described in this evaluation will correct the data contained in Table 5.2-11 to reflect the installed equipment. In addition, minor corrections are being made to the text in UFSAR section 5.2.4.5 relative to allowable leakage and makeup capability. Further, the ability to detect leakage has been reviewed against the applicable requirements and found to be acceptable.

The revised data reflects actual tank level and pressure characteristics and corrected instrument range information. In addition, the corrected information was reviewed to confirm that the leak detection instrumentation meets the applicable functional requirements.

SAFETY EVALUATION SENS-98-045
REVISION 0

CCW & ICW PUMP AVAILABILITY TESTING - UFSAR CHANGES

Summary:

The UFSAR review process identified a discrepancy related to the procedure for testing the Intake Cooling Water (ICW) pumps. According to the UFSAR, the ICW pumps are rotated in service periodically to ensure their continued availability during emergency conditions (Masterdat item #15 and Condition Report 96-2573). Plant procedures did not address the implementation of this UFSAR commitment. A similar statement was also provided for both Component Cooling Water (CCW) and ICW pumps in another section of the UFSAR.

The present evaluation justifies and provides the necessary Unit 1 UFSAR change package to correct these discrepancies in the UFSAR.

SAFETY EVALUATION SENS-98-046
REVISION 0

SOLID WASTE MANAGEMENT SYSTEM - UFSAR CHANGES

Summary:

The UFSAR review process identified several discrepancies related to equipment in the solid waste system which has not been used at the plant. Also, solid waste handling procedures described in the UFSAR were not being performed as described, and other procedures and methods had taken their place. These discrepancies were documented in Masterdat items #132, 134, 290, and 291, and Condition Reports 96-2538 and 96-2592.

The present evaluation justifies and provides the necessary Unit 1 UFSAR change package to correct the UFSAR Section 11.5 and make it consistent with the actual operation of the plant. Changes to Section 11.5 include:

1. Deletion of discussions about drum rollers, volume tank and the process of boric acid/waste management concentrator bottoms solidification which is no longer performed,
2. Deletion of discussion about the solid waste baler compactor, which is no longer used,
3. Addition of discussion about the use of portable ion exchanger resin dewatering systems, and
4. Labeling solid radwaste generation tables as "estimated" values developed prior to initial plant startup as well as the incorporation of some minor editorial changes.

SAFETY EVALUATION SENS-98-050
REVISION 0

UFSAR CHANGES TO CONTAINMENT FAN COOLER LUBRICATION FREQUENCY

Summary:

PMAI 96-11-325 described a discrepancy between the Unit 1 UFSAR Section 6.2.2.3.3 and existing plant maintenance procedures Section 6.2.2.3.3 identifies a Containment Fan Cooler (CFC) motor and fan bearing lubrication frequency of six months. These bearings are maintained on an eighteen month frequency consistent with unit operating cycles and ALARA considerations.

The purpose of this evaluation was to demonstrate the acceptability of existing fan cooler bearing maintenance intervals, which are aligned with unit operating cycles. The information provided in UFSAR Sections 6.2.2.3.3 concerning the frequency of lubrication for the CFC fan and motor bearings was revised accordingly. This brings the UFSAR commitments for fan cooler maintenance into agreement with current plant maintenance practices.

SAFETY EVALUATION SENS-98-052
REVISION 1

DISABLING THE LOW TURBINE VACUUM TRIP

Summary:

This Safety Evaluation demonstrates the acceptability of operating with the low vacuum turbine trip defeated. The primary purpose of the low vacuum turbine trip valve is to mechanically trip the turbine when the vacuum degrades to a point that creates excessive back pressure in the condenser. Defeating the low vacuum turbine trip, by engaging the low vacuum trip bypass lever, removes the turbine's ability to automatically trip and requires operator action to trip the turbine on low condenser vacuum (high condenser back pressure).

Defeating the low vacuum turbine trip will not have a negative impact on plant safety or operation and is acceptable from a stand point of nuclear safety.

Revision 1 clarified item #1 of the 'Actions Required' section. This item requires that Operations shall alert the operators by a Night Order that the low vacuum trip has been temporarily disabled and that this item be included on the Shift Turnover Check Sheets until the subject TSA was removed.

SAFETY EVALUATION SENS-98-054
REVISION 0

INSTRUMENT RANGE AND ACCURACY IN UFSAR

Summary:

Twenty-six (26) different tables in the UFSAR contain instrument ranges and accuracies. These values were originally included in the FSAR during the licensing process to demonstrate that appropriate instrumentation was being used. There appears to be no specific requirement to maintain these values in the UFSAR. Furthermore, having these values listed in the UFSAR causes an additional burden in the plant change process by forcing 10 CFR 50.59 evaluations for insignificant instrument accuracy and range changes. This evaluation has been written to replace all range and accuracy values from the UFSAR with a note identifying the general design requirement.

A review of various licensing documents determined that requests in those documents for range and accuracy details is specific for the Preliminary Safety Analysis Report stage. In fact, Appendix A of R.G. 1.70 specifically states. "this appendix describes safety-related interfaces ... that should be presented at the preliminary design stage," (emphasis added). The identification of specific instrument ranges and accuracies is considered to be a detail necessary only for the plant licensing stage and is not considered a necessary detail to be maintained in the current UFSAR.

New instruments or modified instrument parameters are provided in accordance with Nuclear Engineering Quality Instructions (Qls). As such, whenever a change is made, instrument ranges and accuracies are properly considered and selected. Instrument ranges are selected in accordance with standard engineering practices. Likewise, instrument accuracies are selected such that existing instrument loop performance and safety analysis assumptions remain valid. Where applicable, instrument accuracies are also evaluated for their impact on setpoints in accordance with the FPL Setpoint Methodology. For post-accident monitoring instrumentation, the instrument range and accuracies required by Regulatory Guide 1.97 are considered.

