ST. LUCIE UNIT 2
DOCKET NUMBER 50-389
CHANGES, TESTS AND EXPERIMENTS
MADE AS ALLOWED BY 10 CFR 50.59
FOR THE PERIOD OF
JANUARY 6, 1996 THROUGH MAY 25, 1997

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
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) and 50.71(e)(4). This report is intended to meet this requirement for the period of January 6, 1996, through May 25, 1997.

This report is divided into three (3) sections; the first, changes to the facility as described in the Updated Final Safety Analysis Report (FSAR) performed by a Plant Change/Modification (PC/M); the second, changes to the facility or procedures as described in the Updated FSAR not performed by a PC/M and tests and experiments not described in the Updated FSAR; 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 84099

ICW PUMP MATERIAL CHANGES

Summary

This modification consists of allowing the use of ASTM-A-479, Type XM-19 austenitic stainless steel (Nitronics 50) in addition to the original material, ASTM B-164, for the ICW pumps' shafts, couplings and keys. The new material offers superior pitting/crevice corrosion resistance and should provide improved material performance.

This modification provides functionally equivalent components and does not alter the performance characteristics or operation of the ICW pumps.

PLANT CHANGE/MODIFICATION 85153

RAB VENTILATION SYSTEM DAMPER MODIFICATION

Summary

Dampers GD-3 and GD-4 serve as backdraft dampers for reactor auxiliary building (RAB) main supply fans HVS-4A and HVS-4B. Each of these gravity driven dampers closes to prevent free-wheeling of its associated fan when that fan is secured and the other fan is operating. This PC/M modifies the dampers to install an adjustable stop on the damper linkage to prevent the horizontally mounted damper from opening beyond the 90 degree full open position. Additionally, damper mullions are being modified to facilitate maintenance access to the bearings.

The use of a 90 degree stop is standard practice for horizontally mounted dampers. The stop provides added assurance the dampers will close when required. The mullion modifications improve maintenance access to damper bearings, thus facilitating the ability to lubricate and maintain the bearings.

These changes do not alter the performance characteristics of the dampers and are expected to provide improved damper reliability.

PLANT CHANGE/MODIFICATION 91463

RANGE INCREASE FOR FEEDWATER AND MAIN STEAM INDICATORS AND RECORDERS

Summary

This modification expands the indicated range for feedwater and main steam flow indicators and recorders. The original instruments were provided with a range of 0-6E+6 lbm/hr. Although this original range enveloped the stretch power feedwater and main steam flows (5,893,450 lbm/hr), the devices were operating close to their maximum upper limit. This PC/M increases the range of the subject instruments to 0-7E+6 lbm/hr.

As a result of the range change, the Feedwater Flow/Steam Flow controller proportional bands were adjusted and the Quick Opening Time Constant settings for the Steam Bypass and Control System were adjusted to compensate for the range change. System transmitters were also recalibrated. There is no net change to system operation or performance.

As a part of the PC/M, ABB-CE evaluated the changes and provided recommended coefficient settings to ensure that systems will provide the same performance and meet the same requirements as the original design.

PLANT CHANGE/MODIFICATION 93191

INSTALLATION OF ANPS WORK STATION IN CONTROL ROOM

Summary

This modification adds a customized work station in the control room for the Assistant Nuclear Plant Supervisor (ANPS). The work station was installed in proximity to the Reactor Turbine Generator Boards (RTGBs) and required the relocation of four computer printers. Installation also improves RTGB access control.

This change was evaluated to ensure that no adverse interactions with safety related equipment were created. A human factors review concluded there was no adverse impact associated with the change.

PLANT CHANGE/MODIFICATION 94093

ICW AND CW SYSTEM INSTRUMENT IMPULSE LINE MODIFICATION

Summary

This modification allows for the use of alternate materials (6 Moly SS, Titanium, and Hastelloy C-276) for a portion of the instrument impulse lines associated with the Intake Cooling Water (ICW) and Circulating Water (CW) systems. This modification also allows the use of tubing in place of piping. The alternate materials and use of tubing was evaluated and found to be an acceptable equivalent substitution. All applicable design requirements are satisfied and ICW and CW system operation is not affected by the changes.

PLANT CHANGE/MODIFICATION 95028

INSTALLATION OF AN ACOUSTIC FEEDWATER FLOW METER

Summary

This modification installs an acoustic flow meter to monitor feedwater flow. The new flow meter provides improved performance over the existing feedwater flow venturis which are not easily removed and cleaned and are susceptible to fouling. Feedwater flow venturi fouling results in conservative power measurements which result in plant operating inefficiencies.

Safety Evaluation JPN-PSL-SEIP-95-031 was issued to evaluate and justify the temporary use of the LEFM along with the existing venturi. Several months of data were obtained and evaluated. This PC/M provides the details for permanent installation of the LEFM.

This modification does not change any mechanical components of the feedwater system. The original feedwater flow measurement system remains installed as a redundant system. Although secondary side thermohydraulic parameters have changed, the changes were evaluated to be within design limits. LEFM accuracy testing was performed at Alden Laboratories and was formally documented in Test Report MPR-1576. The results of this testing were included in calculation PSL-BFJI-94-001, PSL 1&2 Verification of Secondary Calorimetric Uncertainties, which concluded the LEFM will not increase the uncertainty of the power calculation beyond the 2% limit at full power and that it provides uncertainties less than those analyzed under JPN-PSL-SEFJ-94-016, Impact of Increased Power Measurement Uncertainty on Safety and Setpoint Analyses for PSL 1 Cycle 12 and PSL 2 Cycle 8, for power levels less than 100%.

The LEFM software underwent a formal software verification and validation.

