1990 ANNUAL OPERATING REPORT

ST. LUCIE UNITS 1 & 2

AND

1990 ST. LUCIE UNITS 1 & 2

STEAM GENERATOR IN-SERVICE INSPECTION REPORT

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SECTION 1

ST. LUCIE

ANNUAL 10 CFR 50.59 REPORT

A summary of changes to the facility as described in the Final Safety Analysis Report (FSAR) (10 CFR 50.59 (A) (1) (i)) is submitted by separate letters at the same time as the annual FSAR update for each unit (July 22 for St. Lucie Unit 1 and April 6 for St. Lucie Unit 2).

Changes to procedures as described in the FSAR (10 CFR 50.59 (A) (1) (ii)) and tests and experiments not described in the FSAR (10 CFR 50.59(A) (1) (iii)) are attached.

10 CFR 50.59 Evaluations

Temporary Changes via Jumper/lifted Leads Requests

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-07

Component/System Affected:

Refueling Machine main control cabinet

Description of Change:

This jumper/lifted lead request is to install a jumper across LS-HUL to simulate a signal to the hoist up limit switch. The other jumper is to simulate a signal to LS-FSR to indicate the fuel spreader is retracted.

Safety Evaluation Summary:

The hoist box on the refueling machine is not installed and undergoing repairs. These jumpers are necessary to provide electrical power to move the refueling machine. This will enable unlatching of the CEA's using electrical power, rather than having to hand crank the refueling machine.

Caution tag to be installed on control stick on refueling machine to indicate these jumpers are installed. The jumpers are to be removed upon completion of unlatching CEA's.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-09

Component/System Affected:

Heating and Ventilation Exhaust (HVE) fan 8B

Description of Change:

This jumper/lifted lead request is to run HVE-8B with power from the "A" train or opposite train.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. The "A" power bus for the purge fans contacts will be worked on. This jumper/lifted lead request allows the fans to continue operation.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-17

Component/System Affected:

1A & 1B Diesel Generators

Description of Change:

This jumper/lifted lead request results from the inadvertant diesel generator start caused by the Engineered Safeguards Features Actuation Signal (ESFAS) Plant Change Modification (PC/M). Start signals to the diesel generator from Safety Injection Actuation Signal (SIAS), Containment Isolation Signal (CIS), and Containment Spray Actuation Signal (CSAS) were lifted. Undervoltage diesel generator start has not been affected.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. In mode 6 operation the lifted lead only isolates the diesel generators from ESFAS and are not involved in the operation of other equipment.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. Again, this lifted lead does not impact other equipment operation, only the diesel generator start. ESFAS is not required to be operable in Mode 6.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-18

Component/System Affected:

Control Room Outside Air Intake Radiation Monitoring

Description of Change:

To perform calibration of Control Room outside air intake radiation monitoring.

Safety Evaluation Summary:

Only one channel of Control Room outside air intake radiation monitoring can be jumpered at a time. Once the jumper is installed work must be progressed in an expeditious manner. If work must be stopped prior to completion jumpers must be immediately removed. If the Control Room outside air intake system actuates from another source, immediately stop work and remove jumpers.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. One channel of Control Room outside air radiation monitoring will remain operable at all times. This is capable of fully actuating the Control Room outside air intake ventilation system.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Only one channel of Control Room outside air radiation monitoring will be jumpered at a time. Containment Isolation Signal for Unit 2 will not be affected.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. The Control Room outside air intake ventilation system actuation signals are not addressed in the Technical Specification, equipment addressed by the Technical Specification 3.7.7.1 will remain operable.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-30

Component/System Affected:

Turbine trip from 20ET Solenoid and 20 Overspeed Protection Control

Description of Change:

Need to stroke turbine valves. Lifting lead 38 in RTGB 101 will deenergize 20ET solenoid. This will allow testing of govenor, reheat, throttle and intercept valves of turbine.

Safety Evaluation Summary:

Since the Unit is shutdown in Mode 5 with more than adequate shutdown margin, there is no increase in the probability of risk to safety of health and public.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Unit being shutdown in Mode 5 with adequate shutdown margin.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evalated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-31

Component/System Affected:

CEA Rod #30

Description of Change:

A problem occurred with the indicator for CEA #30. During the Unit 1 outage, the Reed Switch Position Transmitter (RSPT) for CEA #30 days was replaced. The purpose of this jumper/lifted lead is to use spare cable, previously for part length CEA #35, from the quick disconnect in containment to the terminal connections in the cable spreading room. Since all of the color codes are the same, one tag is used for the cable in the cable spreading room cabinet. Verification of proper hookup will be accomplished via rod drop testing prior to startup.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. The spare cable will be tested via rod drop testing.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. The same indicator is being used for the RSPT, however using spare cable from part length CEA quick disconnect to cable spreading room.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-33

Component/System Affected:

Transfer pumps/Domestic water pumps

Description of Change:

This jumper/lifted lead was installed temporarily to avoid domestic water pump lockout while testing low level alarm.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. FUSAR section 9.2.6.3 specifically states the domestic water system performs no safety functions.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-38

Component/System Affected:

Main Feedwater Regulating Valve

Description of Change:

Special postmaintenance testing channel check on main feedwater regulating valves.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Operators have manual control capability of all feed system components as per FUSAR section 7.7.1.3.1.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. As previously stated, operators have manual control capability of all feed system components as per FUSAR section 7.7.1.3.1.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specifications.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-39

Component/System Affected:

27-1 Undervoltage (degraded voltage) relay for the 1A2 480V load center terminals 11 & 12.

Description of Change:

This jumper/lifted lead request maintains the referenced relay in the tripped position as per Technical Specification 3.3.2.1. The relay must be replaced or repaired by the next calibration channel check.

Safety Evaluation Summary:

This jumper/lifted lead request is covered in the FUSAR since a 2-out-of-2 logic is used for these undervoltage relays.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. As referenced in the FUSAR page 8.3-5a, each Class IE 480V bus (1A2 & 2B2) utilizes two undervoltage definite time relays in a 2-out-of-2 coincident logic scheme.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-75

Component/System Affected:

Line 5 of Gaitronics alarm relay cabinet

Description of Change:

This lifted lead reflects a change requested by operations to split channel 5 on the Gaitronics between Unit 1 and Unit 2. This lifted lead was originally done under jumper/lifted lead number 7-55, but an NCR was generated in order to update the evaluation.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 1 Request Number: 0-77

Component/System Affected:

Flow Indicator Switch FIS-21-9A

Description of Change:

This jumper/lifted lead is to facilitate lifting lead for annunciator S-3 to remove a hard electrical ground. This ground is causing spurious, false annunciations. The cause of the ground is water intrusion and apparent degradation of the conduit from FIS-21-9A in the Component Cooling Water (CCW) area.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. The worst case in the FUSAR is loss of offsite power with both emergency diesel generators failing to start, natural circulation would be maintained for at least 3 hours, average restoration of offsite power is 36.6 minutes.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. One CCW Heat Exchanger alone can safely accommodate a Loss of Cooling Accident (LOCA) heat load as per section 9.7.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. Only need one Component Cooling Water/Intake Cooling Water Heat Exchanger to handle design basis LOCA.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-01