SAFETY EVALUATION SENS-98-056
REVISION 0

UFSAR UPDATE FOR GRID STABILITY ANALYSIS

Summary:

The UFSAR Review Project identified the fact that the present grid stability analysis contained in the Unit 1 & 2 UFSARs are out of date. It was also determined that NRC Information Notices (INs) suggested that licensees should periodically review their grid analysis. The latest IN on the subject states that the NRC Standard Review Plan Section 8.2 provides current guidance for assessing the adequacy of the offsite power system; therefore, FPL Transmission System Planning was requested to update this analysis.

The Standard Review Plan (SRP), section 8.2.III.1.f, requires that grid stability analysis show the preferred source of power is not completely lost as a result of the outage of a single element which disconnects: a) the grids largest source of power, b) the grids largest load, or c) most critical transmission circuit. An updated dynamic stability analysis has been performed and results of that analysis are being incorporated into both UFSARs.

Additionally, all references to 240 kV system were revised to 230 kV. FPL's 230 kV system was referred to as 240 kV prior to 1988. The nominal design voltage of these facilities has always been 230 kV which is a standard transmission voltage level in North America. FPL now refers to these transmission facilities as 230 kV to avoid confusion on the line ratings which are based on amperes.

SAFETY EVALUATION SECS-98-064
REVISION 0

OPERATION OF SPENT FUEL CASK CRANE "RESTRICTED ZONE" ADJACENT TO FUEL HANDLING BUILDING

Summary:

Administrative Procedure (AP) No. 0010438 restricts the Spent Fuel Cask Crane from entering an area adjacent to the Fuel Handling Building. This is applicable for both Units 1 and 2. The 'restricted zone' is located on the east and north side of the building and is identified on figure 5, 'Fuel Area Safe Load Path' of AP-0010438. The Cask Cranes are restricted from traveling into the 'restricted zone' unless both hooks are in the full up position or it travels south through the two-foot access located at the center of the L-shaped door. Currently, crane limit switches are set to prevent movement of the cranes in accordance with the above restriction.

Both the Unit 1 and 2 Spent Fuel Cask Cranes are classified as Quality Related - II/I Seismic to prevent adverse seismic interaction with the Safety Related Fuel Handling Building located below and adjacent to the crane.

Maintenance has requested to allow movement and use of the Spent Fuel Cask Crane within the 'restricted zone' to remove material, casks, and equipment that may be located with in the 'restricted zone'.

Temporary alteration of the Unit 1 and 2 Spent Fuel Cask Crane limit switches to allow movement of either crane within the 'restricted zone' adjacent to the associated Fuel Handling Building, while the 'L-shaped' door is closed, has been evaluated and is acceptable.

The purpose of this evaluation is to verify that this load handling operation is consistent with the NUREG-0612 and FSAR requirements. Based on this evaluation, implementation requirements are provided to ensure conformance to NUREG-0612 and FSAR.

SAFETY EVALUATION SENS-98-075
REVISION 0

<u>'98 FSAR REVIEW FIND REQ' CHANGES OR CLARIFICATION TO FSARS IN ACCORDANCE W/10CFR50.59 FOR I&C ISSUES</u>

Summary:

The purpose of this safety evaluation is to provide a method to update, correct, or add clarifications to the FSAR following an FSAR review project on selected systems. User comments were issued for those comments, corrections, updates, or clarifications, which could be classified as administrative or editorial in nature. The changes addressed by this safety evaluation did not meet the criteria to be addressed by a user comment alone. However, these changes are minor in nature and did not warrant individual stand alone 50.59 evaluations. This evaluation addresses a variety of changes related to I&C and electrical subjects. Safety Evaluation PSL-ENG-SENS-98-062 will address the other subject fields.

SAFETY EVALUATION SEMS-98-080
REVISION 0

ISOLATION OF THE NITROGEN SUPPLY TO THE NaOH TANK

Summary:

This safety evaluation supports isolation of the nitrogen supply to the NaOH tank to allow tank maintenance activities. It also justifies the use of an open vent valve or removed rupture disc to provide vacuum relief capability. In addition, it revises the description of the non-safety related nitrogen supply system in the UFSAR.

SAFETY EVALUATION SENS-98-084
REVISION 0

CONDENSATE STORAGE TANK SAMPLING - UFSAR CHANGES

Summary:

The 1996 UFSAR/procedure consistency review process identified two minor discrepancies with respect to the sampling/monitoring of the Unit 1 condensate storage tank (CST). These discrepancies were documented as Masterdat Item #s 920 & 921 and are summarized below.

Item #920
UFSAR Section 9.2.8.2 states that makeup water for the CST is passed through a degasifier prior to storage in the CST and that provisions have been made to recirculate the CST contents through the degasifier if dissolved oxygen levels are too high. This description is inaccurate; the degasifier is not used. Dissolved oxygen levels in the CST are limited by a nitrogen blanket maintained on the tank and are reduced via the condenser make-up spray system.

Item #921
UFSAR Section 9.2.8.3 states that periodic CST grab samples are taken and analyzed to ensure there is no radioactivity buildup in the tank. Such sampling is not performed since the CST is not expected to contain significant quantities of radioactivity. This practice is in compliance with 10 CFR Part 20, which requires monitoring which is "reasonable under the circumstances."

The safety evaluation provides the justification for UFSAR changes to resolve the above discrepancies.

SAFETY EVALUATION SENS-98-085
REVISION 0

FLASH TANK OPERATION - UFSAR CHANGES

Summary:

The 1996 UFSAR review process identified two discrepancies related to the current operation of the flash tank applicable to both units (Masterdat items 281 and 945). According to the findings, the flash tank is bypassed to the holdup tanks in each unit during normal plant operations which is contrary to the operation described in the UFSAR.