PLANT CHANGE/MODIFICATION 95248

INSTALLATION OF VIBRATION MONITORING PROBES ON RCP 2A2

Summary

This modification installs a set of vibration monitoring probes on the lower motor casing of reactor coolant pump (RCP) 2A2 at a previously abandoned probe location. The new probes are of the same make and model and are installed in the same location and configuration as the set of original probes which were relocated to the top of the mechanical seal via previous modification. The new probes are seismically mounted and are used for monitoring and troubleshooting purposes. There is no impact on the operation or qualification of the RCP.

PLANT CHANGE/MODIFICATION 96012

RCGVS PRESSURE INDICATOR CONTROL LOGIC MODIFICATION

Summary

This modification temporarily changes the control logic of rector coolant gas vent system (RCGVS) pressure indicator PIA-1140 to annunciate on low pressure instead of high pressure.

PIA-1140 is located between the RCGVS primary and secondary isolation valves (isolation from the reactor coolant system) and is designed to provide operators with an indication of leakage past one or more of the primary isolation valves. As a result of leakage past one or more of the primary isolation valves, the annunciator is in constant alarm. Changing the control logic will clear the alarm and will provide an alarm on a low pressure condition. The new alarm is useful to operators in that it would be indicative of a leak past one or more of the secondary valves. This modification is effective until the Cycle 10 refueling outage when the RCGVS isolation valves are scheduled for replacement.

Note: The RCGVS isolation valves were replaced and the control logic was restored to its previous configuration during the Cycle 10 refueling outage.

PLANT CHANGE/MODIFICATION 96099

DRAIN VALVES FOR STEAM SUPPLY TO THE AFW PUMP 2C

Summary

This modification provides for system improvements by increasing the reliability of the steam supply system for the 2C auxiliary feedwater pump turbine. This modification relocated the warmup line inlet to allow better draining of the steam supply line and reduce operating transients when opening the steam admission valves. Additionally, upstream isolation valves were installed in each warmup line to enhance system maintenance capabilities. The modification conforms to the same quality group, materials, and classification as the system. There is no adverse impact on the operation or qualification of the 2C AFW pump.

PLANT CHANGE/MODIFICATION 96124

REACTOR VESSEL O-RING REPLACEMENT

Summary

This modification replaces the existing reactor vessel O-rings with a spring energized O-ring design recommended by Helicoflex (the original equipment manufacturer) and ABB-CE. The new design reduces the amount of honing and/or welding (and associated radiation doses) currently required to address minor surface anomalies of the vessel flange.

The new O-rings use metal strip silver jacketing in lieu of electroplated silver and they use an internal helical spring to provide sealing force during flange compression. This design has been successfully used in over 75 nuclear installations. This change provides a functionally equivalent design and has been evaluated by ABB-CE as being acceptable for use.

PLANT CHANGE/MODIFICATION 96133

ADDITION OF MANUAL ISOLATION VALVES TO DIESEL DRIVEN AIR COMPRESSOR

Summary

This modification installs normally open manual isolation valves in the discharge lines from the Emergency Diesel Generator (EDG) air start compressors to facilitate air compressor maintenance. Four isolation valves were installed downstream of the existing check valves and upstream of the sensing lines on the electric driven air compressors which are equipped with their own isolation valves. The valves do not interfere with over pressure protection or adversely affect functional operation of the system. This modification eliminates the practice of typically taking out of service both compressors in order to perform maintenance on one of the compressors. This modification allows the isolation of either air compressor in each train for maintenance and is considered a design enhancement. There is no adverse impact on the operation or qualification of the EDG.

PLANT CHANGE/MODIFICATION 96138

SHUTDOWN COOLING ISOLATION VALVE MODIFICATION

Summary

This PC/M modifies shutdown cooling system hot leg suction isolation valves V3651, V3652 and V3480 to prevent potential pressure locking. These valves were previously identified via evaluation JPN-PSL-SEMP-93-036 as being susceptible to pressure locking. This change is being performed to satisfy the requirements of NRC Generic Letter 95-07.

The modification drills a 3/16" hole on the upstream (reactor coolant system) side of each valve disks in order to provide venting of high pressure fluid from the valve bonnet area. This is a standard method of addressing pressure locking concerns and is endorsed in NUREG-1275 Vol. 9. The downstream seat of the valves is not been affected.

This modification does not adversely affect operation of the subject valves or the shutdown cooling system.

PLANT CHANGE/MODIFICATION 96143

THERMO-LAG RELATED MODIFICATIONS

Summary

This PC/M implements changes related to concerns over the performance of Thermo-lag 330-1 and the fire protection of various plant equipment. As a result of NRC concerns, FPL performed individual fire area evaluations to assess potential options (e.g., redundant equipment, conduit reroutes, manual actions, etc.) which would eliminate the need for protection of affected conduits.

This modification reroutes conduits protected with Thermo-lag 330-1. The rerouted conduits contain cables for the 480V power feed to battery charger 2A from motor control center 2A5; low voltage 125V dc power from bus 2A to RTGB-205 and RTGB-206; and low voltage 120V ac power from instrument bus 2MA-1 to isolation cabinet MA/SA for power distribution to RTGBs 201, 202, 203, 205 and 206. This change eliminates Thermo-lag surveillance and maintenance requirements associated with the affected conduit.

This modification also restores the protection provided to the pressurizer high-low discharge pressure interfaces by eliminating the protection requirement and operator action for the PORV block valves and adding fire protection to the PORV cables. Thus, operator manual actions have been replaced with a passive protection and are no longer needed for a fire in the electrical penetration rooms.

The changes provided by this PC/M are limited to the fire protection aspects of system design and do not affect the safety functions of the related plant equipment.