Component/System Affected:

HCV-25-5 & HCV-25-6 Containment Purge Valves

Description of Change:

This jumper/lifted lead request was for installing an air jumper around the following valves, SE-25-7 and SE-25-8. This was done to support mechanical maintenance by failing open the HCV-25-5 & HCV-25-6 containment purge valves.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Containment integrity was not set due to Mode 5 operations during a refueling outage. Therefore, it was not necessary for the purge valves to be closed.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. In Mode 5 operation containment integrity is not required.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-03

Component/System Affected:

Turbine Trip Instrumentation

Description of Change:

This lifted lead request disabled turbine trip instrumentation to enable work on limit switches.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Turbine trip instrumentation is not required to be operable during Mode 5 operation.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-04

Component/System Affected:

Reactor Protection System (RPS) Channel A & B Steam Generator Level Trips

Description of Change:

This lifted lead entailed lifting of multiple leads in RPS cabinets A & B in order to simulate normal level on steam generator A. This was done to allow work on the control element drive system.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. During Mode 5 operations, the steam generator level trips are not required to be operable.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-8

Component/System Affected:

FIS-14-15B

Description of Change:

This jumper/lifted lead allows for a temporary hookup (less than 72 hours) of instrumentation across the following components of the B channel loss of CCW circuitry: (1). Test resistor on output of flow transmitter for CCW flow channel B. (2). Resistor on square root extractor output. (3). Agastat coil on time.

Safety Evaluation Summary:

This will allow the nickel A instrumentation to begin recording when the output of the square root extractor goes less than 4.3 volts. Inturn, this will allow recording of any fluctuations in the circuit such that the root cause of the spurious trip signal can be identified. To obtain more data this jumper/lifted lead was changed to greater than 72 hours installed. At this time it appears spikes are being received from flow transmitter. Channel B is out of service.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. With channel B of loss of CCW trip in bypass, there is still the required 2 out of 3 logic as referenced in section 7.2.1.1.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. Channel B is already in bypass, this instrumentation is merely monitoring for a change in transmitter output.

The proposed activity does not reduce the margin of safety as defined in the bases for any Technical Specification. Margin of safety is still assured with the 2 out of 3 logic with the B channel in bypass.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-12

Component/System Affected:

Main containment purge valves, FCV 25-4, 5

Description of Change:

This jumper/lifted lead was installed to allow FCV-25-6 to open without opening FCV 25-4 and 5. The purpose of this jumper/lifted lead was to allow the Test Group to perform a local leak rate test; the jumper/lifted lead was removed upon completion of the test.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. FCV-25-4 and 5 remained sealed closed containment isolation valves as required by Technical Specification 3.6.1.7.

The proposed activity does not create the possibility of an accident of a different type than any previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-13

Component/System Affected:

Reactor Coolant Pump (RCP) 2A2 Stator Temp. (TI-1165), Upper Guide Bearing Temp. (TI-1166), Lower Guide Bearing Temp. (TI-1167).

Description of Change:

This jumper/lifted lead facilitates monitoring the RCP 2A2 Upper and Lower Guide Bearing Thermocouples and Stator Temperatures with a recorder. This was due to suspected inaccuracy of the 2A2 upper oil reservoir level indication. Level was lowered; however, no evidence of oil smoke in atmosphere is evident, no significant amount of oil has accumulated in RCP oil collection system and there has been no increase or abnormalities noted in other 2A2 instrumentation.

Safety Evaluation Summary:

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Jumpers are for recorders, for 2A2 RCP Stator Temperature, Upper Guide Bearing Temperature and Lower Guide Bearing Temperature.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-16

Component/System Affected:

FSE-27-12, of the Hydrogen Sampling System

Description of Change:

FSE-27-12 solenoid is to be replaced. Its solenoid is deenergized under Plant Work Order (PWO) # 6816. This meets the requirement of Technical Specification 3.6.1 since this lifted lead removes valve position indication.

Safety Evaluation Summary:

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. FSE-27-12 is a failed closed valve and is deenergized which puts it in the failed closed position.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. The valve is in its containment isolated (closed) position.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-20

Component/System Affected:

HCV-09-1B Valve Box, Annunciator P-16 & Local annunciator

Description of Change:

One switch in PS-09-1B2 has failed causing the annunciator to lock in. Need to replace switch and restore wiring in accordance with control wiring diagram(CWD).

Safety Evaluation Summary:

This jumper/lifted lead is to document work already completed as per request of the Instrumentation & Control department and the Assistant Nuclear Plant Supervisor(ANPS). A Plant Work Order was issued and performed to remove a nuisance alarm that is locked in with no reflash capability. The wires to the failed pressure switch PS-09-1B2 were lifted and the local annunciator switch was then wired to the control room annunciation. Alterations to the wiring per CWD 656 (2998-B-327), Main Feedwater Isolation Valve HCV-09-1B, were made as follows:

1. Blue wire was lifted off terminal 13.
2. Red wire was lifted off terminal 14.

The reason this was done was to prevent the failed pressure switch from locking in the annunciator in the control room.

3. Orange wire was lifted off terminal 11 and lended to terminal 14.
4. Yellow wire was lifted off terminal 9 and lended to terminal 13.

This wired the local switch into the control room annunciator circuit.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Use of switch, normally used for local annunciation, for annunciation in control room has not affected equipment safety.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-21

Component/System Affected:

HCV-09-1A Valve Box, Annunciator P-6 & Local annunciator

Description of Change:

One switch in PS-09-1A2 has failed causing the annunciator to lock in. Need to replace switch and restore wiring in accordance with control wiring diagram(CWD).

Safety Evaluation Summary:

This jumper/lifted lead is to document work already completed as per request of the Instrumentation & Control department and the Assistant Nuclear Plant Supervisor(ANPS). A Plant Work Order was issued and performed to remove a nuisance alarm that is locked in with no reflash capabilities. The wires to the failed pressure switch PS-09-1A2 were lifted and the local annunciation switch was then wired to the control room annunciation. CWD 655 (2998-B-B27) and Main Feedwater Isolation Valve HCV-09-1A, had the following alterations:

1. Blue wire was lifted off terminal 13.
2. Red wire was lifted off terminal 14.

The reason this was done was to prevent the failed pressure switch from locking in the annunciator in the control room.

3. Orange wire was lifted off terminal 11 and lended to terminal 14.
4. Yellow wire was lifted off terminal 9 and lended to terminal 13.

This wired the local switch into the control room annunciator circuit.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. Using local switch for annunciation in Control Room has not affected equipment safety, only loss of local indication.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-22

Component/System Affected:

HCV-09-1B Test Stroke

Description of Change:

To repair and/or replace open limit switch and fix oil leaks. Reason for this request was to prevent the valve from continuous testing due to short in limit switch.