The present evaluation provides the necessary Unit 1&2 UFSAR change packages to justify current and future operating practice. The option for the flash tank use will be kept in the UFSAR for plant operations when a minimum amount of fuel failures is experienced or whenever the hydrogen or fission gas stripping function is required.

SAFETY EVALUATION SENS-98-090
REVISION 0

USE OF PRIMARY SYSTEM IN-LINE SAMPLE ANALYSIS FOR DISSOLVED GASES

Summary:

This evaluation examines the use of a primary system in-line sample analyzer for monitoring dissolved gases (oxygen and hydrogen) in the RCS during normal operations, plant shutdown, and plant startup. Currently, hydrogen and oxygen are analyzed by grab sample analysis, and this method is described in the UFSAR. The proposed change uses in-line gas analyzers that provide the periodic samples for continuous operation and the continuous gas reading during plant shutdown when the RCS hydrogen is being added back into the RCS, and can be used during normal plant operation.

Use of the in-line analyzer for monitoring dissolved gases in the RCS does not alter the configuration of the sampling system. The in-line analyzer utilizes the connections normally used for grab sample analysis. As such, the UFSAR is being updated to describe the new in-line analyzer as an acceptable means for monitoring dissolved gases in the RCS in addition to gases analyzed by grab sampling.

SAFETY EVALUATION SENS-98-091
REVISION 0

USE OF CVCS PURIFICATION/DEBORATING ION EXCHANGERS - UFSAR CHANGES

Summary:

The 1996 UFSAR review process identified four discrepancies related to the operation of purification and deboration ion exchangers for both units (Masterdat items 855, 1585, 1633 and 1634). According to the noted discrepancies, the UFSAR states that each of the Chemistry and Volume Control System (CVCS) ion exchangers are used for a specific purpose, however, per current plant operating procedures, all three CVCS ion exchangers are interchangeable. Also, the UFSAR states that one ion exchanger (the deborating) is used to reduce boron concentration at the end of cycle (EOC), when in fact current plant operations uses any one or two ion exchangers for deboration activities. The present evaluation provides the necessary Unit 1&2 UFSAR Change Packages to address these discrepancies and to justify current operating practice.

SAFETY EVALUATION SENS-98-096
REVISION 0

LIQUID WASTE MANAGEMENT SYSTEM - UFSAR CHANGES

Summary:

The 1996 UFSAR review effort identified a number of discrepancies for the Unit 2 UFSAR related to the Liquid Waste Management System (LWMS) operations described in UFSAR Section 11.2. Specifically, the operation of the boric acid concentrators, waste concentrator and supporting components is not performed as described in the UFSAR. These components are no longer used for both units, and alternative waste processing methods have been implemented. It is the intent of this safety evaluation to analyze and evaluate these methods for regulatory compliance and to revise the Unit 1 UFSAR accordingly. As a result of the changes in this evaluation, Masterdat item 278, related to periodic testing of the LWMS, will be partially addressed. Also, this evaluation justifies the abandonment of the liquid waste discharge radiation monitor RE-26-64.

In addition, several other minor editorial items are discussed in this evaluation and UFSAR changes are provided. This evaluation justifies and provides the necessary Unit 1 UFSAR change package (FCP) to correct the above discrepancies and make the LWMS description consistent with the actual operation of the plant.

SAFETY EVALUATION SEMS-98-108
REVISION 0

BORON DILUTION SYSTEM ABANDONMENT

Summary:

CR 98-0669 concerns the Unit 1 boric acid dilution system which was installed per PC/M 80119. After installation, several problems were identified during startup of the system, and as a result, the PC/M was cancelled. The equipment installed by the PCM is still installed and connected to operating systems (CVCS, Instrument Air and Primary Water) but because the PC/M was cancelled, the design documentation was not updated to reflect the field configuration.

This evaluation addresses the impact on plant operation and nuclear safety with regard to abandoning this system in place. Relevant restrictions were identified to insure that the criteria in the FSAR and Technical Specifications were met.

SAFETY EVALUATION SENS-98-110
REVISION 0

REACTOR COOLANT PUMP MOTOR COOLING - UFSAR CHANGES

Summary:

In the 1996 UFSAR Consistency Review, Masterdat Item 676 described a discrepancy between Unit 1 UFSAR Section 9.2.2.3.1 and existing plant procedures. Section 9.2.2.3.1 states that the pump manufacturer recommends the Reactor Coolant Pump (RCP) motor not be operated for more than 7 minutes if cooling water flow to the motor is lost. Procedures require the RCP be secured if cooling is lost for 10 minutes. In a review of documentation and discussions with the pump and motor vendors, it was confirmed that the 7 minute time limit was provided arbitrarily by the pump vendor and that there was no basis for the value.

The adverse effect of loss of cooling water to the motor would be overheating of the motor bearings and stator. Sufficient instrumentation, alarms and procedures are provided for these parameters to warn of loss of cooling to the RCPs and provide sufficient time to take appropriate action. Therefore, this evaluation provides the justification to revise the UFSAR to remove the 7 minute requirement.

SAFETY EVALUATION FPER-99-003
REVISION 0

FIRE PROTECTION EVALUATION FOR HVAC DUCT PENETRATION

Summary:

The purpose of this safety evaluation is to evaluate the installation of HVAC ducting through fire barriers without a fire damper. This evaluation was necessary to justify the installed ducting/pipe as acceptable for the as-installed plant conditions, and to update the UFSAR. There are no changes to plant operation, operating practices, operating philosophy, or safety as a result of these changes, nor are there any restrictions on plant operation.