PLANT CHANGE/MODIFICATION 96153

QUICKLOC INCORE INSTRUMENT FLANGE DESIGN

Summary

This modification replaces the existing reactor instrument nozzle penetration and sealing system with the ABB-CENO designed Quickloc system. Included with this change is the replacement of 56 incore instrument assemblies which enter the vessel through the nozzle flanges.

The Quickloc modification replaces 10 incore instrument flanges which are used for incore instrument (ICI) and heated junction thermocouple (HJTC) probe insertion into the reactor vessel. These flanges serve as a part of the primary system pressure boundary. The new design reduces personnel radiation exposure by simplifying instrument nozzle assembly, disassembly and ICI maintenance. Reactor coolant system pressure boundary design requirements have been maintained.

The new ICI assemblies are compatible with the Quickloc design and are functionally equivalent to the original assemblies. There is no affect on the interface between the ICIs and the existing computer system (DDPS).

PLANT CHANGE/MODIFICATION 96159

CONDENSATE PUMP MECHANICAL SEAL VENTING

Summary

This modification provides a means of venting condensate pump mechanical seals prior to pump startup and provides a blowoff line on the mechanical seal inlet strainers to allow for on-line strainer cleaning. On-line strainer cleaning will increase condensate pump availability since the pumps will no longer have to be removed from service for this evolution. Existing vent valves will be used and manual blowoff valves will be added.

The condensate pumps are non-safety pumps. This change does not alter condensate pump design, function or performance characteristics. Although additional potential failure modes are introduced (e.g., inadvertently leaving a valve open), the benefits of the modification outweigh the consequences of such failures.

PLANT CHANGE/MODIFICATION 96161

STEAM GENERATOR TUBE PLUGS AND TUBE STAKES

Summary

This modification evaluates newly designed Combustion Engineering standard, extended and short mechanical steam generator (SG) tube plugs, standard welded tube plugs and standard and full length tube stakes. This change does not affect tube plugging limits which are established by the reload safety analysis.

The new plug designs are equivalent in form, fit and function to existing plug designs and the requirements of the original construction code of record have been satisfied. The plugs and stakes have been designed for use in CE designed SGs in accordance with ASME Code.

The only change in the standard mechanical plug is a taper dimension in the ID of the open end of the plug. The extended and short length plugs have slightly changed in length. The extended plug will accommodate a different location in the tube for sealing in the event the normal sealing area is damaged. The short plug can be used when there is an interference during installation. All plugs meet the same testing criteria. Additionally, the new design includes significantly more metal at the plug tip and thus does not require the antiwear collar when installed behind a tube stake.

Tube stake designs have been evaluated by ABB-CE to ensure the stake/tube assembly remains out of critical vibration frequencies during normal and abnormal conditions and to ensure compatability with tube plugs.

This PC/M also allows the grinding of the SG divider partition plate corner patch plate bolt heads in the event of an interference during plugging activities. The subject bolts are not loaded except for the initial tightening during fabrication. The bolts are not credited in any structural evaluations or calculations. ABB-CE has evaluated modification of the bolt heads and concludes it is acceptable to remove up to 50% of the bolt head area and still maintain adequate shear area.

PLANT CHANGE/MODIFICATION 96163

CONTAINMENT BUILDING RADIANT ENERGY SHIELD MODIFICATION

Summary

This modification removes Thermo-lag 330-1 wall-type radiant energy shields in containment and replaces them with 16 gauge stainless steel sheet metal panels. Additionally, 24 gauge stainless steel sheet metal will be used to cover four Thermo-lag covered conduits in containment. The replacement barriers have been evaluated to provide adequate protection and to have no impact on any safety related structures, systems or components.

The subject walls are being replaced as a result of NRC statements regarding the combustibility of Thermo-lag 330-1 and since the UFSAR and 10 CFR 50 Appendix R (Section III.G.2.f) require radiant energy shields to be constructed from non-combustible materials.

PLANT CHANGE/MODIFICATION 96167

BYPASSING THE 2A PURIFICATION FILTER

Summary

This modification addresses the bypassing (isolation) of the 2A purification filter by closing isolation valve V2360 and opening bypass valve V2355. Letdown flow will be routed around the 2A filter to the purification ion exchangers. This PC/M makes permanent the temporary change authorized via Temporary System Alteration (TSA) 2-96-028.

The change increases flow through the boronometer and eliminates the boronometer low flow alarm during periods when only one charging pump is in operation. Additionally, the change eliminates the need for disposal of highly radioactive filters.

Isolation of the 2A filter has no effect on the plant's safety analysis since no reduction in fission product or activation product inventory is credited to the filter. The purification ion exchangers will continue to remain in service, as will the 2B purification filter located downstream of the ion exchangers. Experience has shown that the ion exchangers are effective removers of particulates and eliminate the difficulty of disposing highly radioactive filter elements.

Plant chemistry specifications and limits on reactor coolant system specific activity have not changed. As a result, no increase in fuel cladding perforations will result from filter isolation. The 2A filter may be placed back in service in the event of unexpected fuel rod failures.

PLANT CHANGE/MODIFICATION 96174

REPLACEMENT OF RAB SUPPLY FAN HVS-4A WITH A NON-EQ MOTOR AND SHIELD WALL

Summary

HVS-4A is an environmentally qualified (EQ) fan motor which provides supply air for the reactor auxiliary building (RAB). The motor for this fan failed and was replaced with a non-EQ motor and a seismically mounted local shielding wall via Temporary System Alteration 2-96-24. This TSA configuration was evaluated and found acceptable via safety evaluation JPN-PSL-SEES-96-011, Revision 1. This modification provides the justification to make the TSA a permanent plant change.