Safety Evaluation Summary:

This jumper/lifted lead is necessary due to the valve open limit switch being full of oil. This caused shorting between terminals 24 & 25 which enabled and disabled continuously, the part stroke test circuit. This jumper/lifted lead disables completely the part stroke circuitry. Testing can still be facilitated (if needed) with assistance from the Instrumentation & Control department.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR. The activity affects the test circuitry and not the mechanical integrity of the system as referenced in the FUSAR section 15.2.5.1.

The proposed activity does not increase the consequences of malfunction of equipment to safety. Containment integrity will be maintained.

The proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety. Test circuitry that already malfunctioned on the valve for this jumper/lifted lead does not affect the safety function of the related valves.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-49

Component/System Affected:

HCV-08-1B Test Panel

Description of Change:

Change drawing 2998-B-327 sheet 316 to show the blue wire in cable 20316A-SB is open. The spare white/black wire is to used in its place.

Safety Evaluation Summary:

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR. The installed spare wire is to be used for a broken one so that a new wire does not have to be pulled.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR. Use of the installed spare is for the exact purpose for which it was originally installed.

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification. The installed spare wire is the same size wire as the blue wire in cable 20316A-SB.

10 CFR50.59 Evaluation for Temporary Changes via Jumper/Lifted Leads

Unit: 2 Request Number: 0-57

Component/System Affected:

6.9KV Switchgear 2B-1 Metering

Description of Change:

This jumper/lifted lead request will eliminate the use of a broken fixed contact block. This same jumper was installed previously on Unit 2. This request is a personnel safety concern as it defeats the bus PT transformers isolating capability. This is on the 2B-1 6.9KV bus. A caution tag will be installed on the outside of the cabinet to warn personnel of the fact the bus PT transformer isolation capability is defeated.

Safety Evaluation Summary:

The 6.9KV buses power the Reactor Coolant Pumps and the Main Feedwater Pump which are necessary for power operation but not for the safe shutdown of the reactor.

The proposed activity does not increase the consequences of malfunction of equipment important to safety previously evaluated in the FUSAR.

The proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FUSAR.

The proposed activity does not increase the consequences of an accident previously evaluated in the FUSAR.

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the FUSAR.

The proposed activity does not reduce the margin of safety as defined in the bases for any Technical Specification.

10 CFR 50.59 Evaluations

Summaries of Evaluations Approved by the St. Lucie Facility Review Group

ST. LUCIE UNIT 1
USE OF SEALING COMPOUND ON VALVES I-FCV-23-3,4,5

INTRODUCTION:

PSL-1 Blowdown Containment Isolation valves I-FCV-23-3 and 5 and SGBD system isolation valve I-FCV-23-4 have had a history of leakage problems. Numerous leak repairs have been performed on these Steam Generator blowdown system containment valves on both units.

The purpose of this evaluation is to provide a method for temporarily repairing body to bonnet leaks on valves I-FCV-23-3,4 and 5. The method of repair will be sealant injection. The valves shall be replaced or permanently repaired during the next scheduled outage or other suitable time period. These valves are nonisolable and normally open.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased. These valves are not accident initiating components. These valves serve to isolate containment and the SGBD system in the event of an accident.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this repair. The valves are required for maintaining containment and system isolation, and their ability to do so will not be affected.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. The valves are required for maintaining containment and system isolation, and their ability to do so will not be affected by this repair since the capnuts perform an identical function as the heavy hex nuts and belt loadings are not affected by the injection of sealant. The gasket and/or sealant does not perform a safety function.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased. The consequences of the failure of the injection seal is the same as the failure of the gasket, which would result in a loss of system fluid into the containment penetration room or containment.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created. The proposed repair does not provide a new mode of normal or emergency plant operation.

Chemistry limits are not altered and no other change is proposed to the plant design, modes of operation or assumptions in the basis for the Technical Specifications or Safety Analysis. Therefore, this repair does not reduce the margin of safety as defined in the basis for any Technical Specification.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created. Limiting the injection of sealant to just interacting outside the valve pressure boundary precludes a malfunction of a different type.

ST. LUCIE UNIT 1
SAFETY EVALUATION FOR TEMPORARY MODIFICATION
OF POLAR CRANE BRAKE SYSTEM

INTRODUCTION:

This Safety Evaluation is to provide justification for removal of the damaged portion of the polar crane main hoist #1 brake drum lining and to modify the control circuit for the main hoist #1 and #2 brakes.

The machining of the damaged position of the drum is necessary to remove cracks and thereby prevent their propagation.

The circuit modification will allow the interchanging of functions of the #1 and #2 brakes on the main hoist in terms of engagement after deenergization of the main hoist motor or the creep motor.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since these temporary modifications and associated operational restrictions will assure the capability of the polar crane main hoist to safely carry loads up to and including the reactor head and its lift rig.

The consequences of an accident previously evaluated in the FUSAR have not been increased by these temporary modifications. The modifications performed under this evaluation are to the polar crane braking system. The polar crane continues to comply to the requirements of NUREG-0612.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created by this temporary modification. The manufacturer has indicated that the machining of the No.1 brake drum and the circuitry revisions to switch the primary and time-delay braking functions will not adversely affect the ability of the main hoist to safely carry loads only up to and including the reactor head and its lift rig. The modifications to the polar crane braking system do not create any new failure modes which could impact the operation of equipment important to safety.

The margin of safety as defined in the bases of any Technical Specifications has not been reduced.

ST. LUCIE UNIT 1
SAFETY EVALUATION FOR TEMPORARY MODIFICATION OF POLAR CRANE BRAKE SYSTEM AND LOAD CELL CONTROL FUNCTIONS

INTRODUCTION:

This Safety Evaluation is to provide justification for removal of the damaged portion of the polar crane main hoist #1 brake drum lining, modify the control circuit for the main hoist #1 and #2 brakes, and modify the interlock functions performed by the Load Cell.

The machining of the damaged portion of the drum is necessary to remove cracks and thereby prevent their propagation.

The circuit modification for the main hoist brakes will allow the interchanging of functions of the #1 and #2 brakes on the main hoist in terms of engagement after deenergization of the main hoist motor or the creep motor.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since these temporary modifications and associated operational restrictions will assure the capability of the polar crane main hoist to safely carry loads up to and including the reactor head and its lift rig.

The consequences of an accident previously evaluated in the FUSAR have not been increased by these temporary modifications. The modifications performed under this evaluation are to the polar crane braking system and load cell control functions. The polar crane continues to comply to the requirements of NUREG-0612.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR has not been increased by these temporary modifications. The modifications performed under this evaluation are for the polar crane braking system and load cell control functions. This system continues to satisfy the requirements of NUREG-0612.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created by this temporary modification. The manufacturer has indicated that the machining of the No. 1 brake drum and the circuitry revisions to switch the primary and time-delay braking functions will not adversely affect the ability of the main hoist to safely carry loads only up to and including the reactor head and its lift rig.