SAFETY EVALUATION SENS-99-005
REVISION 0

<u>CHANGES TO REACTOR COOLANT AND REACTOR MAKEUP WATER CHEMISTRY REQUIREMENTS</u>

Summary:

This safety evaluation addresses changes to St. Lucie Unit 1 UFSAR Table 9.3-8 and Unit 2 UFSAR Table 9.3-5, Reactor Coolant and Reactor Makeup Water Chemistry. This change involves updating the current reactor coolant and reactor makeup water chemistry specifications to be consistent with current industry standards. This revision also reformats the tables so the data is consistent between both units. The only difference in chemistry parameters listed for the two units is the fluoride limit. This reflects the difference in the fluoride limits listed in the Unit 1 and Unit 2 Technical Specifications.

The Chemistry limits for the RCS parameters are being tightened within the current limits or revised to reflect current Technical Specifications or current industry recommendations. These chemistry specifications provide guidance for maintaining the reactor coolant chemistry program. The identification of chemistry conditions that lead to long term NSSS integrity is a dynamic process. Some of the specifications and chemistry control practices given in the UFSAR are now obsolete and need to be revised to reflect current industry research and experience. The Electric Power Research Institute (EPRI) and ABB Combustion Engineering (ABB/CE) have both issued revisions for primary chemistry control parameters. Typically the revised limits are more restrictive than the original values and rely on consistency between the parameters, such as the relationship between boron/lithium, pH, and conductivity.

SAFETY EVALUATION 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 is to demonstrate that single wythe hollow CMU wall assemblies (nominal 12 inch or greater in thickness) or multi-wythe hollow CMU wall assemblies (nominal 8 inch or greater in thickness) can be implemented without prior NRC approval in 3-hour fire resistance rated wall assemblies. The basis for this determination was documented in this evaluation. This evaluation was necessary to justify the existing CMU wall assemblies which are 12 inch 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. There are no changes to plant operation, operating practices, operating philosophy, or safety as a result of these changes, nor are there any restrictions on plant operation.

SAFETY EVALUATION SENS-99-006
REVISION 0

STARTING THE SDC SYSTEM

Summary:

Shutdown Cooling (SDC) System operation is initiated at a point in the plant cool-down when the RCS conditions drop below SDC design conditions. The method of initiating SDC System operation is described in Unit 1 UFSAR Section 9.3.5.2.2. Condition Report (CR) 96-2840 identified that the method of SDC initiation described in the Unit 1 UFSAR disagreed with the method employed by the Unit 1 Operating Procedure.

Further review by Engineering determined that the SDC operating procedures for both units required significant changes to preclude unacceptable thermal transients to the SDC heat exchangers. A thermal stress analysis of the SDC heat exchangers to determine their useful life under rapid heatup and cooldown transients provides input information for this safety evaluation to justify UFSAR changes and eliminate overly conservative warmup rates.

This Safety Evaluation provides the Unit 1 UFSAR changes that correct the discrepancies by aligning the Unit 1 UFSAR description of SDC warm-up with the operating procedures. Also provided are the recommended procedural changes for initiating SDC system operation that were incorporated into the operating procedures.

SAFETY EVALUATION SEES-99-008
REVISION 0

UFSAR UPDATE FOR CABLE TRAY LOADING

Summary:

The UFSAR Review Project identified the fact that the existing description of cable tray loading design criteria given in sections 7.1.2.3 and 8.3.1.2.3.e) of the Unit 1 UFSAR were not accurate. The original design criteria for cable trays was a maximum 40 percent fill, based on cable tray and cable area, and also noted that a certain percentage of trays exceeded this. It was discovered during the process of converting the original Cable and Conduit List, which was maintained manually, to a computerized database that some trays exceeded this criteria. Calculation RSB-3, issued in 1983, was prepared to address the cable tray fill overloads and develop new criteria. The UFSAR was never changed to reflect the new criteria.

This evaluation provides the 10 CFR 50.59 review required to revise the UFSAR.

SAFETY EVALUATION SENS-99-010
REVISION 1

MAIN STEAM ISOLATION VALVE PERIODIC TEST DELETION OF PART STROKE TEST

Summary:

This safety evaluation justifies discontinuing the main steam isolation valve (MSIV) periodic test, part-stroke test. The part-stroke test is performed during normal plant operation via a local test panel. Testing is accomplished during normal operation by energizing the test stroke solenoid valves. The valve spindle and disk moves into the steam pathway approximately 5/8 inches and then returns to its original full open position. The MSIV part-stroke test is described in the Unit 1 UFSAR, Section 10.3.4. As such, this safety evaluation justifies the deletion of MSIV part-stroke testing and requires a change to the Unit 1 UFSAR.

The main steam isolation valves are designed to perform their safety function during or following a design basis earthquake and are designed such that no single failure causes both isolation valves to remain open. Note, however, that the main steam isolation valve (MSIV) test circuitry is physically separated from the safety related circuits required to close the MSIVs during an MSIS and, except during testing, these circuits are de-energized. The test circuit is automatically isolated from the safety related circuits should MSIS occur during a test.

Revision 1 to this safety evaluation provides clarification on why it is acceptable to delete part-stroke testing based on FRG comments. The changes to this evaluation were administrative and did not change any of the original conclusions to this evaluation.

SAFETY EVALUATION SEFJ-99-012
REVISION 0

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

Summary:

The purpose of this evaluation is 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 Infobulletin 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 SEIS-99-012
REVISION 0

SPLIT LOOP INSTRUMENTATION CALIBRATION

Summary:

This Engineering Evaluation documents the technical basis for expanding the use of Split Loop Instrumentation Calibration for St. Lucie Unit 1 and how this preferred calibration process is in compliance with existing plant design and Technical Specification requirements.

Split Loop Instrumentation Calibration is defined as the calibration process where the loop portions of instrument channels are recalibrated at different time intervals and returned to service prior to or after their associated transmitters. This is an improvement over the current maintenance practice where the complete channel is recalibrated from sensor to output device before returning to service, since it allows greater flexibility in maintenance scheduling.