The replacement motor was purchased as an equivalent Class 1E replacement with the exception of its radiation qualification. Because the new motor was not qualified for a radiation environment, a shielding wall was included as part of the motor replacement. This new shielding was evaluated as providing acceptable attenuation of expected post-accident radiation levels. In addition, the shielding wall was seismically analyzed to ensure its integrity is maintained and that there are no adverse interactions with adjacent equipment. The weight of the shielding wall was evaluated against allowable floor loads and found to be acceptable.

This PC/M was concluded to represent a functionally equivalent replacement for the original motor. The ability of the HVS-4A fan to meet its design requirements has not been compromised.

SECTION 2

SAFETY EVALUATIONS

SAFETY EVALUATION JPN-PSL-SEFJ-92-009
REVISION 0

UNITS 1 & 2
DETERMINATION OF CEA OPERATIONAL LIFE

Summary

The UFSARs for each unit assume a 10 year operational life for control element assemblies (CEAs). The CEA vendor, ABB-CE, clarified that this is a 10 calendar year limit which corresponds to approximately eight effective full power years. The 10 year limit is based on a different CEA design than that currently used at St. Lucie. This evaluation utilizes unit specific CEA inspection data along with data from a similar plant (Arkansas Nuclear One - Unit 2) to determine the operational life of the CEA design currently in use.

CEA operational life limits were determined by evaluating CEA performance against design criteria related to cladding wear, unrecoverable cladding strain limits, fast neutron fluence limit to preclude failure due to cladding material degradation, and CEA absorber material depletion. Current and future fuel management and CEA programmed repositioning strategies were also included in this evaluation.

As a result of this evaluation, CEA inspection and replacement strategies have been revised.

OPERATION WITH REDUCED PRESSURIZER HEATER CAPACITY

Summary

Revision 0 of this evaluation documents the acceptability of removing up to a total of six pressurizer heaters from service if needed; one proportional heater and five backup heaters for a total of 300 kW. This would leave a heater capacity of 1200 kW. The capability to safely shutdown the plant under natural circulation conditions is not adversely affected since the backup heaters required to provide pressure control under a loss of offsite power will be maintained at or above the technical specification limit of 150 kW per heater bank.

Revision 3 was issued to provide clarification regarding requirements for heater capacity.

SAFETY EVALUATION PSL-FPER-96-002
REVISION 1

FIRE PROTECTION EVALUATION AND UFSAR UPDATE
RELATED TO THERMO-LAG WALLS

Summary

This fire protection evaluation reviews the use of a Thermo-lag wall as a fire barrier and considers Thermo-lag as a combustible load in light of NRC Information Notices 92-082 and 95-027 and Generic Letter 86-10.

The UFSAR was revised as follows:

- plant conformance to NRC Branch Technical Position 9.5-1 was revised to note the acceptability of fire barriers which do not meet ASTM E-119 3-hour criteria, but have been evaluated and determined to provide adequate protection for redundant safe shutdown equipment and components;

- the fire rating of an auxiliary building Thermo-lag wall was changed from an ASTM E-119 3-hour rating to the as-built/tested fire rating; and

- various fire zone combustible loading values were updated based on the increased loading associated with the use of Thermo-lag.

SAFETY EVALUATION JPN-PSL-SENS-96-003
REVISION 0

UNITS 1 & 2
CHEMICAL AND VOLUME CONTROL SYSTEM OPERATION

Summary

This safety evaluation was issued to clarify the UFSAR wording with respect to operation of the chemical and volume control system (CVCS). The existing UFSAR wording predominantly described system operation in the automatic mode of operation, including the alignment of the charging pump suction to the volume control tank (VCT). Although the system was originally designed to be operated in automatic, it has traditionally been operated in manual because of a desire to manually control reactivity changes. This operational flexibility was designed into the system; however, it was not described in the UFSAR in any detail.

The UFSARs were revised to note the acceptability of system operation in the manual mode, including the ability to dilute or borate via direct injection through the charging pumps rather than via the VCT. The UFSAR Chapter 15 accident analyses were reviewed and determined to bound the UFSAR changes.

SAFETY EVALUATION JPN-PSL-SEMS-96-007
REVISION 0

UNITS 1 & 2
ADDITION OF MANUAL ISOLATION VALVES
IN THE REACTOR COOLANT GAS VENT SYSTEM

Summary

This safety evaluation documents the acceptability of installing three manually operated isolation valves in the reactor coolant gas vent system (RCGVS). The RCGVS interfaces with the reactor coolant system and provides a pressure boundary function.

The new valves allow the isolation of existing normally closed system solenoid operated vent valves in order to facilitate maintenance on the solenoid valves. The new manual valves are required to be maintained locked open, thereby ensuring proper operation of the RCGVS. In the event it becomes necessary to perform maintenance on one of the solenoid operated valves, the associated manual valve(s) may be closed. Per the evaluation, the appropriate RCGVS technical specification requirements must be complied with when isolating a portion of the system.

The manual valves were designed and fabricated in accordance with ASME Class 1 requirements and were purchased as nuclear grade components. The valves were reviewed for system compatibility, including valve pressure drop, materials and seismic design.

SAFETY EVALUATION PSL-FPER-96-010
REVISIONS 0 & 1

FIRE PROTECTION EVALUATION FOR THE CONTROL ROOM HVAC SYSTEM

Summary

This safety evaluation assesses the need for additional fire protection features since fire dampers are not present in HVAC ducts penetrating the fire wall between the HVS-5A & 5B fan room and the control room HVAC equipment room. The evaluation also allows the removal, if desired, of thermal wrap installed on the control room HVAC ducts located in the HVS-5A & 5B fan room.

This fire protection analysis considered the design and construction of fire area penetration in conjunction with system operating features and area combustible loads. The evaluation concludes that the existing barrier provides adequate separation and that neither the installation of fire dampers in the ventilation ducts nor the use of thermal wrap would augment or materially enhance the safety of the plant.

