St. Lucie Unit 1
Docket No. 50-335
L-98-182
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
JULY 27, 1996 THROUGH JANUARY 7, 1998

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 July 27, 1996, through January 7, 1998. 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 nor require a change to the plant technical specifications; therefore, prior NRC approval was not required for implementation.

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SECTION 1

PLANT CHANGE / MODIFICATIONS

PLANT CHANGE/MODIFICATION 85010

STEAM TRAP DRAIN PIPING - AS FAIL REPLACEMENT

Summary:

Since startup, leakage problems have been experienced in steam line moisture trap drains, piping and fittings. The cause of this leakage is impinging two-phase flow at the elbows, downstream of the moisture traps. This modification provided the details and instructions for the replacement of eroded carbon steel piping and fittings on an "as-fail" basis. The new materials specified, ASTM A-182 F22 (fittings) and ASTM A-335 P22 (piping) are chromium-molybdenum alloys which will provide superior resistance to corrosion-erosion effects. The new materials can also be welded into the existing A-105 and A-106 piping and fittings.

The steam trap drains are non-safety and non-seismic and are subject to the requirements of ANSI B31.1. The new piping runs are geometrically identical to those replaced and piping and fittings specified satisfy ANSI B31.1 requirements.

PLANT CHANGE/MODIFICATION 89193

CHANGE OUT OF 3" PVC VACUUM BREAKER VALVES ON OPEN BLOWDOWN COOLING SYSTEM (OBCW)

Summary:

This is a design equivalent replacement modification which removed the two 3-inch PVC insert-type vacuum breaker check valves on the open blowdown cooling water (OBCW) system and replaced them with 3- inch stainless steel valves. The change was made because the PVC valves became warped, leaked and were high maintenance items.

The OBCW system is non-safety and non-seismic. The replacement valves were evaluated for form, fit and function and were determined to be acceptable.

PLANT CHANGE/MODIFICATION 92203

REPLACEMENT OF SIGMA METERS

Summary:

This modification replaced Sigma 9220 and 9260 series indicators, including indicators installed in the control room and on the hot shutdown panel. The new indicators are Versatile Measuring Instruments (VMI) series 2000 bargraph/digital indicators.

Sigma 9220 and 9260 series indicators have been discontinued for some time and spares were no longer available in plant stores. Although Sigma sold their rights to VMI, spares were not available as Class 1E instruments. The replacement VMI series 2000 indicators are state-of-the-art bargraph/digital indicators with digital readout and improved readability and accuracy. These replacement indicators have been provided as Class 1E through ABB Combustion Engineering nuclear spare parts.

PLANT CHANGE/MODIFICATION 93151

SG 1A REPLACEMENT MODIFICATION

Summary:

This modification provided for the replacement of steam generator (SG) 1A with a replacement steam generator (RSG) designed and fabricated by Babcock and Wilcox International (BWI). SG 1B was also replaced during the same refueling outage. The original SGs were replaced due to corrosion of the tubes which has resulted in tube plugging and increased maintenance requirements.

The RSGs have been evaluated, under separate cover, in the Steam Generator Equivalency Report (SGER). The SGER describes the differences between the replacement and original steam generators and documents the acceptability of the replacement steam generators and concludes that "Evaluation of the existing plant design and licensing bases, steam generator physical interfaces, thermal-hydraulic aspects, structural supports, safety analyses, technical specifications, and off-normal and emergency procedures showed the RSG design to preserve the existing plant design and licensing bases."

The RSGs were designed, manufactured and tested in accordance with the 1986 Edition of the ASME Boiler and Pressure Vessel Code and are equivalent in form, fit and function to the original SGs.

In addition to the replacement of SG 1A, this package provided for the modification of connected instrument lines and the loose parts monitoring accelerometers for proper fitup to the RSG. Contingencies were also provided for the alignment of reactor coolant, main steam, main feedwater and blowdown system piping to the RSG.

PLANT CHANGE/MODIFICATION 93152

SG 1B REPLACEMENT MODIFICATION

Summary:

This modification provided for the replacement of steam generator (SG) 1B with a replacement steam generator (RSG) designed and fabricated by Babcock and Wilcox International (BWI). SG 1A was also replaced during the same refueling outage. The original SGs were replaced due to corrosion of the tubes which has resulted in tube plugging and increased maintenance requirements.

The RSGs have been evaluated, under separate cover, in the Steam Generator Equivalency Report (SGER). The SGER describes the differences between the replacement and original steam generators and documents the acceptability of the replacement steam generators and concludes that "Evaluation of the existing plant design and licensing bases, steam generator physical interfaces, thermal-hydraulic aspects, structural supports, safety analyses, technical specifications, and off-normal and emergency procedures showed the RSG design to preserve the existing plant design and licensing bases."

The RSGs were designed, manufactured and tested in accordance with the 1986 Edition of the ASME Boiler and Pressure Vessel Code and are equivalent in form, fit and function to the original SGs.

In addition to the replacement of SG 1B, this package provided for the modification of connected instrument lines and the loose parts monitoring accelerometers for proper fitup to the RSG. Contingencies were also provided for the alignment of reactor coolant, main steam, main feedwater and blowdown system piping to the RSG.

PLANT CHANGE/MODIFICATION 93153

SG1A MANWAY PLATFORM MODIFICATION

Summary:

This modification is associated with the 1997 steam generator replacement outage. Replacement steam generator (RSG) 1A is designed with the primary manway openings at 45 degrees from horizontal (versus 22.5 degrees) and at an elevation which is 27" lower than the original SG manway. As such, the manway platform required modification.

The platform itself is non-safety; however, it is seismically designed so as to avoid interactions with adjacent safety related equipment. The new platform provides an increase in work/laydown area. Two 3/4" reactor coolant system lines were rerouted and associated supports/restraints redesigned in order to eliminate interferences with the new platform. Also, a column support was redesigned to eliminate interferences with the new design.

PLANT CHANGE/MODIFICATION 93154

SG 1B MANWAY PLATFORM MODIFICATION

Summary:

This modification is associated with the 1997 steam generator replacement outage. Replacement steam generator (RSG) 1B is designed with the primary manway openings at 45 degrees from horizontal (versus 22.5 degrees) and at an elevation which is 27" lower than the original SG manway. As such, the manway platform required modification.

The platform itself is non-safety; however, it is seismically designed so as to avoid interactions with adjacent safety related equipment. The new platform provides an increase in work/laydown area. A conduit support/restraint has been redesigned and valve stem leakoff tubing for safety injection valves V3651 and V3652 have been capped closer to their drain collection header to eliminate interferences (note: elimination of valve stem leakoff lines was evaluated and authorized via PC/M 95144 and SPEC-M-037).

PLANT CHANGE/MODIFICATION 93155

SG 1A BLOWDOWN MODIFICATION

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement steam generators (RSGs) have two blowdown nozzles versus the single nozzle of the original steam generators (OSGs). Also, the RSG blowdown nozzles are located approximately 21" lower than the OSG nozzle.

This modification provided for the installation of new piping to tee together the nozzle discharge piping and to install a 4"x2" eccentric reducer at the nozzles since the RSG nozzles are 4" NPS schedule 160 and the attached piping is 2" NPS schedule 80. The manual isolation valve (V23100) has been replaced and relocated just outside the secondary shield wall. Valve V23102, a globe valve, has been replaced with a gate valve to reduce the pressure drop across the valve.

New piping has been fabricated to ASME Section III, Subsection NC, 1989 Edition (no addenda). A stress analysis for the piping design was performed in accordance with Subarticle NC-3600 of ASME Section III, Subsection NC, 1971 Edition, Summer 1973 Addenda (which is equivalent to the original USAS B31.7-1969 Nuclear Power Piping Code as documented in the Code Reconciliation). Postulated pipe breaks are based on NRC Generic Letter 87-11 as incorporated in the UFSAR. New manual isolation valves were designed and fabricated in accordance with ASME Section III, Subsection NC, 1989 Edition (no addenda).

PLANT CHANGE/MODIFICATION 93156

SG 1B BLOWDOWN MODIFICATION

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement steam generators (RSGs) have two blowdown nozzles versus the single nozzle of the original steam generators (OSGs). Also, the RSG blowdown nozzles are located approximately 21" lower than the OSG nozzle.

This modification provided for the installation of new piping to tee together the nozzle discharge piping and to install a 4"x2" eccentric reducer at the nozzles since the RSG nozzles are 4" NPS schedule 160 and the attached piping is 2" NPS schedule 80. The manual isolation valve (V23100) has been replaced and relocated just outside the secondary shield wall. Valve V23102, a globe valve, has been replaced with a gate valve to reduce the pressure drop across the valve.

New piping has been fabricated to ASME Section III, Subsection NC, 1989 Edition (no addenda). A stress analysis for the piping design was performed in accordance with Subarticle NC-3600 of ASME Section III, Subsection NC, 1971 Edition, Summer 1973 Addenda (which is equivalent to the original USAS B31.7-1969 Nuclear Power Piping Code as documented in the Code Reconciliation). Postulated pipe breaks are based on NRC Generic Letter 87-11 as incorporated in the UFSAR. New manual isolation valves were designed and fabricated in accordance with ASME Section III, Subsection NC, 1989 Edition (no addenda).

PLANT CHANGE/MODIFICATION 93157

SG 1A INSULATION MODIFICATION

Summary:

This modification is associated with the 1997 steam generator replacement outage. The existing reflective and encapsulated mineral wool insulation has been replaced with THERMAL-WRAP blanket insulation. This package also provides the basis for the replacement of insulation on other piping systems where existing insulation has become damaged and it is desired to replace insulation materials with THERMAL-WRAP blanket insulation.

The shell configuration of the replacement steam generators (RSGs) is modified from that of the original steam generators (OSGs) such that existing insulation must be replaced. It was also determined that existing reflective insulation on the RCS loop 1A hot leg elbow must also be replaced in order to provide an acceptable fitup between the RSG insulation and the hot leg piping insulation. Blanket insulation was selected due to its improved thermal performance, ease of handling and cost-effectiveness. Unit 2 also uses blanket insulation, as has other utilities during recent steam generator replacements. Four-inch thick blankets are used for the steam generator and three-inch thick blankets used for the hot leg elbow. The blanket insulation will limit the maximum heat flux to 65 Btu/hr/ft^2. Stainless steel jacketing covers the insulation to provide additional structural support and protection from damage. Blankets are held in place on the RSG by use of friction attachments.

The new insulation is compatible with a post-LOCA containment environment and has been seismically qualified.

PLANT CHANGE/MODIFICATION 93158

SG 1B INSULATION MODIFICATION

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This modification is associated with the 1997 steam generator replacement outage. The existing reflective and encapsulated mineral wool insulation has been replaced with THERMAL-WRAP blanket insulation. This package also provides the basis for the replacement of insulation on other piping systems where existing insulation has become damaged and it is desired to replace insulation materials with THERMAL-WRAP blanket insulation.

The shell configuration of the replacement steam generators (RSGs) is modified from that of the original steam generators (OSGs) such that existing insulation must be replaced. It was also determined that existing reflective insulation on the RCS loop 1B hot leg elbow must also be replaced in order to provide an acceptable fitup between the RSG insulation and the hot leg piping insulation. Blanket insulation was selected due to its improved thermal performance, ease of handling and cost-effectiveness. Unit 2 also uses blanket insulation, as has other utilities during recent steam generator replacements. Four-inch thick blankets are used for the steam generator and three-inch thick blankets used for the hot leg elbow. The blanket insulation will limit the maximum heat flux to 65 $Btu/hr/ft^2$. Stainless steel jacketing covers the insulation to provide additional structural support and protection from damage. Blankets are held in place on the RSG by use of friction attachments.

The new insulation is compatible with a post-LOCA containment environment and has been seismically qualified.

PLANT CHANGE/MODIFICATION 93194

STEAM GENERATOR 1A MAIN STEAM RUPTURE RESTRAINTS

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement of steam generator (SG) 1A required the original steam generator (OSG) and replacement steam generator (RSG) to be moved directly above the SG 1A cubicle and the tower frame. The existing main steam rupture restraint (MSRR) and a section of main steam piping cause an interference with the replacement process. The process required the main steam piping to be severed at the nozzle weld connection to the SG and at the downstream end of the second elbow from the nozzle. Portions of restraint collars RE-MS-11 and RE-MS-14 were permanently removed. Reinstallation of the restraint collars is not required per NRC Generic Letter 87-11 as incorporated in the UFSAR. Technical justification for permanent deletion of the rupture restraints is documented in calculation PSL-1MHC-94-007.

Also included in the modification is a redesign of feedwater support BFH-81, the redesign of the equipment laydown area support steel, MSRR lifting lugs, and replacement of anchor bolting for a portion of the connections attaching the MSRR structure to the tower frame.

PLANT CHANGE/MODIFICATION 93195

STEAM GENERATOR 1B MAIN STEAM RUPTURE RESTRAINTS

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement of steam generator (SG) 1B required the original steam generator (OSG) and replacement steam generator (RSG) to be moved directly above the SG 1B cubicle and the tower frame. The existing main steam rupture restraint (MSRR) and a section of main steam piping cause an interference with the replacement process. The process required the main steam piping to be severed at the nozzle weld connection to the SG and at the downstream end of the second elbow from the nozzle. Portions of restraint collars RE-MS-1 and RE-MS-4 were permanently removed. Reinstallation of the restraint collars is not required per NRC Generic Letter 87-11 as incorporated in the UFSAR. Technical justification for permanent deletion of the rupture restraints is documented in calculation PSL-1MHC-94-008.

Also included in the modification is a redesign of feedwater support BFH-76A, the redesign of the equipment laydown area support steel, MSRR lifting lugs, and replacement of anchor bolting for a portion of the connections attaching the MSRR structure to the tower frame.