The margin of safety as defined in the bases of any Technical Specifications has not been reduced by these temporary modifications.

ST. LUCIE UNIT 1
SAFETY EVALUATION FOR PLANT OPERATION WITH CE MECHANICAL PLUGS IN EDM DAMAGED STEAM GENERATOR TUBES

INTRODUCTION:

During implementation of PCM 251-189 in February 1990, "Steam Generator Tube Plug Replacement - Westinghouse to CE Design Plug", forty tubes in the steam generator A and B cold leg side were damaged by the EDM process operations while removing the Westinghouse plugs. In addition, a problem was identified in the Steam generator A cold leg side. An EDM scar was found to have a taper that was not typical as those demonstrated during previous operations and CE qualification tests. After review, ten of the forty tubes with EDM scars were plugged with the CE mechanical plug and the remaining thirty had welded plugs installed. Prior to final determination of the root cause of the nonperpendicular electrode travel, a number of welded plugs were installed because of the use of an interim acceptance criteria.

SAFETY EVALUATION:

The probability of occurrence of a design basis accident of malfunction of equipment important to safety previously evaluated in the FUSAR is not increased since the flow diode effect does not result in adverse plant conditions such as unacceptable damage to adjacent intact tubes, does not decrease the design margin of the RCS pressure boundary and does not alter existing accident mitigation equipment or systems.

The consequences of a previously postulated design basis accident or malfunction of equipment important to safety previously evaluated in the FUSAR are not made more severe for the same reasons given above the consequences of fishmouthing and plug collapse are no more severe than a steam generator tube rupture, a previously evaluated condition.

The possibility of an accident of a different type than previously addressed in the FUSAR does not exist since fishmouthing does not result in unacceptable damage to adjacent tubes nor adversely impact the performance of the steam generator. Primary to secondary leakage after a plug collapse requires a subsequent tube leak. Failure of a tube plug would be no more severe than a steam generator tube rupture, a previously evaluated condition. A tube plug could "loosen" and fall from the tubesheet after plug collapse. If this scenario were to occur, the plug could come to "rest" against the lower surface of one or more fuel assembly retention grids, flow to affected fuel assemblies would not be significantly affected, due to the small size and geometry of the plugs. If a plug were to migrate to the lower part of the core, existing loose parts monitoring equipment would alert control room operators to the problem, and action as required by Technical Specifications would be taken. Therefore, no new accidents are created.

The margin of safety as defined in the basis for any Technical Specification is not reduced since the total number of tubes plugged has not changed.

ST. LUCIE UNIT 1
SAFETY EVALUATION FOR I-MV-09-08 BY PASS LINE
ELBOW LEAK SEALANT ENCLOSURE

INTRODUCTION:

NCR 1-507 identifies a pinhole leak on the upstream weld on the first elbow on the 1" bypass line around valve I-MV-09-08. This Safety Evaluation will permit the installation of a leak sealant enclosure around the leaking elbow and preclude loss of feedwater or auxiliary feedwater inventory. An engineering evaluation will proceed in parallel for the use of the enclosure as long term modification.

SAFETY EVALUATION:

The implementation of this temporary modification will have no impact on plant safety or operation. A review of the plant Technical Specifications and the Safety Analysis Report has shown that there are no unreviewed safety questions or Technical Specification changes involved.

The probability of an occurrence of an accident previously evaluated in the FUSAR is not increased because the installation of a leak sealant enclosure has been evaluated and it has been determined that the modified system stresses remain within Code allowables.

The consequences of an accident previously evaluated in the FUSAR have not been increased because the design and operation of the auxiliary feedwater system has not been changed and the capability of supplying feedwater to the steam generators is not affected.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased because the installation of the enclosure has been seismically analyzed with acceptable results, the system routing has not been changed so operational performance remains the same, the pressure design of the clamp will preclude the clamp from becoming a source missile or falling object because it has been designed for the system pressure, the chemistry of the leak sealant has been evaluated and will not introduce deleterious materials into the feedwater system, the injection of sealant will be limited to the volume of annular space inside the enclosure per Leak Repair procedures and the clamp design prevents disengagement of the piping.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created because the injection of sealant will be limited to avoid the introduction of sealant into the feedwater system, the chemistry of the sealant is compatible with the piping material, the enclosure is fabricated of carbon steel and therefore has the same coefficient of expansion as the piping and the pressure design of the clamp will preclude the clamp from becoming a source missile or falling object because it has been designed for the system pressure.

The proposed modification does not reduce the margin of safety as defined in the basis for any Technical Specification because the margin of safety as defined in the Technical Specifications is to have two redundant auxiliary feedwater systems capable of providing feedwater to the steam generators to maintain steam generator level for removal of decay heat, cooling the reactor coolant to 325 F temperature. The installation of the leak sealant enclosure does not prevent the modified system from performing its design function.

ST. LUCIE UNIT 1
ST. LUCIE UNIT 1 CYCLE 10, 10 CFR 50.59 FOR MODE 6 OPERATION

INTRODUCTION:

The St. Lucie Unit 1 Cycle 10 review of Mode 6 operation, (Reference 1) presents the evaluation to support the shuffling of fuel from the Cycle 9 loading pattern to the Cycle 10 loading pattern. This review was deemed necessary due to the early shutdown of the Cycle 9.

SAFETY EVALUATION

The St. Lucie Unit 1 Cycle 10 reload does not result in any changes to the overall configuration of the plant for Mode 6, except for the repositioning of the fuel assemblies within the core. Nothing outside the core is altered by this change and the method of plant operation while in Mode 6 remains unchanged. Therefore, the probability of occurrence of an accident or malfunction of equipment for Mode 6 important to safety is not impacted.

For operation at St. Lucie Unit 1 Cycle 10 the method for repositioning fuel inside the core during Mode 6 has not changed from previous cycles. No modifications in the method of plant operation or the plant configuration are required as a result of this change. For Mode 6 the possibility for an accident or malfunction of a different type than previously analyzed in the safety analyses is not created.

The St. Lucie Unit 1 Cycle 10 reload design neutronics input for Mode 6 and the resulting safety analyses has been reviewed, and in all cases the results are well within the acceptance criteria of the design basis. Furthermore, no changes were made in the methods used to evaluate the margin to safety.

As per Federal Regulation 10 CFR 50.59 (b) the above Safety Evaluation provides the basis to conclude that the implementation of fuel shuffling from the Cycle 9 loading pattern to Cycle 10 loading pattern for St. Lucie Unit 1 Cycle 10 Mode 6 reload operation does not involve any changes which introduce an unreviewed safety question.