SAFETY EVALUATION JPN-PSL-SENP-96-019
REVISION 0

UFSAR CHANGES - PLANT HEATUP AND COOLDOWN

Summary

A review of the plant heatup and cooldown operating procedures identified some minor discrepancies between the procedures and the UFSAR. This safety evaluation evaluates these differences and provides the necessary UFSAR changes. No changes to plant procedures were required.

SAFETY EVALUATION JPN-PSL-SEFJ-96-019
REVISION 1

ANALYSIS TO ALLOW AN INCREASED NUMBER OF OUT OF SERVICE INCORE DETECTORS

Summary

This safety evaluation documents the results of a power peaking uncertainty analysis of the CECORE computer code which is used to monitor the core power distribution. At the time this was performed, 10 of 56 detector strings were failed. The UFSAR limit for failed detectors was no more than 25% of the total. The analysis quantified the impact of increasing the allowable number of failed detector strings on the uncertainty associated with the resulting power distributions obtained from the reduced number of detectors. The effect of increasing the allowed number of failed detectors from 25% to 46% was evaluated.

The evaluation concluded that the ability of the incore detector system to perform its intended functions remains unaffected.

Note: Incore detectors were replaced during the Cycle 10 refueling outage.

SAFETY EVALUATION JPN-PSL-SEFJ-96-022
REVISIONS 0 & 1

UNITS 1 & 2
EVALUATION OF THE BEACON CORE MONITORING SYSTEM

Summary

This safety evaluation qualifies the methodology and performance of the Best Estimate Analyzer for Core Operations - Nuclear (BEACON) system for core analysis at Units 1 & 2. This allows the use of BEACON as a replacement for the INPAX code (Unit 1) and the CECORE code (Unit 2).

BEACON is an advanced online 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 system provides core monitoring, core analysis and follow, and core predictions. The heart of the system is an NRC approved, three dimensional nodal code, ANC.

SAFETY EVALUATION JPN-PSL-SENP-96-022
REVISION 0

ELIMINATION OF THE PRESSURE RELIEF FUNCTION FOR THE REACTOR CAVITY PRESSURE RELIEF DAMPERS

Summary

The reactor cavity pressure relief dampers were designed to open and vent the lower portion of the reactor cavity in the event of a loss of coolant accident. This evaluation revises the design bases of the dampers to eliminate this pressure relief function based on NRC approval of the leak-before-break evaluation for St. Lucie Unit 2. As such, the dynamic effects associated with a hot or cold leg piping failure may be removed from the plant's design basis and the relief dampers are no longer required for pressure relief.

SAFETY EVALUATION JPN-PSL-SENS-96-033
REVISION 0

UNITS 1 & 2
WASTE GAS HOLDUP SYSTEM OPERATION

Summary

This safety evaluation clarifies UFSAR wording with respect to the permissible modes of operation for the waste gas holdup system. The UFSAR describes system operation with the flow stream routed from the gas surge tank to the waste gas compressors and the gas decay tanks for holdup prior to discharge through the plant vent. While this is an acceptable mode of operation and is the preferred mode of operation during periods of high reactor coolant fission product inventory, it is also acceptable, subject to the constraints provided by the Technical Specifications and the Offsite Dose Calculation Manual, to discharge gaseous waste directly to the plant vent.

Original plant design assumed normal operation with a certain amount of reactor coolant activity. Under such conditions it would be appropriate to route waste gas to the decay tanks prior to release; however, because of improvements in fuel design and plant operating practices, normal coolant activity levels have been much lower than the original design assumptions. As such, it is acceptable to route gaseous waste directly to the plant vent.

SAFETY EVALUATION JPN-PSL-SENS-96-039
REVISION 1

EVALUATION OF THE CEDMCS COOLING SYSTEM AND ENCLOSURE

Summary

This safety evaluation documents the acceptability of the cooling system and enclosure of the control element drive mechanism control system (CEDMCS) located on the 43' elevation of the reactor auxiliary building. Although the CEDMCS is a non-safety, non-seismic system, the seismic adequacy of the design was considered to ensure no interaction concerns existed. Additionally, the UFSAR was revised to note the additional combustible loading associated with the enclosure itself.

SAFETY EVALUATION JPN-PSL-SENS-96-046
REVISION 0

UNITS 1 & 2
USE OF THE STATION BLACKOUT CROSS-TIE
FOR NON-LICENSED BLACKOUT EVENTS

Summary

The Unit 1 & 2 UFSARs provide a discussion on the use of the station blackout (SBO) cross-tie with respect to the analyzed SBO event. The analyzed SBO event considers that both units are initially at power and, upon a blackout, are both maintained in a hot standby condition (Mode 3) with AC power provided from a single available emergency diesel generator. The blacked-out unit is powered via the SBO cross-tie. There is no discussion in the UFSARs on the use of the cross-tie for a blackout event which occurs while the unit is operating in Modes other than Mode 1 (i.e., blackout events beyond design and licensing basis). This evaluation recognizes the use of the cross-tie for those events and revises the UFSARs accordingly.

REMOVAL OF REDUNDANT HPSI, LPSI AND SHUTDOWN COOLING VALVE POSITION INDICATION

Summary

The subject valve position indication devices were intended to provide operators with an indication of valve percent open. These indications were in addition to the separately provided open/closed indication. Per the plant Emergency Operating Procedures, Normal Operating Procedures and Off-Normal Operating Procedures, when the associated system valves are required to be modulated, operators are required to rely on monitored parameters such as system flow, pressurizer level or reactor coolant system temperature and not this valve position indication.

Per the plant's Total Equipment Database, these indicators are not required to satisfy NRC Regulatory Guide 1.97 requirements for post-accident monitoring. The valves' open and closed indicating lights are used to determine valve position. As such, these historically unreliable indicators have been removed.