PLANT CHANGE/MODIFICATION 93207

SURGE LINE RUPTURE RESTRAINTS RC-30 & RC-31 MODIFICATION

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement of steam generator 1B required the original steam generator hot leg nozzle weld to be severed. The design of the tooling for cutting and welding at the hot leg nozzle required clearance for proper mounting and operation. Surge line rupture restraints RC-30 and RC-31 have been permanently modified, with a small portion permanently cut away from each restraint, in order to provide the necessary clearance for the above tooling. This redesign is considered minor. The change was analyzed (via separate calculation) to demonstrate the capacity of the modified restraints exceeds design requirements.

PLANT CHANGE/MODIFICATION 94067

WIDE RANGE S/G LEVEL ADDITION - PHASE 2 (LT-9014 & LT-9024)

Summary:

NRC Regulatory Guide (RG) 1.97 requires that wide range (WR) steam generator (SG) level instrumentation be available following an accident. This instrumentation is defined by the RG as a Category 1, Type D variable. The original design of the plant for WR SG level included one instrument per SG. The WR level transmitters (LT-9012 & LT-9022) were not qualified for a post-accident environment.

St. Lucie committed to upgrading the environmental qualification of the existing WR instrumentation and adding redundant WR level indication in lieu of meeting all the requirements for a RG 1.97 Category 1, Type D variable. A two phased approach was taken to implement the changes. Phase 1 upgraded the environmental qualification of the existing instrumentation and was completed via Plant Change/Modification (PC/M) 94012. Phase 2, addition of redundant instrumentation, was completed via this PC/M.

This modification adds WR SG level instruments LT-9014 & LT-9024 (one instrument per SG). The new instruments have been designed as safety related, environmentally qualified components. The new instrument sensing lines are designed to B31.7 Class 2 requirements.

PLANT CHANGE/MODIFICATION 94070

STEAM GENERATOR 1A SUPPORTS

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement of steam generator (SG) 1A required the following: 1) unbolting the original steam generator from its sliding base support and removal of the bent shim plate; 2) removal of interferences with the SG upper lateral support by the removal of a portion of the support's steel on the North and South sides of the SG (evaluated in calculation PSL-1MHC-94-014); and 3) the temporary removal of the SG snubbers and portions of the related hydraulic tubing.

Design details were provided for obtaining and verifying the cold gap clearances for the sliding base support, upper lateral support and snubbers for the replacement steam generator once installed. In addition, post-modification testing required determination of actual movements with the plant in Mode 3.

PLANT CHANGE/MODIFICATION 94071

STEAM GENERATOR 1B SUPPORTS

Summary:

This modification is associated with the 1997 steam generator replacement outage. The replacement of steam generator (SG) 1B required the following: 1) unbolting the original steam generator from its sliding base support and removal of the bent shim plate; 2) removal of interferences with the SG upper lateral support by the removal of a portion of the support's steel on the North and South sides of the SG (evaluated in calculation PSL-1MHC-94-014); and 3) the temporary removal of the SG snubbers and portions of the related hydraulic tubing.

Design details were provided for obtaining and verifying the cold gap clearances for the sliding base support, upper lateral support and snubbers for the replacement steam generator once installed. In addition, post-modification testing required determination of actual movements with the plant in Mode 3.

PLANT CHANGE/MODIFICATION 95009

REPLACEMENT OF THE EXCORE NEUTRON FLUX MONITORING AND PROTECTIVE SYSTEM (NI DRAWERS) FOR THE RPS SYSTEM

Summary:

This modification provided for the following: 1) four existing Gamma-Metric wide range' excore detectors and associated cables to be replaced with improved Gamma-Metrics wide range detectors and detector cables; 2) four existing amplifiers to be replaced with Gamma-Metrics amplifier assemblies; and 3) eight existing reactor protection system nuclear instrumentation drawers to be replaced with new Gamma-Metrics nuclear instrumentation (NI) drawers (analog display). This modification was necessary due to equipment aging and obsolescence, as well as operations with a low leakage core design. There is no change being made to the control room indicators themselves.

The new detectors are of the same configuration; however, the design has been enhanced for better reliability and sensitivity. The design used is the same as is currently used for the Appendix "R". detectors and in Unit 2. The new detectors are installed in the existing detector holder tubes, are seismically qualified and have a qualified life of 40 years plus LOCA.

The new amplifiers are installed in the same location as the existing amplifiers. The assembly is qualified to IEEE 323-1971 and is environmentally and seismically qualified.

The new NI drawers are like for like replacements of the old drawers and have been qualified to the same seismic and environmental requirements as the original.

PLANT CHANGE/MODIFICATION 95021

CONDENSING UNITS FOR CONTROL ROOM VENTILATION SYSTEM

Summary:

This modification replaces the outdoor condensers of the three safety related control room air conditioning units. The existing units were in a deteriorated condition and spare parts were not available. The replacement units have been procured as commercial equipment and were dedicated for safety related service through testing and evaluations. The units meet the requirements of FPL Engineering Specification SPEC-M-028. Testing consisted of seismic, hurricane wind and functional tests. Each of the units is capable of providing 50% of the normal cooling load and 100% of the emergency cooling load (assuming simultaneous operation of all necessary equipment).

PLANT CHANGE/MODIFICATION 95037

DEBRIS FILTER AND CONDENSER TUBE CLEANING SYSTEM INSTALLATION

Summary:

FPL performed a study in 1994 in order to improve the hydraulic and thermal performance of the non-safety circulating water system and main condenser. The study concluded that the addition of a debris filter system (DFS) and condenser tube cleaning system (CTCS) provided the most advantageous technical and economical solution. Taprogge America Corporation was contracted to provide the major system components and to install components in the main circulating water piping. This modification provided the requirements for installation of the DFS and CTCS.

The DFS is intended to reduce the macro fouling potential on each of the four condenser inlet lines. The debris collected in the DFS is periodically back-flushed to a connection located on the CTCS at the circulating water outlet. The strainer consists of four 84" diameter debris filter spool sections located on each condenser water box inlet. A 14 inch fiberglass debris filter discharge line directs the backwash to the condenser outlet. Backwashing is accomplished automatically, based on differential pressure. High differential pressure is alarmed locally and in the control room. A DFS control panel is provided for each of the four debris filter systems.

The CTCS cleans the condenser tubes by injecting cleaning balls into the condenser inlet. The balls pass through the condenser tubes and are caught by strainer screens located at the condenser outlet. The balls are then removed from the discharge via the ball extraction system and are pumped to a ball collector back at the inlet. When the system is in operation the balls pass through the collector and are injected into the condenser inlet via injection nozzles. When the system is secured, the balls are simply collected and held in the collector until system use.

This modification does not impact the ability of the circulating water system to maintain design flow.

PLANT CHANGE/MODIFICATION 95040

1A STEAM TRESTLE STRUCTURAL MODIFICATION FOR SGRP

Summary:

This modification is associated with the 1997 steam generator replacement outage. A temporary gantry (TG) was installed just outside the Unit 1 reactor containment building (RCB), adjacent to the construction hatch, for the purpose of moving the steam generators in and out of containment. The layout of the TG was such that it required two masts to be located as close as possible to the RCB. One mast was supported using a conventional reinforced concrete footing; however due to the proximity of the steam trestle, the other mast could not be conventionally supported. As such, this package modified the structural steel framework of the main steam trestle (MST) to serve as the mast support.

The modification consisted of structural steel shapes and plates welded and/or bolted to the structural steel framework of the MST. Baseplates were also temporarily attached to the RCB utilizing concrete expansion anchors. Each of two support beams was attached on top of two existing MST columns. The ends of the support beam cantilever slightly over the south-eastern edge of the MST. A pair of mast beams spanned the support beams. Two large plates were attached to the mast beams to act as the baseplates for the bottom frame of the TG mast. Additional stiffening members were also added to the MST to carry lateral loads and to prevent buckling. A pair of struts with anchored baseplates was used to attach the mast beams to the RCB wall.

The support structure has been designed an analyzed for TG loads during the steam generator replacement outage. After the outage, the only additional loads to the MST are the dead loads associated with the installed/remaining portions of the support structure. The baseplate anchored to the RCB was removed and the wall restored.

PLANT CHANGE/MODIFICATION 95115

SAFETY RELIEF VALVE V3412 SETPOINT AND BLOWDOWN MOD

Summary:

Valve V3412 is a thermal relief valve located in the high pressure safety injection (HPSI) system downstream of the HPSI pumps. This modification replaces the valve in order to increase the set point from 1735 psig to 1750 psig and decrease the blowdown from 25% to 10%. The reason for this change is to provide a positive reseat margin for the valve. Due to tolerances in the lift set point and the higher HPSI system pressure when operating in the piggy back configuration (i.e., when aligned in series with the containment spray pumps), the valve could remain open if actuated due to a large, unanticipated pressure transient. This modification eliminates the negative operating margin of the valve, allowing it to reseat under the extreme conditions described above.

The replacement valve meets ASME Section III, Class 2 requirements and the higher set point has been satisfactorily evaluated against system design pressures.

PLANT CHANGE/MODIFICATION 96001

CONTAINMENT AIR CONDITIONING FOR REFUELING OUTAGES

Summary:

This modification provided the engineering and hardware to adapt the component cooling water (CCW) system to be quickly and easily connected to a temporary, non-safety, chilled water system that will provide cooling water for the containment fan coolers (CFCs) during refueling outages whenever the unit is in Mode 5 or 6. The safety related CFCs were designed to provide containment atmosphere cooling under normal operating and post-accident conditions; however, the CFCs are not required to be operable in Modes 5 and 6 when the chilled water system would be used.

This modification added permanent valves on the CCW inlet and outlet lines for the CFCs. Four 8"x·8"x 6" tees, each with 6" wafer valves have also been added in order to provide for the hose connections to the temporary chilled water system. Temporary hoses will be used for the supply and return of chilled water between the CCW system and the portable chilled water units which will be located outdoors. All hose fittings for the system are 6" quick disconnect fittings which will be secured with blank, pressure rated covers when not in service in order to maintain cleanliness and to provide a backup pressure boundary. Two wall penetrations (with installed piping and quick disconnects) were added south of the H&V area on the north side of the reactor auxiliary building pipe penetration room.

When the system is operational during an outage, the containment portion of the CCW system will be isolated from the remainder of the system by closing valves MV-14-5, -6, -7, and -8. Although the CFCs will be operated at a lower temperature than normal (when operating the chilled water system), the analysis associated with this modification concluded there are no adverse affects on plant safety or operation.

PLANT CHANGE/MODIFICATION 96080

INSTALLATION OF VENT VALVES FOR THE SDC HEAT EXCHANGERS

Summary:

This modification installed vent valves in the channel vent lines of the shutdown cooling (SDC) heat exchangers. The intent of the channel vent lines is to provide for the venting of the tube side of the heat exchangers. The vent valves are stainless steel, ASME Section II, Class 2 (or better) rated to ANSI 1878 pressure class and are appropriate for installation in the USAS B31.7, Class II system.

PLANT CHANGE/MODIFICATION 97008

1A MAIN TRANSFORMER REPLACEMENT

Summary:

This modification replaced the 1A main transformer due to degradation of its electrical core affecting its operation. Oil analysis of the transformer indicated that undesirable concentrations of combustible gases were accumulating in the transformer. Substation Engineering and Systems Components Engineering evaluation of the oil samples resulted in a recommendation to replace the transformer.

The main transformer serves to raise the voltage of the main generator output for efficient transmission of power on the FPL transmission system. The replacement transformer was purchased from ABB under FPL transformer specification 4.1-1 as a replacement for the 1A transformer. The main transformer is a non-safety component.

As a part of this modification, associated fire detection conduit and fire suppression piping were removed and replaced in their original configuration. The non-safety insulator wash system was removed and not replaced. This modification also provided for the replacement of the 1B main transformer cooling fan plugs/receptacles with a terminal block in order to eliminate grounding problems during heavy rains.

PLANT CHANGE/MODIFICATION 97009

PRESSURIZER HEATER SLEEVE NICKEL PLATING

Summary:

This modification provided for the nickel plating of the pressurizer heater sleeves to address primary water stress corrosion cracking (PWSCC) of the Inconel Alloy 600 sleeves. Nickel plating is in response to numerous failures of pressurizer heater sleeves of similar design and material condition at Calvert Cliffs. The nickel plating is 0.010 inches thick and extends no greater then ¼-inches from the top end of the sleeve downward approximately 4-inches. Nickel plating has been used successfully by Frameatome on steam generator tubes since 1985 and on pressurizer heater sleeves at Calvert Cliffs Unit 1 since 1994.

As a result of the restriction of the internal diameter of the heater sleeves (0.905" to 0.885"), the heaters were also replaced. The heater sheaths were changed from Inconel Alloy 600 to 316 stainless steel as a result of sheath failures in the industry. The pressurizer heater cut-off set point was also increased to ensure the heaters are de-energized when uncovered.

Pursuant to the St. Lucie Unit 1 Environmental Protection Plan, Appendix B to the Facility Technical Specifications, an evaluation was performed to determine if any Unreviewed Environmental Question exists as a result of the nickel plating and associated waste treatment processes. This evaluation concluded that no Unreviewed Environmental Question exists since there would be no measurable non-radiological effects to any areas of the site that have not been previously disturbed during site preparation and construction.

PLANT CHANGE/MODIFICATION 97016

SET POINT CHANGES FOR INSTRUMENT AIR COMPRESSORS 1A AND 1B

Summary:

This modification revised the operating set points for non-safety instrument air compressors 1A and 1B. Previously, the set points were 92/98 psig for Full Load/Unload configuration and 88/96 psig for Auto Start/Reset. 1-LOI-T-71 temporarily revised the above set points to 110/115 psig and 100/107 psig respectively. That temporary change was noted to have improved system performance by allowing the compressors to run loaded during surveillance testing and it improved the availability of the air supply in the event of a loss of offsite power; it also provided an adequate supply of compressed air at the desired pressure. Based on the success of the temporary change, this modification was issued to make the change permanent.