ST. LUCIE UNIT 1
EVALUATION OF B_4C CENTER FINGER CONTROL ELEMENT ASSEMBLIES (CEA's)

INTRODUCTION:

During CEA operability testing at Maine Yankee, a control element assembly (CEA) jammed at approximately 80% of its fully inserted position. The center finger of the stuck CEA had become lodged in the center guide tube of the host fuel assembly due to a loss of the mechanical integrity of the center finger tip region (see Figure 1). Combustion Engineering, the designer and manufacturer of the CEAs in use at Main Yankee identified the failed CEA as having the "old style" CEA design; i.e., a CEA which contains B_4C absorber material extending to the tip of the center finger. St. Lucie Unit 1 cycle 10 core operates with twenty (20) CEAs of the old style Combustion Engineering design. St. Lucie Unit 2 operates with CEAs that are not of the old style design.

In general, the safety concern associated with the failure of CEAs is that insufficient shutdown reactivity will be available if required during normal or transient operation. The result of insufficient shutdown reactivity could be as severe as exceeding fuel design limits leading to loss of fuel rod integrity or exacerbating the consequences of a limiting FUSAR transient. In this specific case, the concern is the potential for common mode failure of old style CEAs. Failure in this context means the failure of one or more old style CEAs to insert on demand.

EVALUATION:

St. Lucie Unit 1 is capable of safe operation during Cycle 10 with old style CEAs. The following supports this conclusion.

The St. Lucie Unit 1 inspection results indicate no CEA end cap failures or circumferential cracking for an inspection group having exposures which are representative of the old style CEAs currently in operation.

The St. Lucie Unit 1 review of CEA manufacturing records showed that design and construction of CEAs were in accordance with approved materials, tests, inspections, procedures, and specifications.

The St. Lucie Unit 1 CEA inspection was within the range of exposures experienced by the failed CEAs at Maine Yankee. None of the St. Lucie Unit 1 CEAs had failed.

The evaluation of normal plant operations indicate that continued operation with old style CEAs does not affect the plant's ability to achieve safe shutdown.

The postulated failure of three high reactivity worth CEAs at any time during the remainder of Cycle 10 results in safety consequences which are within the acceptance criteria for the current Unit 1 safety analyses.

To ensure safe operation with the 20 old style CEAs until the end of the current cycle, the following augmented CEA surveillance program will be in place to reduce the likelihood of operation with an inoperable CEA.

Full stroke CEA exercising after trips and cold shutdowns will ensure CEA operability following the most probable precursor to failure, namely a thermal transition.

Quarterly full stroke CEA exercising will provide further assurance that the plant will not operate with a failed CEA.

Incore flux monitoring will be performed semi-weekly to provide detection of a gross CEA failure, and thus providing further assurance that CEAs are not failed and that the plant is operating in a safe manner

Repositioning of 20 old style CEAs to the fully withdrawn position will provide minimum neutron fluence to the tip region, thus minimizing any embrittlement affect.

Therefore, based on the surveillances and evaluations discussed, St. Lucie Unit 1 is capable of safe operation with old style CEAs until the end of the current fuel Cycle (EOC-10). This conclusion has received the concurrence of the Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

ST. LUCIE UNIT 1
CYCLE 10 10 CFR 50.59

INTRODUCTION:

The St. Lucie Unit 1, Cycle 10 reload design established by FPL consists of 92 Batch M natural uranium axial blanket assemblies. Sixteen of the Batch M assemblies contain no burnable absorber rods, twelve assemblies contain four 4 w/o Gd_2O_3, 16 assemblies have twelve 6 w/o Gd_2O_3, forty four assemblies have twelve 8 w/o Gd_2O_3 rods and the remaining 4 assemblies have twelve 6 w/o and four 8 w/o Gd_2O_3 rods. The remaining of the core consists of 8 Batch H, 9 Batch J, 16 Batch K, and 92 Batch L assemblies.

The St. Lucie Unit 1 Cycle 10 Safety Analysis Report (FUSAR) (Reference 1) presents the evaluation of the reload core characteristics with respect to the safety analysis presented in the St. Lucie Unit 1 Cycle 9 FUSAR (Reference 2). This base Safety Analysis was performed to support operations of St. Lucie Unit 1, Cycle 10. After this analysis was completed, the plant shutdown early to end Cycle 9. ANF performed an additional Safety Analysis for Cycle 10 (Reference 3) in order to address the impact of the shortened Cycle 9. This Safety Assessment supplements the Cycle 10 FUSAR.

SAFETY EVALUATION:

Based on the technical evaluation performed and the results of the reanalysis discussed in this Safety Analysis Report, it can be concluded that the St. Lucie Unit 1 Cycle 10 reload design meets all the design criteria, and can be implemented with no changes required to the existing St. Lucie Unit 1 Technical Specifications.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased.

The St. Lucie Unit 1 Cycle 10 reload does not result in any changes to the overall configuration of the plant. The plant's mode of operation remains unchanged.

The St. Lucie Unit 1 Cycle 10 reload design does not result in any changes external to the reactor core which impact the overall configuration of the plant, or the method in which the plant is operated. The possibility for an accident or malfunction of a different type than previously analyzed in the safety analysis is not created.

The St. Lucie Unit 1 Cycle 10 reload design neutronics and fuel design input to safety analysis has been reviewed, and in all cases the results of analysis are well within the acceptance criteria of the design and licensing bases. The acceptance criteria for the safety analysis have not been changed. Based on FPL technical reviews of the FUSAR report and of the safety assessment of the Cycle 9 early shutdown, it can be determined that the St. Lucie Unit 1 Cycle 10 reload design meets all the existing acceptance criteria. Therefore the Cycle 10 reload does not result in a reduction to the margin of safety relative to the Technical Specification bases for St. Lucie Unit 1.

ST. LUCIE UNIT 1
ICW SYSTEM ISOLATION VALVE RELINING

INTRODUCTION:

The original interior rubber lining on the subject butterfly valve(s) have demonstrated a recurring problem of disbonding/loosening from the valve body, allowing the turbulent process medium (seawater) to come in contact with the carbon steel valve body. This has resulted in erosion/corrosion deterioration on the valve body. Replacing the entire rubber lining with a polymer epoxy coating accomplishes the required valve body protection. The affected butterfly valves serve in the Intake Cooling Water (ICW) System (ASME Section III, Class 3), a Class 3, Safety Related System, and therefore, this change is classified as Safety Related.

The original valve rubber lining is identified as "Resiloseal Natural P", affixed to the valve body at the factory, The replacement coatings are Palmer International materials, "Ceramalloy CL, CP and Duratough DP", applied in accordance with the "ICW Valve Lining Technical Procedure/Guidelines" provided with the QL--2 RPA for this on-site work and the manufacturer's application instructions. The new coating is reported to be resistant to erosion, corrosion and cavitation, and is compatible for use on carbon steel surfaces within a seawater medium.

EVALUATION:

Design Equivalence of the replacement polymer epoxy coating material to the original rubber lining is assured by:

1) Application of the polymer epoxy coating in accordance with the "ICW Valve Lining Technical Procedure/Guidelines" and the manufacturer's instructions will prevent disbonding.