SAFETY EVALUATION JPN-PSL-SENS-96-060
REVISION 0

UNITS 1 & 2
REVISION OF STEAM GENERATOR BLOWDOWN SPENT RESIN TRANSFER METHODS

Summary

The steam generator blowdown facility is shared by both units. Transfer of spent resin is described in the Unit 1 UFSAR as via gravity feed to the spent resin storage tank. This evaluation revised the UFSAR to recognize alternate means of resin processing. All resins are treated as radioactive until sampled and analyzed by Health Physics. Following this analysis, resins may be used in another ion exchanger or transferred into a storage vessel or shipping container. Transfer of resin may be as described in the Unit 1 UFSAR, or resin can be directly pumped from the ion exchangers to a storage container or shipping liner. Per plant procedures, the shipping liner would then be de-watered prior to off-site shipping. Elimination of the use of the spent resin storage tank during transfer saves a process step and does not pose a nuclear safety concern.

SAFETY EVALUATION JPN-PSL-SENS-96-065
REVISION 0

UNITS 1 & 2
USE OF BREAKAWAY LOCKS ON THE HOT SHUTDOWN PANEL ROOM DOORS

Summary

This safety evaluation documented acceptability of installing a breakaway lock on the door to the hot shutdown panel room of each unit. Breakaway locks provide improved security by allowing an easy means of determining if the room had been entered without control room authorization. Because the locks are of the breakaway type, there is no restriction to operator access should the key not be readily available when access is required.

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

UNITS 1 & 2
HIGH RATE OF CHANGE OF POWER TRIP DESIGN BASIS

Summary

This safety evaluation was issued to clarify the design basis of the high rate of change of power (HRCP) trip of the Unit 1 and Unit 2 reactor protection systems (RPS). The existing Unit 1 & 2 UFSARs and Technical Specifications refer to the HRCP trip as an equipment protective trip which is not required for reactor protection and is not credited in the accident analyses.

ABB-CE, via TechNote 96-04, clarified the original design intent of the HRCP trip function. The TechNote suggests that a lack of discussion in original FSARs has caused some utility and ABB-CE personnel to erroneously assume that the HRCP trip was not credited in a plant's safety analyses. According to 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-SENS-96-072
REVISIONS 0 & 1

UNITS 1 & 2
ASSESSMENT OF CEA MODE SELECTOR SWITCH

Summary

This safety evaluation clarified the Unit 1 & 2 UFSARs with respect to plant operation with the control element assembly (CEA) mode selector switch in the OFF position rather than in the AUTOMATIC position. Both unit's UFSARs describe automatic system operation with respect to maintaining programmed reactor coolant temperature and power level during boric acid concentration changes. Although the UFSARs describe automatic operation, they also mention manual operation. It was not the intent of these UFSAR discussions to limit plant operations to a particular mode (i.e., manual or automatic), rather, the UFSARs are providing a general description of system capabilities. Automatic operation was provided as a part of the system's design to support load-following plant operations.

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

UNITS 1 & 2
HYPOCHLORITE SYSTEM UFSAR CLARIFICATION

Summary

As a result of UFSAR reviews, a discrepancy regarding the hypochlorite system description was identified. Specifically, both the Unit 1 and 2 UFSARs refer to a hypochlorite "generating" system when identifying the system; however, the plants have frequently been using a temporary system for chemical injection. This temporary system does not generate its own chlorine solution, rather it uses a chlorine solution which is provided from an off-site source and is stored in a suitable tank.

The hypochlorite system is a non-safety system common to both units and was designed to produce a sodium hypochlorite solution via electrolytic decomposition of filtered seawater and to periodically inject this solution into the sea water intake bays for the control of biological fouling. As a result of increased maintenance, equipment aging, parts obsolescence, etc., the original system (i.e., the "generating system") is frequently not in service and a liquid chlorine solution from an off-site source is used.

Whether or not a chlorine solution is generated on-site is immaterial to the achievement of the system's function. This evaluation revises the Unit 1 & 2 UFSAR descriptions to replace the term "hypochlorite generating system" with the term "hypochlorite system."

SAFETY EVALUATION JPN-PSL-SESP-96-093
REVISION 0

UNITS 1 & 2
RESPONSE TO NRC RAI REGARDING PRESSURIZED THERMAL SHOCK

Summary

This evaluation provided St. Lucie's response to an NRC request for additional information (RAI) regarding the plant's pressurized thermal shock (PTS) evaluation previously submitted to the NRC. A 10 CFR 50.59 evaluation was performed in order to revise Unit 2 UFSAR Table 5.2-7a, Impact Test Data for St. Lucie 2 Beltline Weld Materials. Several minor enhancements were made to this table, including the addition of temperature data and surveillance weld material property data.

SAFETY EVALUATION PSL-ENG-SENS-97-006
REVISIONS 0 & 1

EVALUATION OF FULL CORE OFFLOADS

Summary

This safety evaluation documented the acceptability of a full core offload as a routine plant outage practice. The ability of the spent fuel pool cooling system to maintain fuel pool temperatures below the 140F limit provided in the NRC Standard Review Plan has been confirmed. The following restrictions were provided in the evaluation to ensure it remains bounding for future use:

- the calculated maximum decay heat load during each full core offload shall be ≤31.7 E6 BTU/hr at 168 hours following reactor shutdown;

- the reactor must be subcritical for at least 168 hours prior to placement of the first offload assembly in the pool;

- prior to offload, pool water level shall be confirmed to be within its nominal operating band;

- with one heat exchanger in service, cooling water flow shall be maintained ≥3560 gpm;

- two fuel pool cooling pumps shall be in operation (exceptions noted); and

- control room annunciation for pool high temperature, pump status and pool level shall be operable.