PLANT CHANGE/MODIFICATION 97032

GL 96-06 THERMAL PRESSURIZATION RELIEF VALVES

Summary:

One of the issues identified in NRC Generic Letter (GL) 96-06 was a potential for thermally induced pressurization due to heating of water filled isolated sections of piping. Of specific concern was the potential for thermal pressurization to jeopardize the containment boundary. A review of Unit 1 containment penetrations identified three locations which warranted design enhancement. Per FPL's commitment in response to the GL, this modification installs a thermal relief valve on each of the affected penetrations.

The affected penetrations are P-46 & P-47, refueling cavity pool purification supply and return lines, and P-42, discharge line for the reactor cavity sump pumps. P-46 and P-47 are classified as Quality Group B and were designed in accordance with USAS B31.1, Class 2, 1969 Edition. P42 is classified as Quality Group D and was designed to USAS B31.1, 1967 Edition. The relief valves were designed and built to ASME Code Section III in accordance with Engineering specified requirements.

A set point of 175 psig was selected in order to simplify procurement and to provide interchangeability. The 175 psig set point assures that containment isolation is not jeopardized and it provides ample margin above maximum expected system operating pressures. Likewise, the selected blowdown of 20% provides adequate margin above normal operating and expected transient conditions. A 5 gpm relief capacity was determined to bound the requirements of the most limiting installation.

PLANT CHANGE/MODIFICATION 97034

CONDUIT REROUTES FOR THERMO-LAG REDUCTION

Summary:

As a result of NRC and industry concerns associated with Thermo-lag, Engineering has performed individual fire area evaluations to determine the options available to eliminate the need for protection of affected conduits. Possible options include conduit reroutes, redundant equipment or manual actions. This modification implements the following:

1) relocates conduits previously protected with Thermo-lag 330-1 fire barriers routed in Fire Zone A-77, Fire Zone C-55W and Fire Zone C-78;

2) relocates cables for charging pump 1B in Fire Zone N-36A and N-75; and

3) revises the safe shutdown analysis to reflect updated information for conduits previously protected for safe shutdown capabilities.

The above modifications impact cables associated with safety related battery 1A, charging pump 1B, the PORVs and equipment powered from 120 Vac instrument bus 1MA. The rerouting of affected conduits has no net affect on the design or operation of any of the related components.

PLANT CHANGE/MODIFICATION 97039

REPLACEMENT OF RADIANT ENERGY SHIELDS

Summary:

Radiant energy shields are used within containment to prevent the radiant heat associated with a fire in one safety train from affecting the redundant equipment in the opposite train. These shields or barriers are constructed of flat Thermo-lag 330-1 panels or marinite boards covered with stainless steel. As a result of NRC and industry concerns associated with the use of Thermo-lag, it was decided to replace the radiant energy shields constructed of that material. As such, this modification affected only the Thermo-lag panels.

The Thermo-lag shields were replaced with 16 gauge stainless steel sheet metal. The stainless steel sheet metal was desirable because it can be formed to fit and then secured by bolting into the same structural members which held the Thermo-lag shielding. The new shields have been seismically supported to prevent interaction with adjacent safety related equipment.

PLANT CHANGE/MODIFICATION 97044

QUICKLOC INCORE INSTRUMENT (ICI) FLANGE DESIGN

Summary:

This modification replaced the existing reactor instrument nozzle penetration and sealing system with the ABB-CENO designed Quickloc system. The Quickloc design reduces the disconnection and reconnection requirements for the incore instrumentation assemblies whenever the reactor vessel closure head is removed. The Quickloc design also reduces time requirements and radiation exposure by reducing the nozzle flange disassembly and re-assembly complexity.

The original ICI flange design required the disassembly/re-assembly of 4 Grayloc stud and nut sets and 6 castle nuts (including torquing requirements) per flange every outage. Quickloc reduces this to one large nut and a new seal assembly between the new hub and the new ICI stalk assembly.

This package also replaced the 45 incore instrument assemblies with functionally equivalent assemblies, provided by ABB-CENO, which are compatible with the Quickloc design. This modification has no adverse impact on the safety related functions of the reactor vessel closure head instrument nozzle primary pressure boundary, the core exit thermocouple system or the incore detectors.

PLANT CHANGE/MODIFICATION 97050

REACTOR CAVITY SUMP SWITCH REPLACEMENT (LS-07-12)

Summary:

This modification provides a replacement for the reactor cavity sump weir tank ultrasonic level switch, LS-07-12. LS-07-12 is a part of the reactor coolant pressure boundary leakage detection system which senses the liquid level of the V-notched weir tank. Since weir tank level is a function of the leakage rate, the level sensor is used to annunciate a 1 gpm leak rate.

All reactor cavity drains pass through the weir tank causing dirt, oils and other contaminants to collect on the sensor, resulting in sensor failure. This modification replaced the ultrasonic device with a float type level switch. The new float switch design increases the reliability and reduces the maintenance (and associated radiation exposure) of the level instrument. The replacement switch has been seismically qualified.

PLANT CHANGE/MODIFICATION 97052

ADDITION OF EXCESS FLOW ISOLATION VALVE IN THE H2 PIPE IN THE RAB

Summary:

This modification installed an excess flow isolation valve in the non-safety, non-seismic hydrogen line which enters the reactor auxiliary building and also added an orifice immediately downstream of the volume control tank (VCT) hydrogen pressure regulator. This change brings plant configuration into conformance with the requirements of the UFSAR. UFSAR Appendix 9.5A requires hydrogen lines in safety related areas to be either seismically designed, or installed within a vented guard pipe, or equipped with excess flow valves such that, in the event of a line break, the hydrogen concentration in the affected area would not exceed 2%. There is no detrimental affect on the ability to supply hydrogen to the VCT.

PLANT CHANGE/MODIFICATION 97065

INSTALLATION OF TWO OPTIMIZED PROPORTIONAL AXIAL REGION SIGNAL SEPARATION EXTENDED LIFE (OPARSSEL) DEMONSTRATION IN-CORE DETECTORS

Summary:

This modification installed two OPARSSEL in-core detectors. The two new detectors replaced incore detectors and core exit thermocouples (CETs) at core locations W-16 and B-5.

The original self-powered neutron detector design (45 core locations) used rhodium-103 as the active neutron flux sensitive element. For Cycle 14 operation, one fixed incore detector assembly was removed from service and replaced with a demonstration PARSSEL assembly (B-5 location). The OPARSSEL assemblies installed by this package use a full core length vanadium wire as the neutron sensitive element. The advantage of using vanadium is its non-depleting aspect since its cross section is approximately 5% that of rhodium. The CETs incorporated into the OPARSSEL detectors are Type K chromel-alumel as in the previous design.

The two OPARSSEL detectors are a part of a demonstration project to evaluate the possibility of replacing the entire set of rhodium detectors with the OPARSSEL design. As such, the new OPARSSEL detectors are not considered operable and the appropriate point IDs have been deleted from the incore monitoring system to preclude their use in power distribution measurement and linear heat rate alarm. Qualification of the OPARSSEL detector assembly will be addressed at a later date in a separate modification package.

PLANT CHANGE/MODIFICATION 97069

MISCELLANEOUS GENERIC LETTER 89-10 MOV MODIFICATIONS

Summary:

NRC Generic Letter 89-10 required operating nuclear plants to develop and implement a program to ensure that control switch settings on all safety related motor operated valves (MOVs) are correctly selected, set and maintained to accommodate the maximum system parameters expected during all postulated events within the plants' design basis. Item "a" of the Generic Letter required the design basis for these MOVs to be reviewed to determine maximum system parameters (e.g., differential pressure, line pressure, flow). This has been completed and documented for the valves in this modification. Item "b" of the Generic Letter required licensees to establish the correct MOV switch settings based on the previously determined maximum differential pressure.

Recent industry experience has shown that more restrictive efficiencies should be used for the calculation of actuator capability for the opening and closing strokes. This required some valves to be modified to increase actuator capability to perform their intended functions at reduced voltages or to provide sufficient margin to account for diagnostic equipment inaccuracies, torque switch repeatability and various operator efficiencies. This package modified valves affected by these considerations. Modifications consisted of one or more of the following: gearing changes, bypass of the close torque switches, installation of spring cap vents, installation of new spring packs, adjustment of torque switch settings based upon revised thrust values, and replacement of thermal overload heaters.

The following valves were affected by this modification: PORV block valves V1403 & V1405; MSIV bypass valves MV-08-1A & -1B; shutdown cooling heat exchanger inlet isolation valves V3452 & V3453 and outlet isolation valves V3456 & V3457; boric acid gravity feed isolation valves V2508 & V2509; HPSI flow control valves HCV-3616 & HCV-3646; and HPSI header flow control valves HCV-3617 & HCV-3647.

PLANT CHANGE/MODIFICATION 97073

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

Summary:

This modification allows replacement of the shield building component cooling water (CCW) piping penetration secondary bellows assemblies and the installation of anti-sweat insulation for CCW process piping between the reactor auxiliary building and the reactor containment building. The formation of condensation inside the bellows assemblies has resulted in standing water and corrosion of the carbon steel components of the penetrations and the associated CCW piping. Access to the piping and internal areas of the penetrations requires removal of the secondary bellows from the CCW penetration assemblies.

The shield building secondary bellows are classified as Quality Group C, seismic, and were designed in accordance with USAS B31.7, Class 3, 1969 Edition. The replacement bellows have been designed in accordance with the Expansion Joint Manufacturer's Association (EJMA) standard and manufactured under ASME Section VIII by the current design holder (Pathway). All materials were procured from an ASME Section III certificate holder. A code reconciliation has been completed and the replacement bellows are functionally equivalent to the original design. The anti-sweat insulation will help to eliminate the formation of condensation within the bellows. The additional weight and combustible loading of the insulation has been evaluated.

PLANT CHANGE/MODIFICATION 97086

CONTAINMENT TOOL ROOM REMOVAL

Summary:

This modification is associated with the 1997 steam generator replacement outage. The containment tool room enclosure was a seismically designed structural steel and grating "cage" adjacent to the construction hatch, which was used for the storage of tools. Replacement of the steam generators necessitated the removal of this tool storage enclosure because of its proximity to the construction hatch. The tool room enclosure was not reinstalled and the tools stored in the area were relocated to appropriate storage areas outside of containment.

STEAM GENERATOR REPLACEMENT REPORT SAFETY EVALUATION

Summary:

This safety evaluation evaluates the physical implementation activities (e.g., rigging and handling, heavy loads, etc.) associated with temporary and permanent changes to equipment and components conducted before, during and following the 1997 steam generator (SG) replacement outage. In addition, the evaluation considers subsequent plant operation with various permanent plant changes to be implemented as a part of the SG replacement project.

The activities evaluated include:

- heavy load evaluations for the rigging/handling of, and transport/haul routes for, the SGs;
- modifications of pipe rupture restraints from their as-designed configuration;
- modifications to the SG manway platforms;
- modifications to the SG insulation;
- modifications to the SG blowdown piping;
- erection and utilization of the SG interim storage facility;
- modifications to the SG supports;
- SG replacement;
- contingency replacement of cold leg elbows;
- shield building concrete construction hatch;
- containment preparations; and
- 1A main steam trestle structural modifications.

Prior to the SG replacement outage and during plant operation, temporary site facilities and the SG interim storage facility are constructed. Also, a portion of the temporary gantry crane (TG) and the hatch transfer system is erected outside of the shield building at the construction hatch. Removal of interferences and connecting secondary piping and installation of the remaining portion of the TG begins with the unit in Mode 5. Activities which would impair the availability of the SGs for decay heat removal are deferred. Actual SG replacement, including the severing of RCS piping, is performed with the unit defueled. Reactor vessel internals are stored within the defueled reactor vessel until SG replacement has been competed. The old SGs are hoisted out of their cubicles using a temporary lifting device and are downloaded onto the hatch transfer system where they are transferred out of containment. The replacement SGs are installed in essentially a reverse order of the removal process. Once the new SGs are installed, including connecting piping (and related testing), and the containment vessel construction hatch dome is welded in place, the containment is "returned" for the continuation of refueling activities.

Additional 10 CFR 50.59 considerations related to the SG replacement project can be found in JPN-PSL-SENP-94-026, PSL-ENG-SEMS-97-054 and PSL-ENG-SEMS-97-085.

SAFETY EVALUATION JPN-PSL-SENP-94-026
REVISION 1

STEAM GENERATOR EQUIVALENCY REPORT

Summary:

The steam generator equivalency report (SGER) compares the replacement steam generators (RSGs) to the original steam generators (OSGs) and documents the unreviewed safety question determination for the RSGs. The following discussion summarizes the more significant items addressed in response to the considerations of 10 CFR 50.59.

- The RSGs operate at the same secondary side pressure, temperature and flow as the OSGs and RSG steam moisture content is lower than the OSGs, thus there is no increase in the severity of the conditions in the steam piping.

- There are 8,523 tubes in each RSG versus 8,519 tubes in the OSGs. This small (<0.05%) increase in the number of tubes is offset by the use of stronger tubes and tubes which are more corrosion resistant (Alloy 690 - RSGs versus Alloy 600 - OSGs).

- The RSG tube bundle configuration is different in tube bundle shape and tube bend radius. The RSG upper tube bundle shape uses tubes with continuous, smooth, long-radius bends versus the OSGs which use horizontal straight runs. The U-bends of the innermost RSG tubes are skewed to provide longer minimum bend radius than the OSG design. The RSG tubes have been evaluated for mechanical resonance and pressure variation to demonstrate that they meet the same frequency, stress and strain criteria as the OSG tubes.

- The RSG tube to tubesheet weld was designed, and constructed in accordance with ASME Section III criteria (like the OSGs) and RSG tubes have been hydraulically expanded through the full depth of the tubesheet.

- The nominal RSG tube ID is 0.006 inches larger than the OSG tube ID. The length of the shortest RSG tube is five feet greater than that of the shortest OSG tube. A detailed hydraulic analysis of these differences has shown the beneficial effect of a longer tube outweighs the detrimental effect of the larger ID. Specifically, the integrated tube rupture break flow of the limiting RSG tube is the same as that for the limiting OSG tube.

- The RSGs are four percent heavier and have a higher center of gravity than the OSGs. An evaluation of these effects has demonstrated the loads on supports and attached piping are less than the design loads used in the original analyses to show compliance with ASME code requirements.