2) Resistance of the polymer epoxy coating to erosion, corrosion and cavitation will prevent valve body deterioration.

A Procurement Classification of QL-2 for the RPA was selected based on: the ICW system is Safety-Related, the relining material is commercial grade, 10CFR21 is not required, QAD approval is required on both the RPA and the P.O., the supplier is QA approved, and documentation is required.

FUSAR Section 9.2 (Table 9.2-1) will be revised to reflect this change.

The above evaluation establishes design equivalence and conformance to the original basis and supports the answers given on the "Nuclear Safety Evaluation Checklist" (JPN Form 4C). It has therefore been shown that there is no unreviewed safety questions or Technical Specification changes involved in this modification pursuant to 10CFR50.59. FUSAR Chapter 9 defines this system as safety related, therefore, this DEEP is classified as safety related.

ST. LUCIE UNIT 1
EVALUATION OF B_4C CENTER FINGER
CONTROL ELEMENT ASSEMBLIES (CEA) REV.2

INTRODUCTION:

Recent inspections at the Maine Yankee Nuclear Plant identified three Control Element Assemblies (CEA's) with missing center finger end caps and one CEA with a circumferential crack in the center finger end cap weld region. The failed CEAs are of an old Combustion Engineering CEA Design, which has the center finger with B_4C pellets extending to the tip. The term "old" CEA is defined in the context of this discussion as a CEA of St. Lucie Unit 1 is currently operating with 17 old CEA's.

EVALUATION:

The St. Lucie Unit 1 inspection results definitively indicate no CEA end cap failures or circumferential cracking for an inspection group which is over three times the size of, and represents the projected exposures of, the old CEA's currently in operation. In addition, review of CEA manufacturing records indicates that design and construction of CEAs were in accordance with approved materials, test, inspections, procedures, and specifications. Therefore, the probability of CEA inoperability during Cycle 10 is considered low. Nonetheless, the following augmented CEA surveillance program will be in place for the remainder of Cycle 10 to further reduce the likelihood of operation with an inoperable CEA.

1) Full stroke CEA exercising after trips and cold shutdowns will ensure CEA operability following the most probable precursor to failure, namely a thermal transition.

2) Quarterly full stroke CEA exercising will provide further assurance that the plant will not operate with a failed CEA.

3) Incore flux monitoring will be performed semi-weekly to provide detection of a gross CEA failure, and thus providing further assurance that CEAs are not failed and that the plant is operating in a safe manner.

Further, an evaluation of normal plant operations indicates that continued operation with the 17 old CEAs does not affect the plants ability to achieve safe shutdown.

Therefore, based on the surveillances and evaluations discussed, St. Lucie Unit 1 is capable of safe operation with 17 old design CEAs until the end of the current fuel Cycle (EOC-10)/ The conclusion has received the concurrence of the Combustion Engineering Nuclear Safety Committee.

ST. LUCIE UNIT 1
SAFETY EVALUATION TO PERMIT A MODIFIED LITHIUM REACTOR COOLANT PROGRAM FOR ST. LUCIE UNIT 1, CYCLE 10

INTRODUCTION:

Lithium hydroxide is used to control the Reactor Coolant System (RCS) pH to maintain a zero coefficient of solubility for dissolved corrosion products, i.e. crud. This results in crud going into solution in hotter regions of the RCS (the core) and crud deposition occurring in cooler regions of the RCS (the steam generators). The overall core crud load would be reduced by preventing crud from depositing on the fuel surfaces. Although the crud load in the steam generators would increase, the new effect is to minimize the activation of corrosion products by reducing their residence time in the core. The coordinated lithium-boron control program followed in Cycle 9 was predicated on the zero-coefficient of solubility of the crud being at pH of 7.4, based on a crud composition of mostly cobalt and nickel substituted ferrites. The purpose of the program is to reduce the activation of crud and thereby reduce out of core radiation fields.

SAFETY EVALUATION:

The modified lithium program which minimizes the time the lithium concentration is above 2.2 ppm does not involve an unreviewed safety question and does not involve a change to the Technical Specifications.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

The modified lithium coolant chemistry program proposed for cycle 10 will result in a zircaloy corrosion rate that is lower than the corrosion rate observed following use of the elevated lithium coolant chemistry program in cycle 9, which was found to be within acceptable limits.

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

No changes to the fuel or to the fuel operating environment, other than increasing the lithium concentration, are being proposed.

The modified lithium RCS water chemistry program proposed for use in PSL 1 cycle 10 does not involve an unreviewed safety question as far as its effect on nuclear fuel performance is concerned. The accelerated corrosion effect on zircaloy from extended exposure to 3.5 ppm lithium will be mitigated by operating below this level. The resulting zircaloy oxidation will be within the fuel rod design criterion. As such, no safety analysis is impacted nor is any new analysis required.

ST. LUCIE UNIT 1
HPSI HEADER VALUE POSITION CHANGE

INTRODUCTION:

This safety evaluation will permit the opening of the eight high pressure safety injection (HPSI) header isolation valves to a full open position. The valves are currently set for approximately two-thirds open. The purpose of this modification is to increase HPSI system flow delivery to the reactor coolant system (RCS) whenever the HPSI pumps are actuated and, therefore, increase the margin between actual and required flows.

SAFETY EVALUATION:

Fully opening the HPSI system header isolation valves does not alter any other component or equipment. The safety function of the HPSI system is actually enhanced by this modification. Therefore, this modification has no adverse impact on the probability or consequences of an accident or malfunction previously evaluated.

Fully opening the HPSI system header isolation valves does not create the possibility of a new or different kind of accident or malfunction important to safety. The only plant system impacted by this modification is the HPSI system.

Fully opening the HPSI system header isolation valves does not increase the probability of exceeding a safety limit, since the increase in HPSI system flow does not require a change to any of the plants' Technical Specifications. The technical specifications reviewed for potential impact were those regarding maximum allowable heatup and cooldown rates with single HPSI pump (figure 3.1-1b), boration systems (3/4.1.2), P/T limits (3/4.4.9), emergency core cooling systems (ECCS)(3/4.5), and emergency core cooling systems design features (5.5). No changes to these technical specifications are required. Further, the safety analysis conclusions have not changed and the plant protection and engineered safety features systems setpoints remain unchanged. Therefore, there is no reduction on the margin of safety as defined in the bases of any technical specification. The system modification, which permit the HPSI system header isolation valves to be full open, can be performed under a 10 CFR 50.59 since it involves neither a change to the plant Technical Specifications nor an unreviewed safety question. Therefore, plant operation with the HPSI system header isolation valves fully open is not a safety concern.