SAFETY EVALUATION JPN-PSL-SEFJ-97-011
REVISIONS 0 & 1

LOW FLOW TRIP SETPOINT CHANGE

Summary

This safety evaluation revised the reactor coolant low flow trip setpoint to gain operating margin. During implementation of the Cycle 9 low flow trip setpoint procedure, pre-trip alarms were generated. It was found that the calculated values of the trip setpoints were close to the actual measured flow values as indicated by steam generator pressure drop readings. The methodology used in generating these original setpoints was identified as being conservative with respect to the incorporation of uncertainties.

The methodology used for the revised setpoint is consistent with that used in other St. Lucie setpoint calculations. For the revised setpoint, the uncertainties were combined using the square root sum of the squares (SRSS) method. From a safety analysis standpoint, the reactor trip for degraded flow remains unchanged and the new setpoint values comply with technical specification requirements.

SAFETY EVALUATION PSL-ENG-SEES-97-019
REVISION 0

DELETION OF REQUIREMENT FOR REFUELING MACHINE UNDERWATER CAMERA

Summary

This safety evaluation documented the acceptability of removing the underwater camera from the refueling machine. Additionally, the evaluation also documented the acceptability of using a portable temporary camera, if desired. Per the UFSAR, both a viewing port in the refueling machine trolley deck as well as electronic and visual indication of refueling machine position over the core are provided. The underwater camera provides supplemental visual indication for refueling operations and is a tool to aid in refueling. The camera is not required nor credited with the prevention or mitigation of a fuel handling accident.

Removal of the camera will result in elimination of work presently resulting in high radiation doses.

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

UNITS 1 & 2
WASTE GAS SYSTEM ANALYZER OPERATION

Summary

This safety evaluation addresses a Condition Report which identified a concern related to the UFSAR description for operation of the waste gas system analyzers and provides the documentation necessary to revise the UFSAR descriptions.

The waste gas analyzer is typically operated by sampling the in-service Gas Decay Tank (GDT) at a single point. However, the Unit 2 UFSAR states that the gas analyzer is used to "sequentially measure several points in the system." Although this configuration describes a method of operation based on a gas analyzer feature that allows for automatic sequential sampling operation, it does not correctly reflect the way the system is operated per plant procedures (a single sample point is used depending on system operation, Gas Surge Tank (GST) to GDT or GST to plant vent).

The continuously monitored single sample points are selected based on the desired method of gaseous effluent release. If gas samples exceed a predetermined level of radioactivity and holding of the gaseous effluent is desired, then the in-service GDT path is utilized and continuously monitored for oxygen; likewise, if little or no radioactivity exists in the sample, then the GST to plant vent path is utilized and continuously monitored. Both methods of gaseous effluent control and release are monitored for radioactivity. This evaluation concludes that operation of the waste gas analyzer and the waste gas system in a manual and/or batch mode with continuous monitoring of a single sample point does not adversely impact plant safety and does not conflict with the Technical Specifications or the Offsite Dose Calculation Manual.

SAFETY EVALUATION PSL-ENG-SESJ-97-026
REVISION 0

UNITS 1 & 2
RESPONSE TO NRC SUPPLEMENTAL RAI
REGARDING PRESSURIZED THERMAL SHOCK

Summary

This evaluation provided St. Lucie's response to a supplemental NRC request for additional information (RAI) regarding the plant's pressurized thermal shock (PTS) evaluation previously submitted to the NRC. Safety evaluation JPN-PSL-SESP-96-054 documented Engineering's response to the original RAI. A 10 CFR 50.59 evaluation was performed in order to revise Unit 2 UFSAR Table 5.2-7a, Impact Test Data for St. Lucie 2 Beltline Weld Materials, and Table 5.3-2, Reactor Vessel Beltline Weld Toughness Properties. Several minor enhancements were made to Table 5.2-7a, including the addition of temperature data and surveillance weld material property data. An informational note was added to Table 5.3-2.

SAFETY EVALUATION PSL-ENG-SEMS-97-034
REVISIONS 0 & 1

EVALUATION FOR THE USE OF SIMULATED INCORE DETECTOR ASSEMBLIES IN CORE LOCATIONS R9 AND T13

Summary

This safety evaluation documented the acceptability of installing two simulated fixed incore detector assemblies provided by ABB-CE. The evaluation also considered the effect of broken detector pieces left in the thimble tubes at these detector locations. The simulated assemblies were installed during the Cycle 10 refueling outage.

During inspection of the upper guide structure (UGS) incore instrumentation (ICI) guide tubes, broken ICI detector segments were located in locations R9 and T13 and could not be removed. Because of this problem, the locations could not be used for incore instrumentation. It was decided to push the broken pieces down into the thimble tubes so that they remain shielded well under water during the UGS lift. Simulated ICI assemblies were subsequently installed.

The simulated assemblies are dimensionally similar to the normal detector assemblies and were designed for this application. The number of remaining available detectors remains above UFSAR and Technical Specification requirements.

The broken detector pieces left in the thimble tubes were determined to have no adverse effect on plant operation or safety. This conclusion was based on an evaluation performed for the effects related to vibration, seismic loads, loose parts migration/potential consequences, and effects pertaining to neutronic aspects and core bypass flow.

SAFETY EVALUATION PSL-ENG-SENS-97-036
REVISION 0

INTAKE STRUCTURE VENTILATION SYSTEM OPERATION

Summary

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

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

SAFETY EVALUATION PSL-ENG-SENS-97-038
REVISIONS 0 & 1

PRESSURIZER CODE SAFETY VALVE MODIFICATION

Summary

Plant Change/Modification (PC/M) 96139M provided for the replacement of the pressurizer code safety valves during the 1997 refueling outage. Quality Report 97-0754 documented a Quality Assurance (QA) review of the modification which concluded that a 10 CFR 50.59 safety evaluation should have been performed for the modification since the change represents a change to the facility as described in the UFSAR. Specifically, the QA report noted that the UFSAR describes the valves as having a blowdown of approximately 10% whereas the new valves have a specified blowdown value of 4%. Condition Report 97-0753 was written on the subject.