- The RSG and OSG thermal-hydraulic response to, and sequence of events for, accidents are equivalent. Actions prescribed in plant procedures for the OSGs remain appropriate for the RSGs. Likewise, the Technical Specification limits related to the steam generators remain appropriate for the RSGs.

- Evaluation of safety analysis report (SAR) accidents with the RSGs in place of the OSGs has been performed. Five events were identified in which the parametric results could exceed those for the OSGs: loss of feedwater, containment response to a large break LOCA, steam generator tube rupture, containment response to a main steam line break, and RSG supports and attached piping loads. Each of these issues were further evaluated in the SGER and demonstrated to not represent a decrease in the margin of safety as defined in the basis for any Technical Specification.

Additional 10 CFR 50.59 considerations related to the SG replacement project can be found in JPN-PSL-SEMP-93-033, PSL-ENG-SEMS-97-054 and PSL-ENG-SEMS-97-085.

SAFETY EVALUATION JPN-PSL-SEIS-96-008
REVISION 1

TEMPORARY USE OF GRIPPER ENGAGEMENT MODULE IN UNIT 1 CEDMCS

Summary:

This document evaluates the safety significance of the temporary use of a gripper engagement module (GEM) in the control element drive motor control system (CEDMCS). The use of the GEM will assist in troubleshooting of the CEDMCS and will minimize the probability of a dropped control rod during troublshooting activities and rod exercising during unit startup.

The GEM is a microprocessor controlled monitoring device with features which allow it to detect certain electrical and mechanical failures and prevent an inadvertent rod drop. The GEM supplied by ABB-CE has been specifically designed for use in the St. Lucie Unit 1 CEDMCS. It has been tested and its programming has been verified by ABB-CE. The GEM has also been previously tested on the St. Lucie simulator. The GEM does not interact with or affect the CEA block circuit and cannot prevent a CEA from inserting into the core on a reactor trip.

SAFETY EVALUATION JPN-PSL-SEFJ-96-022
REVISION 1

EVALUATION OF THE BEST ESTIMATE ANALYZER FOR CORE OPERATIONS - NUCLEAR (BEACON)

Summary:

This evaluation qualified the methodology and performance of the BEACON system for core analysis at St. Lucie Units 1 and 2. Although the BEACON system provides monitoring, analysis and prediction functions, this evaluation has a limited scope to allow the use of BEACON as a replacement for the current incore analysis codes INPAX (Unit 1) and CECORE (Unit 2). The functions of BEACON addressed in this evaluation include: 3-dimensional nodal calculations, interface with the plant computer system, fixed incore detector analysis, model calculation, shape annealing factor calculation, monitor summary table, and system redundancy.

BEACON is an advanced on-line core monitoring and support system which primarily uses rod positions, core inlet temperature and fixed incore detector instrumentation signals in conjunction with a complete analytical methodology for generation of near real-time 3-D power distributions. The heart of the system is an NRC approved, three-dimensional nodal code, ANC.

SAFETY EVALUATION JPN-PSL-SEIS-96-048
REVISION 0

REMOVAL OF PSL UNIT 1 HPSI/LPSI FLOW CONTROL VALVE POSITION INDICATORS

Summary:

This evaluation allows the removal of valve position indicators (VPIs) associated with various high pressure safety injection (HPSI) and low pressure safety injection (LPSI) flow control valves. The affected VPIs are listed in UFSAR Table 7.5-2 as safety related display instrumentation; however, UFSAR Section 7.5.1.6.1 states that equipment status and valve positions are displayed by status lights. Note that the affected VPIs provide indication of valve percent open/closed. The affected valves also have open/closed status indicating lights provided in the control room. The open/closed status lights are not affected by this evaluation.

The flow control valve VPIs are not relied upon for any safety related activity. Whenever the LPSI system is used for shutdown cooling, operators rely upon information provided by flow indicators to throttle the LPSI valves to the desired position. Similarly, when the HPSI system is used to control level in the RCS, pressurizer level indicators are relied upon. Since the subject VPIs are not required by NRC Regulatory Guide 1.97 and they are not required for any safety related activity, it is acceptable to remove them. This evaluation is associated with Plant Change/Modification 96027.

SAFETY EVALUATION JPN-PSL-SEMS-96-067
REVISION 3

INSTALLATION OF TEMPORARY LEAK REPAIR ENCLOSURE ON HVS-1B COOLING COIL SUPPLY LINE

Summary:

Operational leakage was discovered in the component cooling water (CCW) supply header to one of six cooling coils for the safety related 1B containment cooling unit (HVS-1B). A visual inspection characterized the leak as 6 to 8 separate pin hole leaks in the joint brazing material in the brazed joint between the 3" carbon steel pipe stub and 3" copper cooling coil header.

This evaluation addresses the installation of a temporary leak repair closure on the subject joint. NRC Generic Letter 90-05 and subsequent correspondence documents are used as the guidance for performing the leak repair. The leak repair enclosure is considered a stopgap measure to stop the leakage. The use of stopgap measures to limit operational leakage is consistent with regulatory guidance for leak repairs on Class 3 moderate energy piping and components. The temporary leak repair has been evaluated to not affect the structural integrity of the flawed piping and is reversible (i.e., can be removed to allow future permanent repairs). Operation of containment cooler 1B and the CCW system is not adversely affected.

SAFETY EVALUATION JPN-PSL-SEES-96-068
REVISION 1

EVALUATION OF TEMPORARY OPERATING CONFIGURATION WITH 1B MAIN TRANSFORMER OUT OF SERVICE

Summary:

While operating at 100% power, a gas detection alarm was received for the 1B main transformer. Investigation showed that air was being introduced into the transformer oil and that continued operation could lead to a breakdown of the insulating oil with resultant loss of the transformer.

Normal plant operation is with both the 1A and 1B main transformers in parallel. System design allows for operation at reduced power with a single main transformer, provided certain conditions are met. Operation with a single main transformer is addressed in UFSAR Section 8.2.1.4 and does not affect or reduce the plant's ac electrical power sources as required by the Technical Specifications. The main transformers are non-safety and are not considered important to safety. This evaluation allowed continued plant operation until the 1B transformer could be restored to normal service. Appropriate operating restrictions were provided.

SAFETY EVALUATION JPN-PSL-SENS-96-069
REVISION 0

HIGH RATE OF CHANGE OF POWER TRIP DESIGN BASIS

Summary:

This safety evaluation was issued to document the design basis of the high rate of change of power (HRCP) trip of the Unit 1 and 2 reactor protection system (RPS). Both the Unit 1 and 2 UFSARs describe the HRCP trip as an equipment protective trip which is not required for reactor protection and is not credited in accident analyses; however, information provided by the reactor plant vendor, ABB-CE, clarified the importance of this trip.

ABB-CE issued TechNote 96-04 in order to clarify the original design intent of the trip. The TechNote suggests that a lack of discussion in original FSARs has caused some utility and ABB-CE personnel to erroneously assume the HRCP trip was not credited in a plant's safety analysis. Per ABB-CE, the presence of the HRCP trip precluded the specific analysis of events initiated from subcritical conditions. This evaluation incorporates the conclusions of the ABB-CE TechNote into the Technical Specification Bases and the UFSARs.

SAFETY EVALUATION JPN-PSL-SEIS-96-077
REVISION 0

TEMPORARY USE OF SHUTDOWN GROUP TIMER MODULE FOR CEDM #50

Summary:

After CEA #50 dropped on 9/16/96 the timer module was replaced and the rod was recovered. Coil traces indicated that a shutdown group timer was inadvertently used in lieu of a regulating group timer. This evaluation documented the acceptability of continued plant operation with the shutdown group timer until the timer could be replaced.

The function of the control rod drive system is to control the power level of the reactor by the withdrawal or insertion of the CEAs. Rod #50 belongs to a regulating group which is used in the fully withdrawn position during normal operation. Should movement or control of that regulating group become necessary, then rod #50 would lag behind the other rods in its group due to the difference in sequence timing intervals. Operators would be expected to experience a deviation alarm prior to reaching Technical Specification limits. If desired, rod #50 could be repositioned by individual CEA control.

The safety function of CEA #50 is to insert on a loss of power to its drive motor (i.e., a trip). The use of a shutdown timer has no impact on the "drop time" or tripping function of CEA #50.

SAFETY EVALUATION JPN-PSL-SENS-96-079
REVISION 0

CONTROL ROOM ANNUNCIATORS

Summary:

This is a generic safety evaluation which was written for temporary system alterations (TSAs) to allow the disabling of input signals to control room annunciators. The evaluation allows such a TSA provided the following acceptance criteria are satisfied:

1) the input signal(s) to the annunciator is (are) verified as not valid and the failed instrument loop(s) feeding the annunciator is (are) declared out of service; and

2) the TSA does not disable or adversely affect any operable instrument loops.

The disabling of an annunciator with known invalid signals enhances the control room environment by removing the distraction provided by the annunciator, thereby improving operators' ability to focus on valid information.

SAFETY EVALUATION JPN-PSL-SEMS-96-083
REVISION 0

SAFETY EVALUATION OF AUXILIARY FEEDWATER SYSTEM WITH "AB" 125 VDC POWER SUPPLY TRANSFER

Summary:

This safety evaluation reconciles a discrepancy between the UFSAR and plant procedures. Specifically, UFSAR Section 10.5.3 describes the manual cross-tie of the A or B 125 Vdc bus to the AB 125 Vdc bus during a postulated failure of the 125 Vdc bus that is aligned to the AB bus, in combination with a main feedwater pipe break or auxiliary feedwater (AFW) pipe break. This action, as described in the UFSAR, is performed in the control room via key lock switches and is required within 10 minutes of event initiation under certain design basis events assumed for AFW. Existing plant procedures do not specifically provide instructions for transferring the AB bus to an energized bus to restore power to the 1C AFW pump and associated valves.

A condition report (96-2825) was written to document the above. Long term corrective actions included a revision to the emergency operating procedures (EOPs). This evaluation documents that the proposed EOP changes are consistent with the original operator actions described in the UFSAR. This evaluation also revised the UFSAR description of the event to be consistent with current plant design and post-TMI operating philosophy (e.g., the original UFSAR description was written prior to the automatic feedwater actuation system).

SAFETY EVALUATION JPN-PSL-SEMS-96-086
REVISION 0

CLARIFICATION OF AUXILIARY FEEDWATER SYSTEM START TIME REQUIREMENTS

Summary:

The auxiliary feedwater (AFW) system is a safety related system which, for several analyzed events, is assumed to be initiated at some point during the event. Depending upon the event type (i.e., overheating or overcooling), different event elapsed times are assumed for AFW initiation. This evaluation revises the UFSAR and Design Basis Documents to clarify the actual analytical requirements and, where appropriate, distinguish the analytical requirement or limitation from the value actually used in the analysis. For example, the UFSAR analysis for a steam line break event assumes AFW initiation at 180 seconds, whereas the analytical requirement for AFW initiation is at a time less than or equal to 205 seconds. Thus, the value assumed in the actual analysis (180 seconds) is conservative with respect to the analytical requirement (205 seconds) and a note is being added to the affected UFSAR table to clarify this point.

SAFETY EVALUATION JPN-PSL-SENS-96-091
REVISION 0

HYPOCHLORITE SYSTEM UFSAR CLARIFICATION

Summary:

St. Lucie plant uses a sodium hypochlorite system as a means of controlling biofouling of heat exchanger surfaces for sea water systems. The non-safety hypochlorite system consists of hypochlorite generating equipment, which manufactures sodium hypochlorite from sea water, and a delivery system, which provides for the controlled injection of hypochlorite into the sea water intake wells.

A review noted the UFSAR identified the system as a "hypochlorite generating system," thus implying the system exclusively generates its own hypochlorite. While this is generally true it is not always true. A hypochlorite solution can be brought in from offsite and stored in a suitable system storage tank for subsequent injection via the hypochlorite delivery system. This evaluation simply replaces the term "hypochlorite generating system" with "hypochlorite system" in order to more accurately describe the system.

SAFETY EVALUATION JPN-PSL-SEFJ-97-005
REVISION 0

EVALUATION OF SIT CROSS-TIE CONFIGURATION ON LOCA ANALYSIS

Summary:

It was identified that during surveillance activities the safety injection tanks (SITs) for Units 1 and 2 remain cross-connected for short durations via a common nitrogen header. This configuration had not been previously considered in plant analyses. In the normal alignment the SITs remain isolated. A review of the safety analyses for both Unit 1 and Unit 2 determined that only LOCA analyses could be affected by the cross-tie configuration.

The SIT cross-tie configuration is not explicitly modeled in the LOCA analysis for either unit. This safety evaluation assesses the impact of the above SIT cross-connecting on the large break and small break LOCA analyses of record for both units. In all cases, the analyses of record were determined to remain bounding.

SAFETY EVALUATION JPN-PSL-SEFJ-97-006
REVISION 0

FSAR LBLOCA ANALYSIS UPDATES

Summary:

In 1996 the NRC declared the Siemens Power Corporation (SPC) 1991 large break LOCA (LBLOCA) evaluation model unacceptable for licensing applications due to changes made by SPC to the reflood heat transfer model as approved by the NRC in the 1986 evaluation model. Additionally, the NRC declared the previously approved 1986 evaluation model to be unacceptable without an adequate correction to the non-physical behavior of the reflood heat transfer correlation in the range of flooding rates from 1.0 in/sec to 1.77 in/sec. The Unit 1 UFSAR analysis of record was performed using the SPC 1991 evaluation model.

This safety evaluation incorporates into the UFSAR the LBLOCA reanalysis performed as a result of the above. Subsequent to the NRC notification, the LBLOCA analysis of record was reanalyzed using the SPC 1986 evaluation model with appropriate changes made to the interpolated heat transfer coefficient in the range of flooding rates discussed above. This reanalysis was reviewed and accepted by the NRC as documented in a 12/6/96 letter to FPL.