ST. LUCIE UNIT 1
SAFETY EVALUATION FOR REACTOR COOLANT PUMP (RCP)
MOTOR BEARING LUBE OIL DRAIN VALVES

INTRODUCTION:

This safety evaluation is prepared to document the acceptability of the as installed configuration of the Unit 1 upper and lower RCP motor bearing lube oil level indication. Currently plant drawings do not call for the presence of a drain line between the isolation valve and the sight glass, which is the as installed configuration. All of the affected piping is quality group D piping, non-safety related and non-seismic design (seismically supported). The existing condition has no impact on plant safety or operation. A review of the plant Technical Specifications and the Safety Analysis Report has shown that there are no unreviewed safety questions or Technical Specification changes involved.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since the as installed configuration does not affect any accident initiating components. The drain lines do not impinge upon any RCS piping and are classified as non-safety, quality group D lines.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. The existing configuration does not alter the function or design of any existing components, and thus does not increase the possibility of their failure.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased since the existing configuration does not create a new path for uncontrolled radioactive releases. The drain lines will not adversely affect any radiation monitoring equipment or equipment which performs a containment isolation function.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created since the existing configuration will not inhibit or otherwise adversely affect the operation of any equipment important to safety. The affected piping is non-safety, quality group D piping and downstream of the bearing lube oil header isolation valve. The configuration of the drain line precludes the possibility of leakage due to a single failure.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created since the existing configuration does not add or affect any equipment capable of initiating an accident. The affected piping is non-safety, quality group D piping and serves no safety function.

The existing configuration does not reduce the margin of safety as defined in the basis for any technical specification since the existence of the drain line will not impact the operation of the RCP motor bearing lube oil system or the RCP itself.

ST. LUCIE UNIT 1

EVALUATION OF B_4C CENTER FINGER CONTROL ELEMENT ASSEMBLIES (CEA's)

INTRODUCTION:

Recent inspections at Maine Yankee identified three CEA's with center fingers missing end caps. Given that St. Lucie Unit 1 is currently operating with similar design CEA's (20 out of 73), an evaluation was performed to examine the impact of potential multiple CEA failures on plant operation. The evaluation is not complete, however, the results currently available provide added assurance that in the event of multiple CEA failures St. Lucie Unit 1 can be safely shut down. In addition, detection of CEA failures is likely through the use of CEA exercises and/or incore/excore monitoring.

EVALUATIONS:

Currently, no evidence exists that St. Lucie Unit 1 has experienced a CEA failure of any type. To the contrary, objective evidence exists which demonstrates that all 73 CEA's are functioning as designed.

During the spring refueling outage, the CEA's were shuffled with no indication of abnormal operation.

CEA surveillance for reactivity measurements (rod bank worth), involved movement of CEA's from their fully withdrawn to fully inserted position. All CEA's functioned normally and measured rod worths met acceptance criteria. For example, the difference between predicted and measured total CEA bank worth for the current cycle startup was 2.2% compared to an acceptance criteria of 10%.

Since the spring refueling outage, full insertion and withdrawal of the 73 CEA's has been performed prior to startup on 4/23/90, 5/9/90 and 5/24/90. Prior to startup on 6/14/90, all regulating banks underwent full insertion and withdrawal, and functioned normally. (Twelve of the reference CEA's are part of the regulating banks.)

The CEA's currently in the St. Lucie spent fuel pool have exposures of 4200-4400 EFPD, which is roughly equivalent to the nominal CEA exposures projected for EOC-10. Since no failures or abnormalities of the spent fuel pool CEA's have been observed, this indicates further assurance of the integrity of the incore reference CEA's. Inspection to confirm the condition of representative discharge CEA's is planned.

Initial criticality conditions were attained within acceptance criteria. For example, measured HZP critical boron differed from design calculation by 27 ppm compared to ±50 ppm acceptance criteria.

Upon reactor trips and/or cold shutdowns, reference CEA exercising (full insertion /withdrawal) will be conducted.

ST. LUCIE UNIT 1
INSTALLATION OF TEMPORARY MONITORING EQUIPMENT
ON THE 1A1 REACTOR COOLANT PUMP (RCP)

INTRODUCTION:

This Safety Evaluation is to provide justification for the temporary use of additional shaft displacement vibration monitoring equipment on the 1A1 Reactor Coolant Pump (RCP). This will allow additional data to be collected on the RCP performance during alignment and balancing of the shaft. The probes and associated support brackets for this temporary modification are allowed for use during any mode of operation for the facility. All other temporary monitoring equipment and temporary jumpers shall be removed prior to the unit entering startup operations (mode 2). The use of the additional monitoring equipment will have no adverse effect on any permanently installed equipment and will be disconnected prior to startup operations.

SAFETY EVALUATION:

A review of the plant Technical Specifications and the Safety Analysis Report has shown that no unreviewed safety questions and no Technical Specification changes are involved with this temporary modification.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since the additional vibration monitoring equipment has no effect on the operation nor will the structural integrity or function of the motor/pump be affected.

The consequences of any accident previously evaluated in the FUSAR have not been increased by the temporary vibration monitoring equipment or loss of control room vibration indication or alarm.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR has not been increased by these temporary modifications.

The probability of occurrence of a malfunction of equipment important to safety has not been increased by these temporary modifications. The modifications performed under this safety evaluation are to the vibration monitoring system of the 1A1 RCP.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created by this temporary modification. The vibration monitoring system is a passive system. The system performs no automatic controlling functions.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created as a result of these temporary modifications. The temporary loss of vibration indication and alarm for the RCP 1A1 does not create any new failure modes which could impact the operation of any equipment important to safety.

ST. LUCIE UNIT 1
INSTALLATION OF TEMPORARY VIBRATION MONITORING EQUIPMENT
ON 1A1 REACTOR COOLANT PUMP (RCP)

INTRODUCTION:

This Safety Evaluation is to provide justification for the temporary use of additional shaft displacement vibration monitoring equipment on the 1A1 Reactor Coolant Pump (RCP). This will allow additional data to be collected on the RCP performance during alignment and balancing of the shaft. The probes and associated support brackets for this temporary modification are allowed for use during any mode of operation for the facility. All other temporary vibration monitoring equipment and temporary jumpers shall be removed prior to the unit entering startup operations (mode 2). The use of the additional monitoring equipment will have no adverse effect on any permanently installed equipment and will be disconnected prior to startup operations.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since the additional vibration monitoring equipment has no effect on the operation nor will the structural integrity or function of the motor/pump be affected.

The consequences of any accident previously evaluated in the FUSAR have not been increased by the temporary vibration monitoring equipment or loss of control room vibration indication or alarm. The RCP vibration monitoring system is not used to mitigate the consequences of any accident evaluated in the FUSAR. Since the RCPs have not been functionally affected nor will the added equipment affect its structural integrity, the RCP's will continue to perform their intended purpose and provide adequate coastdown flow following an accident.