Although the PC/M provided an analysis which addressed the change in valve blowdown, it did not include a 10 CFR 50.59 evaluation. This evaluation documents the 10 CFR 50.59 consideration for changing the blowdown of the subject valves and revises the UFSAR.

SAFETY INJECTION TANK PRESSURIZATION REQUIREMENTS FOR MODE 3 & 4 OPERATION WITH REDUCED REACTOR COOLANT SYSTEM PRESSURE

Summary

This safety evaluation was written to evaluate and correct UFSAR wording pertaining to operation of the safety injection tanks at reduced reactor coolant system pressures. Existing UFSAR wording is too explicit and excessively precise in its language and it conflicts with the relevant technical specification requirements and plant procedures.

SAFETY EVALUATION PSL-ENG-SEMS-97-043
REVISION 0

UNITS 1 & 2
FIRE PUMPS STARTING SEQUENCE

Summary

This evaluation revises the UFSAR to clarify that the two fire pumps do not start sequentially. The fire protection code (NFPA-20) refers to sequentially starting fire pumps. This requirement is intended to prevent a water hammer and possible electrical system overload. Since the St. Lucie design incorporates a hydropneumatic tank which maintains the piping system full of water and since the two fire pumps are powered from separate electrical busses, the need for sequential pump starting is eliminated. This evaluation also clarifies that the pumps start at a system pressure of "equal to or greater than 85 psig" rather than "at 85 psig."

SAFETY EVALUATION PSL-ENG-SENS-97-044
REVISION 0

CONTAINMENT SUMP SCREENS

Summary

Condition Report 97-1102 documents gaps identified in the containment sump screens. This evaluation was prepared as a response to the CR.

The containment sump screens act as a barrier to prevent debris from entering the emergency core cooling system and containment spray system. Per the UFSAR and the original NRC Safety Evaluation Report, these screens were designed in accordance with NRC Regulatory Guide 1.82, Revision 0. This evaluation documents the design and licensing requirements for the sump screens and clarifies the design bases, including a discussion on the acceptability of gaps in the screen. An analysis is provided for the sump divider screen (the screen which segregates the sump into A and B trains) to determine the limiting divider screen gap size. The UFSAR was revised to more accurately describe the overall sump screen design.

SAFETY EVALUATION PSL-ENG-SENS-97-045
REVISION 0

UFSAR CHANGE FOR CONTAINMENT SUMP SCREENS

Summary

As a result of repairs required for the emergency core cooling system containment sump screens, questions were raised regarding the acceptability of using a 20 gage (wire) mesh screen for repairs to the 18 gage mesh sump screen. The UFSAR describes only an 18 gage mesh size with an open area of 0.0081 square inches. This evaluation documented the acceptability of using the 20 gage mesh screen for sump screen repairs. The 20 gage mesh was considered acceptable since: it has an equivalent opening of 0.090" diameter required for filtering of particles that could become lodged in the fuel; it does not adversely affect the capability of the screens to withstand maximum debris loading; it will not adversely affect the flow rate through the screens; and it is constructed of 304 stainless steel. The UFSAR was revised to note the acceptability of a screen mesh size other than 18 gage.

SAFETY EVALUATION PSL-ENG-SEMS-97-070
REVISION 0

UFSAR COMBUSTIBLE LOADING UPDATE

Summary

NRC Information Notice 92-82 stated that Thermo-lag was a combustible material. This evaluation revises UFSAR fire zone combustible loading information as a result of including Thermo-lag as a combustible. The storage of various combustible materials related to plant operation and maintenance has also been included. The evaluation concluded that, with this additional combustible loading, adequate fire protection is provided to ensure the continued availability of redundant safe shutdown equipment and components.

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

UNITS 1 & 2
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 (LOCA). The UFSAR descriptions include a list of specific sanitation supplies based on Office of Civil Defense Sanitation Kit III which was apparently developed for use in fallout shelters (supports 25 persons for 2 weeks). In addition, the UFSARs note that a 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 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 will recognize the site's Emergency Plan with respect to ensuring adequate food, water, medical and sanitary 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.

SECTION 3

RELOAD SAFETY EVALUATIONS

PLANT CHANGE/MODIFICATION 96172

ST. LUCIE UNIT 2 CYCLE 10 RELOAD

Summary

This engineering package provided the reload core design of the St. Lucie Unit 2 Cycle 10. The Cycle 10 core is designed for cycle lengths between 12,793 and 13,545 EFPH, depending upon variation in the Cycle 9 length of between 9,902 and 11,024 EFPH, respectively. The cycle lengths for Cycle 10 included an end of cycle inlet temperature coastdown to 535°F followed by a coastdown in power to approximately 85% power. Cycle 9 is expected to reach an EOC exposure of approximately 10,738 EFPH.

The primary design change to the core for Cycle 10 is the replacement of 64 irradiated fuel assemblies with fresh Region M fuel assemblies. The fuel is arranged in a low leakage pattern similar to the design of the Cycle 9 core. The mechanical design of Region M fuel is the same as that of Region L (Cycle 9) with the exception of incorporating coreless HID-1L grids. The two designs are equivalent from the standpoint of performance and functional requirements.

The safety analysis of this design was performed by Asea Brown Boveri Combustion Engineering Nuclear Operations (ABB CENO) and independently reviewed by Florida Power and Light Co.