In addition to compliance with Technical Specifications, this evaluation provides a comparison with the St. Lucie Unit 2 Split Loop Calibration engineering analysis and calibration activities conducted during and prior to the Fall 1998 Unit 2 Refueling outage. All these items provide an additional level of assurance that separating instrument loop component calibrations will have negligible effect on equipment performance and that these components will perform within design requirements.

SAFETY EVALUATION SEES-99-016
REVISION 0

OPERATION OF UNIT 1 DIESEL GENERATOR WITHOUT KW RECORDER

Summary:

The 1A Emergency Diesel Generator (EDG) Watt Recorder W-REC/954 failed. Condition Report 99-0306 requested that Engineering determine root cause and address EDG operability without the information provided by this recorder. The purpose of this Safety Evaluation is to determine the acceptability of utilizing other existing instrumentation for the determination of Diesel Generator loading and provide operator guidance in the use of diesel amperage to determine kilowatt loads during EDG surveillances and emergency load management conditions.

This evaluation concludes that the Unit 1 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.

Presently, diesel loading is determined using control room kilowatt recorders (W-REC/954 and W-REC/964) which indicate the electrical loads associated with the 1A and 1B EDGs. Other control room instrumentation in the form of diesel generator voltage, amperage and reactive power (Mvars), is also available to the operator should there be a situation where the kW recorder is unavailable. This evaluation evaluates the use of these alternate instruments and provides conservative values for operators use should the primary instrument become unavailable.

SAFETY EVALUATION SEFJ-99-018
REVISION 0

FSAR UPDATES FOR POST-LOCA BORON PRECIPITATION ANALYSIS & PRE-ACCIDENT IODINE SPIKE SGTR DOSE EVENT

Summary:

The purpose of this evaluation is to document the revised St. Lucie Unit 1 post-LOCA long term cooling boron precipitation analysis for inclusion in the St. Lucie Unit 1 Final Safety Analysis Report. This analysis is described in the FSAR Appendix 6C and provides requirements of timing and flows for the initiation of hot leg injection, during the recirculation phase of a LOCA. Additionally, this evaluation included in the FSAR the conclusions of dose consequences for the steam generator tube rupture (SGTR) event assuming a pre-accident iodine spike, as analyzed by the NRC. The SGTR analysis is described in the FSAR Section 15.4.4.

The boron precipitation analysis currently described in the FSAR Appendix 6C was performed as part of the original design basis analysis for St. Lucie Unit 1, with input parameters which may not directly reflect the current plant configuration. The current FSAR analysis results, however, have been evaluated to be applicable for the current operation based on the conservatisms in the existing analysis. The new analysis provides an updated analysis consistent with and bounding the current plant configuration.

During the review of the stretch power license amendment, the NRC calculated the dose consequences for the SGTR event with a pre-accident iodine spike. The conclusions were included in the stretch power license amendment safety evaluation report. The FSAR updates for stretch power did not include the NRC conclusions for this event. The pre-accident iodine spike case conclusion is included in the FSAR as it was a part of the basis for the NRC approval of License Amendment No. 48.

This evaluation includes the SGTR results for pre-accident iodine spike case, and the new analysis for the post-LOCA boron precipitation in the FSAR.

SAFETY EVALUATION SEFJ-99-020
REVISION 1

FUEL ASSEMBLY EXAMINATION AND RECONSTITUTION IN THE ST. LUCIE UNIT 1 SPENT FUEL POOL

Summary:

During Cycle 15, St. Lucie Unit 1 experienced increased iodine activity in the RCS that were indicative of failed fuel. Analysis of the isotopic content of the RCS both during steady state power conditions and just after a reactor trip that occurred on 8/23/99 indicated the possibility of one or more failed fuel rods. This analysis also determined that the suspect fuel rod failure(s) might have been in a fuel assembly or fuel assemblies scheduled to be used again in Cycle 16. In order to reduce the possibility of operating Cycle 16 with failed fuel from the Cycle 15 core and to attempt to reduce the overall activity level of the RCS during Cycle 16 a fuel inspection and reconstitution campaign was performed at the end of Cycle 15.

As part of this campaign fuel assemblies were to be examined via an ultrasonic testing (UT) technique and those fuel assemblies scheduled to be used again in Cycle 16 and found with indications of a failed fuel rod(s) were to be reconstituted. Fuel assembly reconstitution includes replacement of failed fuel rods with inert zircaloy clad stainless steel rods in the fuel assembly or if the fuel assembly cage is not useable, recaging the fuel assembly by placing the intact fuel rods in an intact fuel assembly cage. As part of fuel assembly reconstitution certain individual fuel rods were to be examined via high resolution video camera and eddy current testing (ECT) techniques. This safety evaluation documents justification for the temporary placement of irradiated fuel and the use of fuel assembly examination and reconstitution apparatus in the Unit 1 Spent Fuel Pool.

This evaluation concludes that the proposed fuel assembly examination and reconstitution could be performed in the St. Lucie Unit 1 SFP, subject to the constraints described in this evaluation. The proposed activities do not adversely affect plant safety, the performance of refueling activities or the safe operation of the spent fuel pool.

Revision 1 to this safety evaluation documents the justification for placement of up to a maximum of five failed fuel rods in the guide tubes (vs. lattice) of the parent Region S fuel. This evaluation concludes that placement of up to a maximum of five failed fuel rods in the guide tubes of the parent fuel assemblies is acceptable subject to the constraints described in the evaluation.

SAFETY EVALUATION SEMS-99-023
REVISION 1

SAFETY EVALUATION FOR LETTER OF INSTRUCTION PROCEDURE 1-LOI-17.01

Summary:

PC/M 99012 removed Thermo-Lag from the standoffs in Fire Damper Assembly 25-123. During this time, supply fans HVS-5A and HVS-5B needed to be out of service. Letter of Instruction Procedure 1-LOI-17.01 was prepared to outline the method of ventilating the 1A and 1B Battery Rooms, the 1A and 1B Switchgear Rooms, the Cable Spread Room and the Static Inverter Room (the affected rooms) during this modification. Various exhaust fans operate with exterior and interior doors open or ajar to bring cooling air into these rooms.