SAFETY EVALUATION PSL-ENG-SENS-97-010
REVISION 0

UFSAR CLARIFICATION OF CLASS B2 CONTAINMENT PENETRATIONS

Summary:

The UFSAR description of a Class B2 containment penetration describes the isolation valves as "never opened during power operation." While this is true for normal plant operations, the valves are periodically stroke tested in accordance with approved plant procedures. This safety evaluation justifies a revision to the UFSAR description to delete the term "never opened."

A review of the affected systems shows they would never be operated during power operations. Compliance with plant technical specifications provides assurance that containment integrity is maintained. Stroke testing of outboard containment isolation valves can be accomplished within the requirements of the technical specifications by deactivating and securing the associated inboard isolation valve. With the inboard valve deactivated and secured, containment integrity is assured and no single failure can prevent containment isolation.

SAFETY EVALUATION PSL-ENG-SECS-97-014
REVISION 0

SAFETY EVALUATION FOR SPECIFICATION SPEC-C-035: INSTALLATION OF TYGON TUBING FOR VENTING AND DRAINING IN-SERVICE EQUIPMENT

Summary:

This safety evaluation is issued in support of Specification Spec-C-035, which allows the installation of vent and drain rigs to operating plant equipment. Plant personnel are allowed to install such rigs for in-service equipment provided the requirements of the specification are satisfied. This is in lieu of the previous practice which essentially required the applicable system to be declared out of service. The specification provides installation instructions for vent and drain rigs for both safety related and non-safety equipment. The vent and drain rigs themselves are considered non-safety.

This evaluation considers the structural and seismic impact of attaching the vent and drain rigs to piping systems. System operation is addressed via plant procedures; as such, operational issues associated with the opening and closing of vent and drain valves are not within the scope of the evaluation or specification. This evaluation concludes the installation of vent and drain rigs per the subject specification does not have any adverse impact on the operation of the affected system(s) and that technical and licensing requirements remain satisfied.

SAFETY EVALUATION PSL-ENG-SEMS-97-016
REVISION 0

TEMPERATURE CONTROL VALVE FOR THE HYDROGEN COOLERS (TCV-13-15) MANUAL OPERATION

Summary:

Temperature control valve TCV-13-15 controls turbine cooling water (TCW) flow to the main generator hydrogen coolers. The TCW system is a non-safety system which provides cooling water to various turbine related equipment. The UFSAR description of the TCW system notes that the components served by TCW are provided with automatic temperature control valves in the cooler outlet piping. This safety evaluation documents the acceptability of operating the subject valve in the manual mode.

Hydrogen gas temperature in the main generator is maintained by four heat exchangers with TCV-13-15 located on the common outlet line. Placing TCV-13-15 in manual prevents the valve from automatically controlling temperature; however, discussions with the turbine vendor and the FPL turbine specialist concluded that long term operation with the valve in manual will not increase the rate of machine degradation as long as hydrogen gas temperatures are maintained within the recommended operating band. Since the temperature band is relatively wide and since TCW temperature is relatively constant, operation with the subject control valve in manual is acceptable.

SAFETY EVALUATION PSL-ENG-SEES-97-020
REVISION 0

INSTALLATION OF TEMPORARY CCTV SECURITY CAMERAS IN THE TURBINE BLDGS.

Summary:

This safety evaluation allows the installation of closed circuit television (CCTV) security cameras in the Unit 1 & 2 turbine buildings. The installation is to be used to augment existing Security Department surveillance activities.

There is no safety related or essential equipment in the affected areas. The cameras are to be powered from existing 120 Vac power receptacles. CCTV cables will be ty-wrapped to existing supports and/or conduits. The cables will not be routed in cable trays or mixed with plant cables and attachment to process piping is not permitted. The installation of cameras, junction boxes and cables does not adversely affect any plant equipment. The initial installation of the CCTV equipment via this evaluation is considered temporary. A permanent installation at a later date is anticipated.

SAFETY EVALUATION PSL-ENG-SENS-97-024
REVISION 0

10CFR50.59 EVALUATION FOR WASTE GAS SYSTEM ANALYZER OPERATION

Summary:

This safety evaluation addresses UFSAR discrepancies related to the operation of the Unit 1 and 2 waste gas system analyzers as documented in Condition Report 97-0350. The descriptions in the Unit 1 and Unit 2 UFSARs require correction to provide clarification for gas analyzer operation and to address manual operation for continuous single point sampling as an acceptable practice in addition to automatic programmed sequential sampling for oxygen concentration. The Unit 1 UFSAR changes are limited to minor editorial corrections; the Unit 2 changes are more substantial.

The function of the waste gas system analyzers is to detect oxygen so that action can be taken to preclude the formation of potentially explosive mixtures of hydrogen and oxygen because the system is not designed to withstand the effects of an explosion. Continuous monitoring of a single sample point for either of the normal lineups, the gas surge header or the in-service gas decay tank, are acceptable and do not adversely impact safe plant operation or plant safety.

SAFETY EVALUATION PSL-ENG-SEFJ-97-033
REVISION 0

COMPARISON OF FPL TO SPC CYCLE 13 PHYSICS PARAMETERS

Summary:

Juno Beach Condition Report (CR) 96-0023 addressed the incorrect development of scram curves for Cycle 13 and 14 due to feedbacks not being appropriately frozen in the FPL calculations due to a computer code error. As a result, a parameter by parameter comparison of the data calculated by FPL for St. Lucie Unit 1 Cycle 13 to the data calculated by Siemens Power Corporation (SPC) for Cycle 13 was performed. The review determined that rod ejection data for Cycles 13 and 14 were incorrect for the part-strength rod in the lead bank (all other parameters were determined to be calculated correctly). This safety evaluation was written to document an analysis of the effects of the above on the Cycle 14 analysis.

The Cycle 14 rod ejection results were evaluated to account for the above error. It was concluded that the results of the analysis of record remained bounding when considering the effect of the error. Specifically, the change in the hot zero power physics data did not alter the conclusion that the hot full power cases are bounding. The Cycle 15 rod ejection calculations were reviewed and found not to be effected by the same error. (This safety evaluation was completed approximately 45 days prior to the end of Cycle 14).

SAFETY EVALUATION PSL-ENG-SEMS-97-035
REVISION 1

10CFR50.59 SAFETY EVALUATION FOR INSTALLATION OF A TEMPORARY LEAK REPAIR ENCLOSURE ON HCV-08-2B

Summary:

Operational leakage was discovered in the body to bonnet flange joint of HCV-08-2B, the atmospheric dump valve for the 1B main steam supply header. A visual inspection characterized the leak as a minor steam leak at the valve to gasket interface.

This evaluation addresses the installation of a temporary leak repair enclosure on the subject joint. The leak repair enclosure is designed for system pressure and temperature conditions. The additional weight of the enclosure (40 lbs.) is considered to have an insignificant effect on pipe stress and support loads when considering the total weight of the valve (1600 lbs.). The sealant used is appropriate for the conditions and the volume injected is limited to the calculated volume of the enclosure. NRC Generic Letter 90-05 does not apply to this temporary repair since "leakage through a flange gasket is not considered to be a flaw in the piping by Section XI of the ASME Code and is excluded."

The leak repair is considered temporary until a permanent repair can be made during the next outage of sufficient duration.

SAFETY EVALUATION PSL-ENG-SENS-97-041
REVISION 0

PARTIAL STROKE TESTING OF SIT DISCHARGE VALVES

Summary:

This safety evaluation demonstrates the acceptability of performing a partial stroke test of the safety injection tank (SIT) discharge check valves. Check valve full-stroke, partial flow testing has previously been performed to successfully address NRC requirements for SIT check valve testing as delineated in Generic Letter 89-004. Check valve partial stroke tests are performed following valve maintenance activities.

Partial stroke testing may be performed in any plant operating mode such that the SITs are not required to be operable by the Technical Specifications. Fuel movement shall be suspended during the test. SIT discharge for the partial stroke test may be aligned to the refueling cavity, pressurizer or refueling water tank. Nitrogen pressure in the SIT is used as the motive force for the test. The test is initiated by opening the SIT discharge line motor operated valve and/or controlling discharge flow by throttling the "SI loop check valve leakage" valve. As a SIT drains, the level and pressure are closely monitored to ensure Technical Specification operating limits are not exceeded. This partial stroke testing is bounded by existing analysis for full-stroke testing of the subject check valves.

10CFR50.59 SAFETY EVALUATION FOR FIRE PUMPS STARTING SEQUENCE

Summary:

This evaluation is in response to Quality Assurance (QA) audit QSL-FP-96-23 and Condition Report 97-0466. The evaluation justifies the acceptability of the existing starting point (i.e., header pressure) and starting sequence for the fire pumps and revises the UFSARs (Unit 1 and Unit 2) accordingly.

The existing UFSAR wording stated the fire pumps start automatically when header pressure drops to 85 psig. This evaluation clarifies that wording to note automatic starting of the fire pumps occurs at greater than or equal to 85 psig. Also, Appendix A to NRC Branch Technical Position 9.5-1 notes that "Details of the fire pump installation should as a minimum conform to NFPA 20..." NFPA 20 includes a requirement for the sequential starting of fire pumps as opposed to the St. Lucie design which allows simultaneous starting of the pumps. The basis for the sequential starting requirement is not stated in the NFPA Code. The postulated basis for such a requirement is to prevent water hammer and/or to prevent electrical overload. The NFPA was contacted and agreed this is a reasonable basis for this section of the Code. The scenario postulated for a water hammer to occur would be a pump start with the system piping voided. Since the fire water system at St. Lucie is maintained under pressure via the hydropneumatic tank which is pressurized by the domestic water pumps, water hammer is not a concern. Electrical overload is also not a concern since the two pumps are powered from separate electrical buses. Based on the above, the UFSAR changes were considered acceptable.

SAFETY EVALUATION PSL-ENG-SENS-97-047
REVISION 0

OPERATION OF THE UNIT 1 FIRE DETECTION COMPUTER WITH THE UNIT 2 DATA-LINK CABLE LEADS LIFTED

Summary:

The fire detection computers of each unit provide their respective control rooms with monitoring and alarm information for fire protection purposes. These computers have a coaxial "data-link" or "cross-tie" feature which allows the Unit 1 and Unit 2 fire detection computers to back each other up in the event one of the computers is out of service. The data-link cable associated with sending Unit 1 data to the Unit 2 computer became inoperable. In order to allow maintenance on the cable, the leads had to be lifted at the Unit 1 computer and end-of-line resistors had to be installed to provide an internal current loop path for continued computer operation. In this temporary configuration, Unit 1 data is not available to the Unit 2 Fire Detection System computer. There is no effect on operation of the Unit 2 computer or on the ability of the Unit 2 computer to send data to the Unit 1 computer.

This evaluation documents the 10 CFR 50.59 consideration for the temporary operation of the Unit 1 fire detection computer with the data-link cable leads lifted and end-of-line resistors installed. Repair/replacement of the data-link cables was addressed via Condition Report 97-1244.

SAFETY EVALUATION PSL-ENG-SENS-97-050
REVISION 0

ROUTINE PERFORMANCE OF FULL CORE FUEL OFFLOADS

Summary:

The purpose of this safety evaluation is to document that the performance of a full core fuel offload at the end of Cycle 14 and on a routine basis thereafter is an acceptable evolution which does not present an unreviewed safety question. Present regulatory guidance discusses and implicitly permits the use of full core offload evolutions and industry refueling practices have evolved to view the temporary removal of the entire core during refueling as superior to the fuel shuffle evolution.

It is recognized that following a full core defueling the decay heat load on the fuel pool cooling system is greater than the heat load experienced during a fuel shuffle evolution where only the permanently discharged fuel is moved to the spent fuel pool. The capability of the spent fuel pool cooling system to handle this greater decay heat load must be verified for each refueling evolution and this evaluation provides specific requirements to that effect, including provisions to ensure the calculated value of the maximum decay heat load to the fuel pool cooling system is □33.7E6 BTU/hr and the maximum fuel pool temperature for the anticipated core offload evolution is □140 °F with one fuel pool cooling pump in operation.

This safety evaluation also demonstrates that temporary placement of the reactor internals in the defueled reactor vessel and the placement of the reactor head on the vessel flange for temporary storage is an acceptable evolution which does not adversely affect plant operation, safety or require a change to plant Technical Specifications.

SAFETY EVALUATION PSL-ENG-SEMS-97-053
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR POST ACCIDENT SAMPLING SYSTEM FSAR RECONCILIATION

Summary:

This safety evaluation provides the justification to revise the UFSAR description of the post accident sampling system (PASS). Since the initial installation of the system, changes have been made based on regulatory requirements and system enhancements; however, the UFSAR description has not been clearly updated. This evaluation provides the necessary UFSAR changes to reflect current plant configuration and operation of the PASS.

The changes to the PASS as described in the UFSAR include:

1) removal of the requirement for measurement of dissolved (separated) gas flow and hydrogen concentration;
2) removal of the requirement for measurement of the dissolved oxygen and pH in the liquid sample;
3) removal of the requirement for the gas blender with a dilution range of up to 1:1000;
4) removal from the UFSAR of component details and values (e.g., pressures and temperatures) where they are not regulatory requirements; and
5) removal of the UFSAR requirement for the PASS to sample containment atmosphere.

SAFETY EVALUATION PSL-ENG-SEMS-97-054
REVISION 2

REVIEW OF PLANT OPERATION WITH REPLACEMENT STEAM GENERATORS

Summary:

This safety evaluation reviews Unit 1 Cycle 15 plant operation with replacement steam generators (RSGs) and supplements the review contained within the Steam Generator Replacement Report and the Cycle 15 reload modification package. The purpose of this evaluation is three-fold:

1) to predict NSSS and balance-of-plant (BOP) plant system operating parameters using an independent thermal/hydraulic model with zero percent plugging in the RSGs;
2) to review the effects of major plant system parameter changes with respect to the values observed during Cycle 14 operation and/or original stretch power operation; and
3) to review plant instrumentation and documentation changes made prior to Cycle 14 (25% tube plugging) for adequacy with respect to operation with the RSGs.