Revision 1 required that the Battery Room exhaust fans operate at all times and the supply fans are started as soon as possible after the supply duct has been put back in service. This revision and the effect of the changes have no affect on the conclusion of this safety evaluation.

The justification for the temporary ventilation arrangement for the PC/M 99012 modification, as outlined in 1-LOI-17.01, was a result of 'operating experience' achieved through the trial run when the supply fans are in standby and could be immediately started if required. Three additional reasons why this scenario should be acceptable are: 1) the outside temperature during the evening should be 10F less than design, 2) LOI for the temporary ventilation arrangement is not conducted at power, only when the unit is in mode 5, mode 6 or defueled with much of the electrical equipment in these rooms not operating and 3) at the worst case the supply fans could be restarted in approximately two hours if the trend of temperatures for these rooms increases toward the limitation of 104F.

SAFETY EVALUATION SEMS-99-027
REVISION 0

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

Summary:

This safety evaluation compares the assumptions made in the Unit 1 and 2 FSARs for the use of fire protection suppression equipment (i.e., fire hydrants, hose houses, and hose stations) to the fire fighting strategies per Administrative Procedure 1/2-1800023. Differences are evaluated and required changes identified.

SAFETY EVALUATION SENS-99-028
REVISION 1

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

Summary:

The purpose of this safety evaluation was to identify the plant conditions required to safely operate St. Lucie Unit 1 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} less than 1.0) irradiated fuel continues to produce substantial quantities of heat due to the decay of fission products, primarily through the emission of gamma 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 level of decay heat decreases as the time after shutdown increases. The water inventory present in the RCS to absorb decay heat fluctuates during this evolution but it will always be less than the normal operating level. During this evolution, the RCS is drained to approximately the mid-plane of the hot leg piping; this condition is referred to as "mid-loop". More precisely, the plant is considered to be in mid-loop conditions when the reactor vessel water level is below the top of the hot leg and at or above the mid-plane of the hot leg piping. For the purposes of this evaluation, "reduced inventory" was defined as a water level beginning 3 feet below the reactor vessel flange and continuing down to the top of the hot leg. This definition is consistent with that used by the NRC.

In the event of a loss of shutdown cooling while at a reduced inventory or mid-loop condition, the deposited gamma-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 from the loss of inventory. Substantial quantities of steam will be evolved during any inventory boil-off.

FPL 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 is 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.

This safety evaluation effectively amends previously developed FPL guidance on shutdown operations to reflect the use of more conservative, i.e., higher decay heat generation curves for St. Lucie. These changes are permitted using the criteria of 10 CFR 50.59 as outlined in NRC correspondence on the same subject.

Revision 1 to this evaluation was generated to incorporate a revised attachment prepared by ABB-CE. This revised attachment explicitly considered the effect on ICI vent capacity of the installation of upper flange thread protection devices. This attachment concluded that the ICI vent capacity without thread protectors bounds the ICI vent capacity available when thread protection is installed.

SAFETY EVALUATION SEMS-99-030
REVISION 0

UNIT 1 EDG AIR START SYSTEM DESIGN PRESSURE

Summary:

This evaluation reviews the acceptability of revising FSAR information to reflect two pressure regimes within the Unit 1 EDG Air Start System. FSAR Table 9.5-4 indicates a common EDG Air Start System design pressure of 223 psig, whereas originally specified relief valve settings establish a 250 psig design pressure for the Air Compressor Sub-System and a 220 psig design pressure for the Air Accumulator/Delivery Sub-System.

SAFETY EVALUATION 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 FSAR 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 address the identified conditions, and determined there were no operabilitv concerns associated with the apparent discrepancies.

This evaluation 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. Finally, this evaluation supports any changes to procedures 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, and also makes supporting changes to the FSAR.

SAFETY EVALUATION SECS-99-035
REVISION

SAFETY EVALUATION FUEL HANDLING BUILDING SCAFFOLD ERECTION ON SPENT FUEL HANDLING MACHINE

Summary:

The purpose of this evaluation was to assess the adequacy of erecting temporary scaffolding over the Unit 1 Spent Fuel Pool for maintenance activities associated with the Fuel Handling Building HVAC system. The scaffolding was secured atop the Spent Fuel Handling Machine.

Condition Report 99-0116 was generated to document rust chips found on and around the Spent Fuel Handling Machine and floor area about the pool. The suspected source of these rust chips was corrosive attack to the HVAC duct system registers. Similar corrosive attack on the Unit 2 Spent Fuel Pool and new fuel storage area HVAC duct registers was found to be attributable to inadequate coating and lack of maintenance accelerating corrosive attack to these components.

This safety evaluation provided the basis for erecting scaffolding to access the registers and ducts such that the registers may be cleaned and coated and the ducts may be vacuumed out. The scaffolding was erected in the vicinity of and above safety related components, and therefore, it was installed to resist seismic loads.

SAFETY EVALUATION SEMS-99-038
REVISION 1

REVIEW OF UNIT 1 SDC SYSTEM OPERATION

Summary:

Shutdown Cooling 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 reaches a nominal temperature of 135F. The Unit 1 SDC System is described in UFSAR Section 9.3. 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 1 SDC system design and operation including single train operation and single failure. Recommendations were 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 was reviewed to establish RCS level and SDC flow limits to ensure the reliability of SDC.