The RSGs incorporate a number of design and manufacturing innovations. Review of these differences is provided in the Steam Generator Equivalency Report, including a thermal/hydraulic analysis of the RSGs based on a model developed by Babcock & Wilcox. This evaluation provides an independent review of expected NSSS response and extends the performance review to include turbine cycle parameters. This evaluation uses an analytical model previously developed by ABB-CE Nuclear Operations for the NSSS and Southern Electric International (SEI) for the BOP systems. The operating characteristics anticipated with the RSGs were developed using the SYSFLOW, EVAP and PEPSE computer codes.

Elimination of the Cycle 14 steam generator tube plugging will result in an increase in RCS flow, a decreased core delta-T and an increase in steam generator pressure. Plant operation and response to off-normal conditions with the RSGs has been evaluated in this evaluation and in the other evaluations previously mentioned. Each of the evaluations concludes that operation with the RSGs has no adverse affect on the plant.

SAFETY EVALUATION JPN-PSL-SEES-97-061
REVISION 0

TEMPORARY DISABLING OF THE 1B 125 VDC SYSTEM GROUND ANNUNCIATION

Summary:

This safety evaluation was written in support of trouble shooting operations to locate and eliminate the source of numerous intermittent alarms apparently caused by a momentary ground on the 1B 125 Vdc system. The evaluation allowed the temporary disabling of the 1B dc ground annunciator and installation of one or more voltmeters to periodically monitor for dc grounds. There is no adverse affect on the operation of the 125 Vdc system as a result of this temporary configuration. This evaluation was associated with Temporary System Alteration TSA-1-97-017.

SAFETY EVALUATION PSL-ENG-SEES-97-065
REVISION 1

CROSSTIE OF THE SPENT FUEL POOL PUMPS POWER SUPPLY

Summary:

During refueling outages, electrical system maintenance requiring the de-energization of either 480V motor control center (MCC) 1A8 or 1B8 is performed. These MCCs each power a non-safety spent fuel pool pump, among other loads. Although not a Technical Specification requirement, one fuel pool pump is required to be operating to maintain adequate cooling with a full core discharge. In order to provide additional cooling and some level of redundancy while performing maintenance on the electrical system, it is desired to have power available to both spent fuel pool pumps while either MCC 1A8 or 1B8 is out of service. This safety evaluation allows the temporary installation of a power feed from the MCC remaining in service to the pump normally powered by the MCC to be removed from service.

Implementation of the above electrical cross-tie consists of installing a spare 100 amp circuit breaker in a spare compartment of the MCC remaining in service, disconnecting the line side of the spent fuel pool pump circuit breaker in the MCC to be removed from service, and routing a temporary "jumper" cable between the MCCs from the load side of the spare circuit breaker to the line side of the spent fuel pool pump circuit breaker. Electrical train separation and isolation is maintained by using a jumper cable which meets the flame test requirements of IEEE 383-1974, by providing electrical isolation from safety related power sources via the use of properly coordinated protective devices, and by not routing the jumper cable in any raceways or conduits containing safety related cables. Additionally, the jumper cable will not be supported from any seismic supports or safety related equipment. Although MCCs 1A8 and 1B8 are loaded onto emergency diesel generators 1A and 1B respectively during a loss of offsite power, the spent fuel pool pumps reset to the off condition (i.e., they do not automatically restart) and must be manually loaded onto the diesel generator, if desired.

SAFETY EVALUATION PSL-ENG-SEMS-97-066
REVISION 0

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

Summary:

This safety evaluation addresses the addition of hydrogen peroxide to the reactor coolant system (RCS) during shutdown cooling (SDC) in order to oxygenate the coolant and facilitate crud burst/removal for Units 1 & 2. The addition of hydrogen peroxide to the RCS during SDC is a common industry practice. The resulting oxygenation causes a portion of the RCS crud to slough off internal surfaces and into the process stream where it then can be removed by the purification system. To ensure chemistry limits are not violated, the RCS temperature shall be □ 200F and the RCS hydrogen concentration shall be less than 5 cc/kg prior to the addition of hydrogen peroxide. Approximately 4 gallons of 30% hydrogen peroxide solution will be initially added to the chemical addition tank upstream of the charging pumps; additional hydrogen peroxide will be incrementally added based on oxygen and hydrogen peroxide analysis results.

SAFETY EVALUATION PSL-ENG-SEMS-97-067
REVISION 0

INSPECTION OF THE UNIT 1 REFUELING WATER TANK DURING PLANT OPERATION

Summary:

This safety evaluation provides for the visual inspection of the refueling water tank (RWT) bottom with the use of a diver or remotely operated submersible vehicle during normal power operations and during refueling outages. The RWT is a safety related tank that provides a source of borated water for the emergency core cooling system.

The NRC approved an alternative non-code repair on the RWT bottom in 1994. This repair involved the installation of a fiberglass reinforced vinyl ester liner on the tank bottom. In the associated relief request, FPL committed to a full hands-on inspection of the RWT bottom, with the tank dry, every third outage, and a remote visual inspection with the use of a diver or remotely operated submersible vehicle (ROV) equipped with a camera during other outages.

The only operational issue associated with the on-line RWT inspection is that a diver or the ROV could potentially block flow in one of the RWT penetrations. This evaluation considers the RWT penetrations and their associated flow velocities in the lower portion of the RWT and provides appropriate precautions, including the following:

1) a tether is required for the diver or ROV;
2) the diver must wear a safety harness attached to the tether and must remain in voice contact with support personnel outside the RWT;
3) diver and ROV should avoid time spent within a 6' radius of the 24" safety injection system piping;
4) time spent in the RWT shall be minimized; and
5) foreign material exclusion precautions shall remain in effect at all times when the RWT hatch is open.

SAFETY EVALUATION PSL-ENG-SECS-97-069
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR SITE SANITARY SYSTEM MODIFICATIONS

Summary:

The Unit 1 site sanitary system has been identified as a system requiring periodic modifications and maintenance. The Unit 1 site sanitary system is a non-safety system that process sanitary waste from the north side of the plant site, including toilet and shower facilities located within the power block (both units). The system also includes a sanitary treatment facility which is a stand-alone system under the jurisdiction of the Florida Department of Environmental Protection.

A review of system functions and interactions has concluded that the system is not required to be considered within the scope of the FPL Quality Assurance Program since it is classified as Not Nuclear Safety and since it has no potential interaction with equipment important to safety. As such, this safety evaluation allows modifications to the system to be performed outside of the formal modification process. Updated drawings will be maintained by Engineering. The specific boundaries for the applicability of this evaluation are identified in an attachment to the evaluation.

SAFETY EVALUATION PSL-ENG-SENS-97-071
REVISION 0

CONTROL ROOM EMERGENCY SUPPLIES

Summary:

Both the Unit 1 and Unit 2 UFSARs include requirements for the storage of emergency supplies in the control room. These emergency supplies consist of food, water, medical and sanitary provisions intended for control room personnel in the event of a design basis loss of coolant accident. The UFSAR descriptions include a list of specific sanitation supplies based on the Office of Civil Defense Sanitation Kit III, which was apparently developed for use in fallout shelters. In addition, the UFSARs note that a sufficient supply of food and water is stored in the control rooms to support a crew of ten persons for a one-week period.

There is no regulatory requirement for the storage of specific emergency supplies in the control room. As such, the level of detail contained within the St. Lucie UFSARs is unnecessarily restrictive and does not allow for efficient plant operations. Additionally, NUREG-0800, Standard Review Plan, does not discuss control room emergency supplies.

This evaluation revises the UFSARs to delete the specific supply requirements from the UFSARs and to recognize the site's procedures with respect to ensuring adequate supplies are available for control room personnel. This action is consistent with Revision 1 to Regulatory Guide 1.101, Emergency Planning for Nuclear Power Plants, which notes that emergency plan requirements should be maintained separately from the FSAR.

SAFETY EVALUATION PSL-ENG-SEMS-97-074
REVISION 1

10CFR50.59 SAFETY EVALUATION FOR REDUCED LPSI PUMP FLOWRATES

Summary:

This safety evaluation addresses the operation of a low pressure safety injection (LPSI) pump at reduced flow rates during the Cycle 15 steam generator replacement outage. During this outage the reactor vessel will be defueled and the reactor internals and head will be replaced in and on the reactor vessel to reduce dose and minimize the impact on floor space. In this configuration, a LPSI pump will be used to drain down the reactor cavity and recirculate flow through the vessel. Although current minimum recommended flow for continuous operation of the LPSI pumps is 1800 gpm, additional information has been provided which allows this value to be reduced to as low as 800 gpm. The maximum flow rate through the reactor vessel internals without the fuel assemblies in place is limited to 1000 gpm. In addition, this safety evaluation addresses the drain down rates, the use of gravity feed from the refueling water tank and/or use of the high pressure safety injection (HPSI) pump(s) to refill the reactor and refueling cavity.

While in this configuration there is no fuel in the vessel and there are no operability requirements for the LPSI pumps; therefore, operation of a LPSI pump with reduced flow does not introduce any operability concerns. The reactor cavity drain down rate can be increased to minimize the duration during which the LPSI pump would be operated at reduced flow rates. The evaluation also provides consideration of and precautions against vortex formation.

The use of a HPSI pump and gravity feed from the RWT to refill the reactor and refueling cavity will also be performed with the reactor defueled. The use of a HPSI pump and gravity feed provides an alternative to the use of LPSI pump for refill. If a HPSI pump is used, a vent path of □1.75 in^2 shall be established prior to actuation of the pump. Prior to the actuation of a second HPSI pump, either the pressurizer manway or the vessel head shall be open.

SAFETY EVALUATION PSL-ENG-SEIS-97-076
REVISION 0

TEMPORARY USE OF CPP ACTM

Summary:

This safety evaluation justifies the temporary installation of two coil power programmer automatic timer modules (CPP ACTMs) in the rod control system after plant shutdown for the Cycle 15 refueling outage. The CPP ACTM, provided as a form-fit-function replacement of the existing CPP timer module, is a microprocessor controlled monitoring device that has been specifically designed by ABB-CE for use in the Unit 1 rod control system. The CPP ACTM generates the timing pulse to control the rod drive motor coils which produce a stepping of the rods. The temporary use will allow for the assessment of the CPP ACTM for later permanent installation. The assessment consists of latching the associated rods, withdrawing the rods above the lower electrical limit, performing diagnostic analysis and unlatching the rods. (Plant Change/Modification 97018, implemented later in the Cycle 15 outage, made the CPP ACTM design change permanent).

The control element drive system is a non-safety system. The temporary use of the CPP ACTM will not have an impact on the operation of the control rods, the rod block circuitry, the ability to control shutdown margin or on the ability of a control rod to insert into the core on a reactor trip. Shutdown margin will be maintained □ 3600 pcm.

Prior to its installation in the plant, the CPP ACTM has been tested and its programming verified by ABB-CE and it has been tested in the St. Lucie Unit 1 Magnetic Jack Test Facility.

SAFETY EVALUATION PSL-ENG-SECS-97-078
REVISION 1

SAFETY EVALUATION FOR TEMPORARY LEAD SHIELDING INSTALLATION CRITERIA AND RESTRICTIONS

Summary:

This safety evaluation was developed to address the criteria and restrictions for the installation of temporary radiation shielding for systems and components which must remain operable, or systems and components which are not required to be operable, but pose a seismic interaction hazard for systems important to safety which are required to be operable. Temporary shielding may be supported from permanent or temporary plant structures or components, or it may be placed directly on piping systems.

This evaluation supports the temporary shielding process which is controlled via plant procedure HP-55. HP-55 requires prior Engineering approval of shielding installations on operable or in-service structures, systems, or components and on structures, systems, or components which are not required to be in service, but whose seismic failure could directly impact in-service or operable safety related, seismically qualified systems. HP-55 requires the completion of a "Temporary Shielding Placement Form" to document the installation and removal of temporary shielding. Health Physics is responsible for providing a description of the shielding to be installed. Operations then reviews the description to determine: (a) whether any shielding will be installed on structures, systems, or components that are required to be operable/in-service; and (b) whether any shielding could impact operable/in-service safety related, seismically qualified systems. If either (a) or (b) is applicable, then the form is forwarded to Engineering for a review of the proposed shielding installation to ensure that all UFSAR design loading requirements/limits are satisfied. Also, shielding to be installed inside containment is reviewed for impact to the ECCS sump and hydrogen generation. After Engineering approves the proposed installation, the shielding may be installed.

10CFR50.59 SAFETY EVALUATION FOR INSTALLATION OF A SECONDARY SA HEADER AND TIE-IN TO THE SA SYSTEM IN THE TGB

Summary:

This safety evaluation supports the installation of: 1) a branch connection and isolation valve in the Unit 1 to Unit 2 station air (SA) cross-tie line; 2) piping headers to various portions of the Unit 1 and Unit 2 turbine generator buildings (TGBs); 3) a tie-in to the construction air system header in the Unit 2 condenser pit; and 4) a tie-in to the Unit 1 TGB ring header. The SA system is a non-safety and non-seismic compressed air system which is used for the operation of pneumatic tools and equipment for maintenance. This change was necessary due to increased demand on the SA system during plant outages.

The SA system piping has a design pressure of 150 psi at a temperature of 125F and is constructed to ANSI B31.1 requirements. The piping header material and pressure rating of valves (for the added header) meets the design requirements of the SA system. Fabrication and installation of the header and supports was consistent with industry codes and standards and established plant programs. The piping configuration has been reviewed for support adequacy and pipe stresses were found to be within design margins.

Connection of the SA system to the construction air system is controlled via approved plant procedures/guidelines. Construction air system air quality is similar to the SA system and the system operates at similar pressures and temperatures.

SAFETY EVALUATION PSL-ENG-SENS-97-084
REVISION 0

MOVEMENT OF THE UPPER GUIDE STRUCTURE WITH ONE CEA ATTACHED DURING REFUELING

Summary:

While removing the upper guide structure (UGS) during the Cycle 15 refueling outage, one of the control element assemblies (CEAs) remained attached to the UGS. The CEA was attached to the UGS and was fully withdrawn from and suspending above the reactor core. This evaluation supported plant procedural changes to allow movement of the UGS and CEA to the refueling cavity area where the CEA could then be removed.

Normally the UGS is removed and placed in the lower cavity with the CEAs remaining in the reactor core. In the above configuration, the CEA attached to the UGS could not be reinserted into a fuel assembly and thus had to be removed with the UGS. The concern with moving the UGS with the hanging CEA is the potential for dropping the CEA onto the top of the core. The evaluated fuel handling accident for the plant considers the drop of a fuel assembly itself (>1000 lbs.) onto other fuel assemblies. By comparison, the CEA weighs <100 lbs. If the CEA were to fall into the vessel, it would impact the top of the fuel assemblies, which are robust steel structures which include the hold down springs and upper end fittings. It is unlikely that these structures would be damaged should the CEA drop. Thus, these structures afford significant protection to the underlying fuel rods.

STEAM GENERATOR REPLACEMENT PROJECT - 10 CFR 50.59 SAFETY EVALUATION FOR CONTAINMENT POST MODIFICATION TESTING REQUIREMENTS

Summary:

This safety evaluation is associated with the 1997 steam generator replacement project and provides the post modification testing requirements related to containment vessel integrity.

Containment ingress and egress for the replacement and old steam generators was done via the construction hatch. The construction hatch containment penetration is a 28' diameter steel cylinder with a welded enclosure cap (hatch cover) located inside containment. The removal and reinstallation of the hatch cover was performed in accordance with the requirements of the plant's ASME Section XI Repair and Replacement Program. Removal and reinstallation of the cover involved a number of welding and NDE criteria, including pressure testing in accordance with 10 CFR 50, Appendix J. Originally, a periodic integrated leak rate test (ILRT) was to be performed which would have served to satisfy the leakage testing requirements; however, because recent changes to the Technical Specifications to adopt 10 CFR 50, Appendix J, Option B resulted in the ILRT date being deferred until the year 2003, an acceptable testing alternative was required.

The pressure test and Appendix J requirements were satisfied by the performance of a pneumatic test of the construction hatch-to-cover weld joint. This test was accomplished by the construction of a test chamber around the joint. The chamber was pressurized to the containment vessel internal design pressure of 39.6 psig and the weld joint was bubble tested to demonstrate zero leakage.

The secondary side of the replacement steam generators received a hydrostatic test at a pressure approximately 25 times greater than the ILRT pressure. This test, although in a reverse direction from an ILRT, satisfied the Appendix J requirements.

SAFETY EVALUATION PSL-ENG-SENS-97-087
REVISION 0

REACTOR COOLANT LEAKAGE DETECTION WITH THE CONTAINMENT RADIATION MONITORS

Summary:

This safety evaluation resolves discrepancies in the UFSAR description related to the use of the containment atmospheric radiation monitors for reactor coolant system (RCS) leakage detection. The discrepancies involve the following (actual practice versus UFSAR description): 1) the use of a 2x background alarm set point versus a 10% scale deflection as an indication of a step increase in RCS leakage; 2) the use of anisokinetic sampling versus isokinetic sampling; and 3) the use of volumetric flow indication versus mass flow indication.

Each of the above discrepancies were evaluated and found to be insignificant with respect to the ability of the system to perform its design basis function of RCS leakage detection. The alarm set point of 2x background is a more practical indication of RCS leakage than is a 10% meter deflection since, with zero fuel leakage, normal meter variations (i.e., background variations) typically exceed ±10%. Isokinetic sampling of particulates was determined not to be necessary based on the small particulate sizes involved. Volumetric flow indication versus mass flow indication is an insignificant difference.

SAFETY EVALUATION PSL-ENG-SENS-97-088
REVISION 1

ESF TESTING DURING LOW WATER LEVEL OPERATION

Summary:

This safety evaluation justifies the acceptability of conducting engineered safeguards (ESF) testing with the plant in Mode 6 with a low water level condition or in Mode 5 with reactor coolant loops not filled. Testing has traditionally been performed with the plant in Mode 6 with greater than 23 feet of water above the top of the fuel.

The concern associated with ESF testing with a reduced reactor coolant system water level is the operability of the shutdown cooling (SDC) system. In the proposed modes, Technical Specifications require two operable loops of shutdown cooling, with one loop in operation. The loop not in operation is considered to be in "standby." This safety evaluation uses NRC Generic Letter 91-18 for guidance and closely analyzes SDC operability during the ESF testing. The analysis concludes that SDC operability can be maintained during the testing. The evaluation provides several plant restrictions, including a requirement to verify that each loop of SDC has been operable and in operation prior to the initiation of testing. Also, as a prudent measure, the ESF testing is to be performed late in the refueling outage following core reload; this minimizes the core decay heat load and increases the time to boil on a loss of SDC. Should any loop of SDC become inoperable for any reason, ESF testing shall be suspended until two SDC loops have been returned to operable status.

SAFETY EVALUATION PSL-ENG-SEMS-97-089
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR MAIN FEEDWATER ISOLATION VALVE MV-09-7 AND MV-09-8 STROKE TIME CHANGE

Summary:

Plant Change/Modification 97071 modified the Limitorque actuator gearsets and thermal overloads for main feedwater isolation valves MV-09-7 and MV-09-8 to increase the torque capability for these valves. Replacing the gearsets also increased the valve stroke time from 42 to 55.39 seconds. The subject valves have a safety function of closing upon a main steam isolation signal (MSIS) or a safety injection actuation signal (SIAS). The stroke time of 42 seconds is stated in the UFSAR. The UFSAR also includes a statement which implies the accident analysis assumes a 60 second ramp during valve closure. The specifics of the type of "ramp" assumed could not be determined from the containment analysis. Note that the subject valves are gate valves and the flow profile over the valve stroke is non-linear.

This safety evaluation addresses the acceptability of increasing the stroke time and revising the UFSAR to delete the specific stroke time and to only state the required stroke time. The UFSAR statement regarding ramped flow is also deleted. The Unit 1 fuel vendor, Siemens, was contacted and a review of the feedwater addition assumptions for all UFSAR Chapter 15 accident analyses was performed to determine the effects of adding feedwater over a 60 second period. Since the exact flow profile for the feedwater isolation valves was unknown, Siemens assumed a constant feedwater flow addition for the entire 60 seconds. Siemens concluded the analyses of record bound operation for Cycle 15 with the assumption of 60 seconds of feedwater addition. Thus the increase in valve stroke time and the deletion of the "ramp" statement is acceptable.

SAFETY EVALUATION PSL-ENG-SEMS-97-092
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR UFSAR COMBUSTIBLE LOADING UPDATE FOR UNIT 1

Summary:

This safety evaluation revises the UFSAR based on an analysis of the combustible loading increase in various fire zones due to Thermo-Lag fire barrier material and other miscellaneous materials. NRC Information Notice 92-82 and subsequent related documents have identified Thermo-Lag as a combustible material. As such, the UFSAR Appendix 9.5A fire hazards analysis requires revision to reflect the revised combustible loading.

The fire protection program is designed to maintain the ability to perform safe shutdown functions and thus minimize radioactive releases to the environment. It reflects good fire protection engineering practice and is guided by plant fire hazard analyses and by credible fire postulations. 10 CFR 50, Appendix R provides the basis for the methodology used to establish the acceptability of a fire hazard analysis review.

A formal fire hazards analysis was performed for each of the affected fire zones. Conservative caloric values based on accepted industry literature and specific manufacturer's data were adopted for the calculation of combustible loadings. In each case (i.e., for each fire zone), the fire hazards analysis concluded that adequate protection is provided, assuring continued availability of redundant safe shutdown equipment and components.

SAFETY EVALUATION PSL-ENG-SECS-97-095
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR LIFTING AND HANDLING OF THE CCW 1B HX "WEST CHANNEL HEAD"

Summary:

In order to gain full access to the component cooling water (CCW) heat exchanger tube sheets during the 1997 refueling outage, Maintenance will be removing the east and west channel heads. Removal of the west channel head from the "B" heat exchanger will result in a heavy load lift over operating safety related "A" train cooling water piping. A heavy load is considered a load, including the crane hook, which weighs more than 1380 lbs. The channel head weighs approximately 6700 lbs. This evaluation addresses the adequacy of the rigging and load path to be used for the removal and reinstallation of the channel head. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines for the lifting and handling of heavy loads.

In preparation for the CCW heat exchanger maintenance, a barrier was designed for the protection of the "A" train piping from a load drop of the subject channel head. Calculation CIV-407 evaluated this barrier and found it acceptable. Per NUREG 0612, a load drop analysis precludes the need for redundant slings or lifting devices. Additionally, in order to meet the intent of NUREG-0612 the rigging selected shall have a rated capacity of twice the load to be lifted.

SAFETY EVALUATION PSL-ENG-SENS-97-098
REVISION 0

USE OF THE FIRE MAIN TO SUPPLY CIRCULATING WATER PUMP BACKUP LUBE WATER AND THE CONDENSER TUBE CLEANING SYSTEM

Summary:

Although not required for nuclear safety, the service water (SW) system provides water to the circulating water (CW) pumps' backup lube water system and to the condenser tube cleaning system (CTCS) for various seals. Since these non-safety uses of SW are important to plant operation, interruption of SW flow is undesirable. This evaluation allows the use of the fire main system, via wye connections to local hydrants, to provide water for the specific use of the CW pumps' backup lube water system and for the CTCS during periods of SW system unavailability.

During this time the fire main, normally pressurized via the domestic water pumps, will be pressurized via a temporary pump approved by Engineering. The temporary pump suction and discharge shall be connected to existing portable fire pump connections.

The city water storage tanks (CWSTs) provide the water source for the fire protection system. With the system modified as described above, the CWSTs continue to function as designed. The level alarm and automatic makeup capabilities are unaffected by the SW alignment. Thus, there is no credible threat to the water volume reserved and required for fire mitigation. Failure of the temporary pump or related hoses/lines will be readily identifiable by the continuous local monitoring which will be required. There is no effect on the operation of the main fire pumps.

All mitigation capabilities of the fire protection system remain unchanged by the proposed alignment. The maximum anticipated usage of approximately 54 gpm is insignificant with respect to the fire pumps' rated capacity of 2500 gpm and required flow rate of 2350 gpm.

SAFETY EVALUATION PSL-ENG-SENS-97-099
REVISION 0

CHANGE IN CEA RATE OF MOVEMENT

Summary:

This safety evaluation revises the UFSAR to correct the rate of control element assembly (CEA) movement. The UFSAR describes a rate of movement of 40 inches per minute. The equipment originally installed in the control element drive motor control system (CEDMCS) was capable of a rate of up to 40 inches per minute; however, a review of the vendor technical manual concluded the rate has always been set at 30 inches per minute. The CEDMCS is a non-safety system used to control the motion of the CEAs. The safety function of the CEAs is to drop into the core upon a reactor trip signal; this function is not affected. No physical changes are being made to the plant as a result of this evaluation.

SAFETY EVALUATION PSL-ENG-SEMS-97-100
REVISION 1

VALVE V3805 & V3810 QUALITY GROUP A TEMPORARY BOUNDARY CHANGE 10 CFR 50.59 EVALUATION

Summary:

Valves V3805 and V3810 are drain valves located on the safety injection header outside of containment. These valves normally provide the Quality Group "A" to Quality Group "D" class break. Seat leakage was noted on both of these valves when the downstream flange was removed in support of system flushing. This safety evaluation temporarily relocates the "A" to "D" class break to the downstream flange. This temporary alteration allows continued plant operation without exceeding allowable leakage limits.

A review of design documents shows the downstream piping and flange to have been designed and fabricated to Safety Class 1 requirements. Additionally, the subject piping and flanges were originally shown to be within the Quality Group "A" boundary; however, the class break was subsequently changed to be consistent with other design documents. A review of maintenance records could not find that the subject valves, piping or flanges have ever been repaired or replaced. As a part of this evaluation, the existing blind flanges and bolting are to be verified to meet Quality Group "A" requirements. Since the downstream piping and blind flanges meet the requirements for Quality Group "A," the temporary relocation of the class break is acceptable.

SAFETY EVALUATION PSL-ENG-SEMS-97-103
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR SINGLE ELEMENT CEA EXTENSION SHAFT ALTERNATE REPLACEMENT

Summary:

This safety evaluation justifies the installation of single element control element assembly (CEA) extension shafts with straight-latching indication slots as opposed to the older extension shafts which have slots shaped like "7"s. The newer straight-latching design is considered superior to the existing design. This change is necessary since the "7" slot design is no longer made. The straight slot design meets all of the requirements for positioning CEAs and has been successfully used on the duel element CEA extension shafts at St. Lucie and other Combustion Engineering designed reactors.

SAFETY EVALUATION PSL-ENG-SEMS-97-106
REVISION 1

10CFR50.59 SAFETY EVALUATION FOR INSTALLATION OF LEAK CHANNEL ON CONSTRUCTION HATCH WELD

Summary:

This evaluation addresses installation of a leak chase channel covering the Unit 1 construction hatch weld to facilitate pneumatic post-modification leak rate testing. The containment construction hatch was temporarily removed in support of steam generator replacement activities.

This evaluation demonstrates the leak chase channel to be welded over the containment construction hatch cover girth weld can be installed and used to perform the post-modification soap bubble testing addressed in safety evaluation PSL-ENG-SEMS-97-085.

The leak channel is constructed of 12 gage carbon steel. The channel will be installed in multiple sections fitted over the containment vessel construction hatch weld for the full circumference and welded in place using fillet welds on both sides of the channel and a seam weld (or splice plate) at the sections. The leak channel will have 3 test points (at 3:00, 6:00 and 9:00). A test assembly was constructed incorporating at least one seam weld. The test assembly was pressure tested (using water and air) to 66 psig and held for a minimum of 10 minutes. There was no deformation of the channel or leakage from the welds. Based upon the pressure test, the material thickness and weld sizes are considered adequate for the minimum 39.6 psig pressure test of the containment vessel construction hatch weld.