During development of this evaluation, adverse SDC suction conditions caused by flashing at system high points were determined to be more limiting than NPSH and vortexing at reduced RCS levels. Operating limits designed to preclude loss of SDC due to credible single 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 was the inability to recover SDC at mid-loop conditions once RCS temperature exceeds 190F. If SDC is lost at the pre-fuel shuffle mid-loop condition, there is limited time for the operators to respond (approximately seven minutes). This time may be insufficient for the operators to diagnose and correct the initiating problem and then re-establish SDC at a flow rate adequate to maintain RCS temperature. Failure to re-establish SDC within this time frame will result in containment evacuation due to core boiling. Maintaining the RCS at saturation conditions at reduced inventory precludes restoration of SDC in any reasonable time. One conservative alternative that avoids the increased risk associated with hot mid-loop operation is to perform a full core off-load. If hot mid-loop operation is to be employed, it was recommended that the time spent in this condition be minimized to reduce the overall risk of SDC loss.

Unit 1 UFSAR changes were provided by this Safety Evaluation to align the Unit 1 UFSAR description of SDC operation with updated plant procedures. Procedural guidance and limits were 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 single and dual SDC train operation in a mid-loop configuration.

Revision 1 was issued to address the conditions that would exist during Safeguards Testing performed at lowered RCS levels in Mode 5. Restrictions were imposed for lower RCS temperature and increased RCS level relative to the basis for the flow limits specified in Revision 0 and controlling the system test alignment to preclude adverse LPSI pump suction conditions. These controls allow the performance of integrated safeguards testing at 35 ft. elevation (below the reactor vessel flange) as long as RCS temperature can be maintained below 100F.

SAFETY EVALUATION SEMS-99-042
REVISION 0

CONTAINMENT AIR CONDITIONING DURING REFUELING OUTAGES

Summary:

This evaluation addresses the connection of temporary hoses to lines I-8"-CC-38, 39, 41 & 43 of the Component Cooling Water (CCW) system in any Mode and use of CCW piping and the Containment Fan Coolers (CFC) in Modes 4, 5 and 6 for temporary cooling of the containment, during refueling outages. This safety evaluation specifically reviewed the additional weight added to the CCW piping by the water filled 6" hoses, the Containment Vessel Integrity requirements, the isolation of CCW to the containment fan coolers and using temporary chilled water during Modes 4, 5 and 6.

This evaluation also adds valves SB14517, SB14518, SB14519 and SB14520 as containment boundaries in Table 6.2-16 in an FSAR Change Package. These valves were covered in PC/M 96001 but were inadvertently missed when they should have been added to the FSAR at that time.

SAFETY EVALUATION SEES-99-046
REVISION

TEMPORARY POWER CONNECTIONS TO VITAL BUS NO. 1 POWER DISTRIBUTION PANEL DURING MODES 5 AND 6

Summary:

The Unit 1 design includes a 120 VAC Vital Bus No. 1. This bus provides low voltage electrical power for instrumentation and control for feedwater, circulating water, condensate systems, heater drain systems, turbine generator supervisory, DEH governor control cubicle, excitation cubicle, hydrogen control panel, communications, load frequency control cubicles, etc. Preventative maintenance of the 120 VAC SUPS which powers the plant 120 VAC Vital Bus No. 1 power distribution panel requires de-energizing the vital ac bus.

De-energizing the 120 VAC SUPS causes a loss of power to circuits that are normally fed from the plant 120 VAC Vital Bus No. 1 power distribution panel. Control Wiring Diagrams (CWDs) were marked to identify the devices and annunciation that will be de-energized during the loss of power. Operations personnel reviewed the marked-up CWDs and identified several circuits that can not be de-energized in accordance with plant UFSAR requirements. These circuits can be de-energized for the short period of time that it takes to connect/disconnect a temporary source to the plant 120 VAC Vital Bus No. 1 power distribution panel. The only circuit identified that had a potential change to the UFSAR was the Unit 1 P.A. System. Therefore, this circuit is being addressed by this safety evaluation. The Unit 1 P.A. System power distribution panel is powered from the plant 120 VAC Vital Bus No. 1 power distribution panel and provides power to various plant paging system amplifiers and is used to broadcast various emergency signals, including containment high radiation alarms. Several of these signals are important to plant operation during the modes in which the work will take place. This safety evaluation documents the acceptability of temporary power connections installed in the plant 120 VAC Vital Bus No. 1 power distribution panel to support plant operations during Modes 5 & 6.

The scope of this evaluation is limited to the temporary power connection to the plant 120 VAC Vital Bus No. 1 power distribution panel.

SAFETY EVALUATION SEMS-99-052
REVISION 1

UNIT 1 STEAM GENERATOR SECONDARY SIDE FOREIGN OBJECTS

Summary:

This safety evaluation addresses the safety significance of operating St. Lucie Unit 1 with foreign objects in the secondary side of the steam generators. Unit 1 currently has replacement steam generators manufactured by BWI. The major difference between these steam generators and the original steam generators manufactured by CE is the composition of the heat transfer tubes (Alloy 690 vs. Alloy 600). There is also some difference in the local flow characteristics on the secondary side of the steam generators; however, these flow differences are not significant in the evaluation of potential foreign objects.

This evaluation addresses the potential impact of the following foreign objects in the secondary side of St. Lucie Unit 1 steam generator:

1. A potential foreign object was detected by bobbin coil testing at Row 30 Column 71, approximately two inches above the hot leg tubesheet in steam generator B. Bobbin coil inspection of all adjacent tubes was completed, and a similar indication was reported at the same elevation in Row 31 Column 72. Plus Point rotating probe inspection confirmed the potential for a foreign object between these two tubes. No tube damage was present in any of these inspections, and no evidence of a potential foreign object was detected for the remaining adjacent tubes.

2. One piece of metal wire approximately 0.125 inches in diameter and 2 inches long was located in the cold leg periphery of steam generator B in contact with Row 141 Column 80, Row 140 Column 81, and Row 139 Column 80. No damage to the adjacent steam generator tubes was detected.

3. One piece of metal wire approximately 0.125 inches in diameter and indeterminate length was found extending into the tube bundle between Row 139 Column 76 and Row 138 Column 77 on the cold leg periphery of steam generator B. No damage to the adjacent steam generator tubes was detected.