The installation of the leak channel is considered a non-structural attachment and meets ASME Section III, Subsection NE-4435.

SAFETY EVALUATION PSL-ENG-SENS-97-107
REVISION 1

EVALUATION OF TEMPORARY LEAD SHIELDING FOR REACTOR CAVITY DRAIN LINE

Summary:

Following the draining of the refueling cavity in preparation for the startup of Cycle 15 it was determined that a radiation hot spot existed in the reactor cavity drain line. The high radiation level (>1000 Rem/hour) is due to the presence of a small piece of foreign material stuck inside a valve in this line. Efforts to remove the debris by flushing have been unsuccessful. As a result, it is desired to install temporary lead shielding on portions of line 3-CS-51 and 3-CS-511 to reduce the area dose rate until the foreign material can be removed during the next refueling outage. A flexible lead shielding product will be installed over approximately 9 linear feet of piping. This evaluation considered the implications of leaving the temporary shielding in place during power operations.

The total weight of the lead shielding in question is approximately 500 pounds. Engineering has evaluated the proposed configuration and concluded that leaving the shielding draped on the pipe will have no adverse impact on the seismic qualification and structural integrity of the piping for design basis loads in accordance with ASME Section III, Class 2 and the UFSAR. The supports were also reviewed and found to be adequate for the design basis loads. An Engineering walkdown established that no safety related systems or components are located below the pipes; therefore, lead shielding will not have any seismic interaction with any safety related systems or components.

SAFETY EVALUATION PSL-ENG-SEFJ-98-007
REVISION 1

FSAR UPDATES FOR INCORPORATING SMALL BREAK LOCA RE-ANALYSES WITH ASYMMETRIC HPSI FLOWS

Summary:

This safety evaluation justifies a revision to the UFSAR to incorporate the revised small break LOCA (SBLOCA) analysis performed by Siemens Power Corporation (SPC) for the effects of safety injection loop asymmetries. The loop asymmetries have been determined to result in asymmetric high pressure safety injection flows in the cold leg injection lines. The re-analysis resulted in a SBLOCA peak cladding temperature increase from 1804°F to 1953°F and the limiting break size shifted from 0.10 ft^2 to 0.05 ft^2. The SBLOCA re-analysis performed for the loop asymmetry effects has been found to meet all the 10 CFR 50.46 acceptance criteria. Analysis results were reported to the NRC in December, 1997 via 10 CFR 50.46 30-day report.

SAFETY EVALUATION PSL-FPER-98-009
REVISION 0

ST. LUCIE UNIT 1 FIRE PROTECTION EVALUATION FOR LACK OF FIRE DOORS BETWEEN FIRE AREA E/ZONE 41 HOLD UP TANKS AND FIRE AREA C/ZONE 54 LAUNDRY AND DECONTAMINATION AREA

Summary:

An FPL Quality Assurance audit identified a concern related to an NRC exemption from 10 CFR 50, Appendix R requirements. The particular exemption is related to the lack of fire doors located between Fire Area C - Zone 54 (laundry and decontamination room) and Fire Area E - Zone 41 (hold up tanks). The original exemption indicated the combustible loading was negligible in Fire Zone 41; however, it did not address the combustible loading in Fire Zone 54 or its affects on adjacent Fire Zone 41. This evaluation provides an analysis of the fire hazards associated with the above areas and concludes the basis for the Appendix R exemption remains valid.

The evaluation reviewed the fire boundary between the subject fire areas in order to determine if the boundary could withstand the hazards associated with the area in accordance with NRC Generic Letter 86-10, Implementation of Fire Protection Requirements. The analysis considered the physical configuration and fire protection features provided for Fire Zone 54, the configuration of the opening from Fire Zone 54 to Fire Zone 41, current activities associated with Fire Zone 54, redundant safe shutdown circuits and components, and combustible loading in Fire Zone 54.

SAFETY EVALUATION PSL-ENG-SEMS-98-009
REVISION 0

ECCS PUMP NPSH FSAR DISCREPANCIES

Summary:

This evaluation addresses discrepancies identified between the Unit 1 UFSAR, Design Basis Documents and design calculations regarding emergency core cooling system (ECCS) and containment spray system (CSS) pump net positive suction head (NPSH) values. These discrepancies were noted during the preparation of the St. Lucie plant response to NRC Generic Letter 97-04, Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps. This safety evaluation provides revisions to the Unit 1 UFSAR and Design Basis Documents to correct these discrepancies.

The discrepancies pertain to: (1) "piggy back" or series operation of the containment spray pump with the high pressure safety injection (HPSI) pump in the recirculation mode (this mode of operation is not uniquely modeled in UFSAR Table 6.2-9A, NPSH Calculations for ECCS Pumps); (2) simultaneous hot leg and cold leg injection (UFSAR Table 6.2-9A does not include NPSH data for simultaneous hot leg and cold leg injection utilizing LPSI flows greater than 250 gpm); and (3) several discrepancies in UFSAR Table 6.2-9A with respect to the calculation of record from which it was developed.

Calculation PSL-1FSM-98-001 was performed to verify the NPSH available for the subject pumps. In all cases, the NPSH available exceeded the NPSH required. No changes were required to the plant or to plant operating procedures as a result of this evaluation.

SAFETY EVALUATION PSL-ENG-SENS-98-025
REVISION 0

CEA POSITION INDICATION DISPLAY POWER SUPPLY

Summary:

Two independent non-Class 1E systems of control element assembly (CEA) position indication are provided on the control boards. The systems are the pulse counting system and the reed switch system. In addition to these systems, there is a core mimic display which provides CEA travel limit information. The pulse counting system infers CEA position by maintaining a record of the raise/lower pulses; the reed switch system utilizes magnetically operated reed switch transmitters. This latter system provides two different methods of display: a cathode ray tube (CRT) and a digital meter selectable for each CEA. A UFSAR review determined that a statement concerning the originating power source for each of the reed switch system displays is not accurate and needs to be revised. This evaluation justifies a revision to the UFSAR and eliminates the discrepancy with the field.

The UFSAR states the originating power sources of the two reed switch system displays is battery 1B and battery 1A and that these power supplies have been selected to achieve maximum independence. The actual power supplies, although emergency diesel backed, are not from independent buses. This discrepancy has been evaluated and found to be insignificant with respect to the system's ability to perform its design basis function.

Compliance with Regulatory Guide 1.97 (RG 1.97) helps to ensure availability of required plant variables. Per RG 1.97, the only CEA position indication required is "full in or not full in" and is listed as Type B, Category 3. A review of RG 1.97 Table 1 shows there is no power source requirement for Category 3 variables. Therefore, the detailed statements made in the UFSAR concerning power source will be removed. No physical plant or procedure changes are being made as a result of this evaluation.

SHARED SYSTEMS AND INTERCONNECTIONS BETWEEN UNIT 1 AND UNIT 2

Summary:

The Unit 1 UFSAR description states that all interconnections have locked closed valves to isolate each unit. Also, the Unit 1 UFSAR lists several systems and components that are interlocked, but not normally shared. Contrary to the UFSAR description, some of the interlocked systems do not have locked closed valves and are frequently or continuously used by both units during plant operations (e.g., hydrogen, nitrogen and makeup demineralizer regeneration systems). The Unit 2 UFSAR states that locked closed isolation valves are provided for the liquid waste management (hold-up tanks and aerated waste storage tank) and station service air interconnections; however, no locks are provided for the AWST interconnection and only one valve of the service air system is locked closed. This evaluation provides the justification for revising the Unit 1 and Unit 2 UFSARs to resolve the UFSAR discrepancies.

10 CFR 50 General Design Criteria 5 (GDC-5) provides general criteria for the sharing of structures systems or components (SSCs). GDC-5 prohibits the sharing of SSCs, which are important to safety, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions. The systems affected by this evaluation are not considered important to safety and are thus not required to have locked closed isolation valves. The interconnected systems which are considered important to safety are not normally used by both units and are provided with suitable isolation devices.

SAFETY EVALUATION PSL-ENG-SENS-98-032
REVISION 0

UFSAR CHANGES TO THE RCS LEAK DETECTION DESCRIPTION

Summary:

This evaluation was written to resolve discrepancies identified in UFSAR Table 5.2-11, Reactor Coolant Leak Detection Sensitivity. This UFSAR table provides a general overview of the means of identifying reactor coolant system (RCS) leakage via the monitoring of various plant parameters. The discrepancies corrected by this evaluation pertain to quench tank water level and safety injection tank (SIT) water level instrumentation. The affected parameters involve the specific values provided in the table for instrument range, normal reading, and rate of change for a 1 gpm RCS leak.

The corrected parameters are within the UFSAR Section 5.2.4.6 acceptance criteria of being able to detect a 1 gpm RCS leak within 24 hours. As such, the UFSAR changes are acceptable. No changes are being made to the field or to plant procedures as a result of this evaluation.

SAFETY EVALUATION PSL-ENG-SENS-98-034
REVISION 0

10CFR50.59 SAFETY EVALUATION FOR PRESSURIZER LEVEL CONTROL IN AUTO DURING PLANT HEATUP - UFSAR CHANGES

Summary:

UFSAR Section 9.3.4.2.3 describes the manual control of pressurizer water level using the letdown and backpressure control valves during plant heatup. Current plant procedures instruct operators to control these valves in the automatic mode. This evaluation provides the justification to revise the UFSAR to delete the manual control requirement.

The pressurizer level control (PLC) system is not credited in the prevention or mitigation of postulated accidents, nor is it required for plant cooldown. There is no regulatory requirement which would preclude the operation of the letdown and backpressure control valves in the automatic mode. Operation of the PLC system in automatic provides operational flexibility during the heatup process and helps to minimize the volume of radioactive liquid waste produced during heatup. There are no analyzed accidents which are affected by this evaluation.

SAFETY EVALUATION PSL-ENG-SENS-98-036
REVISION 0

MARINE GROWTH CONTROL PROGRAM - UFSAR CHANGES

Summary:

This evaluation provides the justification for deleting a statement in the UFSAR describing an ongoing program for controlling marine growth in the ocean water intake pipeline.

Prior to the construction of Unit 2 there was a concern related to the biofouling of the single ocean intake and ocean discharge pipelines. The ocean intake provides cooling water (ultimate heat sink) for the plant. The concern was that biofouling of the intake would restrict the flow of cooling water to the plant; similarly, biofouling of the ocean discharge pipeline would restrict flow, resulting in increased discharge canal levels. Construction of Unit 2 included the addition of a second intake pipeline and a second discharge pipeline. The additional pipelines, coupled with the operation of Unit 2 (i.e., increased supply of warm water in the discharge pipelines) obviated the need for any further marine growth "program." Additionally, plant experience has shown that pipe flow velocities limit the extent of marine growth such that a terminal growth length is reached (at which point the old growth breaks off as new growth develops).

SAFETY EVALUATION PSL-ENG-SENS-98-037
REVISION 0

UFSAR CHANGES TO SYSTEM PERFORMANCE MONITORING DESCRIPTIONS

Summary:

This evaluation provides the justification to revise the UFSAR in order to resolve a discrepancy regarding performance monitoring of the spent fuel pool (SFP) cooling system. Specifically, the UFSAR notes that "Data will be taken and periodic visual inspections and preventive maintenance will be conducted, as necessary. This periodic inspection will also confirm heat transfer capabilities, purification efficiency and component differential pressures." Contrary to the UFSAR, there are no specific inspections or preventive maintenance activities for the above.

The SFP cooling and purification systems are non-safety, non-seismically designed systems. SFP purification is assured via the required periodic chemical analysis of the SFP. SFP cooling system performance is assured via the control room monitoring of fuel pool temperature, fuel pool cooling pump status and fuel pool level. Any heat load which would cause temperatures to increase above the alarm set point would be investigated and corrected as appropriate. Additionally, both sides of the SFP heat exchanger are exposed to clean, chemically treated fluid which tends to suppress heat exchanger fouling. The fouling factor used in the heat exchanger analysis was extremely conservative.

This safety evaluation demonstrates that sufficient basis exists for the revision of the UFSAR description of SFP performance monitoring with respect to data taking, preventive maintenance and periodic visual inspections.

SECTION 3

RELOAD SAFETY EVALUATIONS

PLANT CHANGE/MODIFICATION 97055

ST. LUCIE UNIT 1 CYCLE 15 RELOAD

Summary:

This engineering package provided the reload core design for St. Lucie Unit 1 Cycle 15. The Cycle 15 core is designed for a cycle length of approximately 14,600 EFPH, based on a Cycle 14 length of 10,577 EFPH. The Cycle 15 design covers a cycle length between 14,383 and 14,832 EFPH, depending upon a Cycle 14 exposure of 10,960 to 10,194 EFPH. The Cycle 15 safety analysis supports an end of cycle T_{ave} coastdown with a maximum reduction in primary coolant temperature of 26^{0}F at full power.

The primary design change to the core for Cycle 15 is the replacement of 76 irradiated fuel assemblies with fresh fuel assemblies. Eight of the 76 fresh fuel assemblies are Batch T fuel that was not used in Cycle 14. The remaining 68 fresh fuel assemblies are Batch U. The core retains 53 irradiated fuel assemblies from the Cycle 13 reload and all 88 fuel assemblies from the Cycle 14 reload. All assemblies in the Cycle 15 reload are of the debris resistant long end cap design. The mechanical design of the Batch U fuel is the same as the Batch T and Batch S reload fuel.

The safety analysis of this design was performed by Seimens Power Corporation (SPC) and by Florida Power and Light Co. All analyses to support this reload were performed with the assumption of steam generator tube plugging not to exceed 15% total with ± 7% asymmetry. The analyses are consistent with the amended Technical Specifications minimum reactor coolant system flow rate of 345,000 gpm. The effects of the replacement steam generators have been evaluated in the reload safety analysis and Steam Generator Equivalency Report. Except for the loss of feedwater transient, no other transients needed to be re-analyzed to account for differences in steam generator parameters.