4. A worker inadvertently dropped a key card, a picture badge, a metal clip and a metal ring inside the secondary side of steam generator B while working on a modification to the manway hinge arm. This evaluation addresses both the mechanical and chemical effects of leaving these items inside steam generator B.

SAFETY EVALUATION SEMS-99-053
REVISION 0

CLARIFICATION OF UFSAR REQUIREMENTS FOR ECCS SUMP OUTER SCREENS

Summary:

The Unit 1 ECCS sump is fitted with two sets of screens, an outer screen composed primarily of ½-inch mesh and redundant inner screens around the inlet to each ECCS recirculation line. The Unit 1 FSAR contains a statement that the cubicle containing the reactor coolant drain tank and the inlets to the recirculation lines is screened at all openings with ½-inch steel mesh. This statement as written can be interpreted as requiring complete enclosure of the drain tank cubicle with steel mesh (i.e., outer screen) with no allowance for gaps or openings in excess of ½-inch. This Safety Evaluation provides clarification to the FSAR to indicate that minor deviations in the outer screens due to structural irregularities and construction tolerances do not impact the required functions of the ECCS sump screens and are therefore acceptable.

The purpose of this evaluation is to clarify the FSAR requirements for the ECCS sump outer screens. The Unit 1 FSAR currently states that the cubicle containing the reactor drain tank and the inlets to the recirculation lines is screened at all openings in the sides and at the top by steel grating and wire mesh to give ½ inch clear openings. Literal interpretation of this statement would preclude even minor openings greater than ½ inch. This Safety Evaluation clarifies the FSAR by adding a statement which allows minor deviations in the outer screens resulting from structural imperfections and construction tolerances.

SAFETY EVALUATION SEMS-99-056
REVISION 1

INSTALLATION OF A TEMPORARY CLAMP ON V3483

Summary:

Condition Report 99-2048 concerns operational leakage discovered in the body to nozzle leak on valve V3483 which is the relief valve for Shutdown Cooling (SDC) loop 1A. This evaluation addresses the temporary clamping of the nozzle to body interface on V3483. An initial visual inspection performed on 10/10/99 indicated that the leak as documented in CR 99-2048 was approximately 5 drops per minute. The leak affects the discharge side of the valve and was considered minor in nature but needed to be stopped to eliminate leakage in the Safeguards Room.

This evaluation demonstrates the acceptability of the addition of a temporary clamp on the body to nozzle interface on V3483. The leak repair enclosure being installed via this safety evaluation was considered as a temporary measure to stop the minor leakage on V3483.

Revision 1 provides for revisions to the clamp required to support field fit-up. The Furmanite calculations and drawing are revised to reflect the field fit-up dimensions. Any additional revisions for field fit-up/installation will be documented by inclusion of a FPL Engineering approved copy of the Furmanite calculations and drawing in the Work Order. This revision does not change the results of the safety evaluation.

SECTION 3

RELOAD SAFETY EVALUATIONS

PLANT CHANGE/MODIFICATION 99016

REVISION 2

ST. LUCIE UNIT 1 CYCLE 16 RELOAD

Summary:

This Engineering Package provides the reload core design of St. Lucie Unit 1 Cycle 16 developed by Florida Power & Light Company and Siemens Power Corporation. The Cycle 16 core is designed for a cycle length of 12,615 EFPH, based on a nominal Cycle 15 length of 14,050 EFPH. The Cycle 16 reload design supports an end-of-cycle Tave coastdown at full power with a maximum reduction in primary coolant temperature of 26F.

The primary design change to the core for Cycle 16 is the replacement of 76 irradiated fuel assemblies with 76 fresh Batch X fuel assemblies and 13 irradiated fuel assemblies with 13 irradiated Batch S (Cycle 13) fuel assemblies currently residing in the spent fuel pool. Cycle 16 core thus will contain 13 irradiated fuel assemblies from Cycle 13 reload, 52 irradiated fuel assemblies from Cycle 14 reload and all 76 fuel assemblies from the Cycle 15 reload. All assemblies in the Cycle 16 reload core are of the debris resistant long end cap design. The mechanical design of Batch X fuel is essentially the same as that of Batches U & T (Cycle 15), Batch T (Cycle 14) and Batch S (Cycle 13) reload fuel. However, the length of axial blankets (U02 rods) and cutback regions (Gadolinia rods) have changed for Batch X. Also, the fuel assembly design for Batch X fuel utilizes radial enrichment zoning.

The implementation instructions provided in this EP, for core reconfiguration from Cycle 15 to Cycle 16, support both an incore fuel shuffle and a full core off-load.

The safety analysis of this design was performed by Siemens Power Corporation (SPC) and by Florida Power and Light Co. (FPL) using NRC approved methodology. The analyses for Cycle 16 support a change to the Core Operating Limits Report (COLR) MTC limit from -28 pcm/F to -32 pcm/F, and a revised fuel centerline melt limit. The revised limit justified for Cycle 16 is 25.1 kw/ft as compared to the current limit of 21 kw/ft. All analyses in support of this EP were performed with the assumption of average steam generator tube plugging level not to exceed 15% with a maximum asymmetry of ±7%. The analyses are consistent with the proposed amendment to the Technical Specifications (TS) reactor coolant system (RCS) design flow (365,000 gpm), RCS flow-low trip setpoint (95% of design flow), COLR methodology list and thyroid dose conversion factors (ICRP-30). It has been determined that the operation of the Cycle 16 reload core does not pose an unreviewed safety question, contingent upon NRC approval of the

above TS changes. This reload can be implemented with no additional changes to the St. Lucie Unit 1 Technical Specifications.

The reconfiguration of the core directly affects its behavior and the capability to assure integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe condition, and the capability to mitigate the consequences of design basis accidents.