The probability of occurrence of a malfunction of equipment important to safety has not been increased by these temporary modifications.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created by this temporary modification. The vibration monitoring system is a passive system. The system performs no automatic controlling functions.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR has not been increased by these temporary modifications. The modifications performed under this evaluation for the temporary installation of additional vibration monitoring equipment and temporary loss of control room indication and annunciation of RCP vibration has no impact on the operation of any equipment. Since the RCP's have not been functionally affected nor will the added equipment affect its structural integrity, the RCP's will continue to perform their intended purpose and provide adequate coastdown flow following an accident.

The margin of safety as defined in the bases of any Technical Specifications has not been reduced.

ST. LUCIE UNIT 1
CONTAINMENT FAN COOLER UNQUALIFIED COATING

INTRODUCTION:

This safety evaluation addresses the presence of unqualified coatings on the 1A, 1B, 1C, and 1D Containment Fan Cooler coil flanges. New cooling coils were installed under PC/M 081-189 during the Unit 1 1990 Winter refueling outage. Upon inspection of the coils prior to installation, the coating on the coil flanges was determined to be improperly applied (i.e. - unqualified). Some of these unqualified coatings could not be removed and replaced due to their proximity to the copper coils and the resultant potential for damage of the copper coils.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since the unqualified coatings do not perform safety function and their failure during a LOCA will not adversely affect the function of any structure, system, or component important to safety, or affect any accident initiating events.

The consequences of an accident previously evaluated in the FUSAR have not been increased since failure of the unqualified coatings will not affect the function of any equipment required to mitigate the effects of an accident.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased since failure of the unqualified coatings will not alter the function of any structure, system or component important to safety, and thus does not increase the possibility of their failure.

The possibility of an accident of a different type than any evaluated previously in the FUSAR has not been created since the unqualified coatings do not perform a safety function and their failure during a LOCA will not adversely affect the function of any structure, system or component capable of initiating an accident

The possibility of a malfunction of equipment of a different type than any evaluated previously in the safety analysis report has not been created since failure of the unqualified coatings will not inhibit or otherwise adversely affect the operation of any structure, system, or component important to safety.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased since failure of the unqualified coatings cannot affect the performance of any structure, system, or component important to safety. The failed coatings cannot adversely affect the ECCS since they will not clog the containment sump, or affect the performance of ECCS pumps and containment spray nozzles.

The proposed modification does not reduce the margin of safety as defined in the basis for any Technical Specification since the unqualified coatings cannot affect the basis for any Technical Specification.

ST. LUCIE UNIT 2
SAFETY EVALUATION FDR USE OF SEALING COMPOUND ON VALVE I-V09252

INTRODUCTION:

The gasket sealing surface on the valve body hinge pin bore of valve V09252 (feedwater supply check valve to 2A steam generator) was weld repaired, in accordance with NCR-2-307. As a contingency in the event the valve continues to leak at this joint during startup, NCR-2-308 was initiated to request installation of Leak Repairs, Inc. capnuts and wire wrap. The valve is located such that installation of the capnuts would be difficult while the leak is in progress. If leakage is detected during startup, the sealant would be injected through the capnuts into the gasket area.

The purpose of this evaluation is to evaluate installation of the capnuts and wire wrap, and should leakage occur during startup, provide a method for temporarily repairing the body to hinge pin leak on valve I-V09252. The method of repair will be sealant injection. The valves shall be permanently repaired during the next refueling outage or outage of sufficient duration.

SAFETY EVALUATION:

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased. Failure of the injection seal is comparable to a gasket failure and is therefore encompassed by the original design bases.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this repair. FUSAR Section 15.2.5 discusses the large feedwater line break (18" line downstream of the check valve). Total failure of this gasket/sealant would in no way approach this scenario.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. The gasket and/or sealant does not perform a safety function. The sealant will be limited to the volume of the gasket area void and therefore, will not adversely affect operation of the valve.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created. The proposed repair does not provide a new mode of normal or emergency plant operation.

Chemistry limits are not altered and no other change is proposed to the plant design, modes of operation or assumptions in the basis for the Technical Specifications or Safety Analysis. Therefore, this repair does not reduce the margin of safety as defined in the basis for any technical specification.

ST. LUCIE UNIT 2
SAFETY EVALUATION FOR WIDE RANGE CONTAINMENT
LEVEL CHANNEL MODIFICATION

INTRODUCTION:

This Safety Evaluation allows temporary repair of the wide range containment level monitoring instrument channel L-07-13A. This repair is necessary as sensor 11 of LE-07-13A is no longer operational. In order to compensate for this condition, the electronics of LT-07-13A are to be repaired. This temporary circuit repair will provide for the proper operation of channel L-07-13A.

This evaluation documents the acceptability of the level transmitter's circuitry repair. The repair will not adversely affect the operation or the existing qualification of the containment level system.

SAFETY EVALUATION:

This temporary repair will have no impact on plant safety or operation. A review of the Plant Technical Specifications and the Safety Analysis Report has shown that this change is not an unreviewed safety question and does not require a change to the Plant Technical Specifications.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased. The inoperability of this system is not considered an initiating event in any accident scenario. The wide range containment water level monitoring channels are utilized solely for post accident monitoring purposes.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FUSAR has not been increased with this circuitry repair. The channel still provides monitoring of containment water level during an analyzed accident. The repair does not result in an increase in probability of a malfunction.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this repair. The wide range containment level channels will continue to monitor the water level in containment during an analyzed accident. The repaired channel L-07-13A will continue to provide post accident monitoring capabilities with the exception of decreased resolution between sensor 11 (approximately elevation 18'3") and sensor 12 (approximately elevation 20'2"0. Water level in this range will be indicated as the elevation of sensor 10 (approximately elevation 16'5"). The redundant channel L-07-13B will continue to provide post accident monitoring capabilities and will provide correct resolution between sensors 11 and 12. Possible inoperability of channel L-07-13B between sensors 11 and 12 will not adversely impact any operator actions associated with accident mitigation as no actions or decision points are anticipated to occur based on a containment water level condition between the elevations of sensors 11 and 12.

The possibility of an accident of a different type and any evaluated previously in the FUSAR has not been created. The repaired instrument loop provides only monitoring capability of wide range containment level during an analyzed accident and will operate as described in the preceding paragraph.

The proposed activity does not reduce the margin of safety as defined in the basis for any technical specifications as the repaired channel will continue to provide the necessary monitoring function of post accident containment water level as required by the Plant Technical Specifications.

ST. LUCIE UNIT 2
ATWS DIVERSE SCRAM SYSTEM

INTRODUCTION:

As part of the project to meet the requirements of the ATWS Rule, 10CFR50.62, a Diverse Scram System (DSS) has been designed by Eaton Consolidated Controls (ECC) and will be installed in the St. Lucie - Unit 2 Engineered Safety Features Actuation System (ESFAS) cabinets. The new DSS components will be comprised of printed circuit boards which will be integrated into the existing ESFAS circuitry and will be installed as a group of individual modules. In addition, the DSS modules are designed to replace all existing modules in the cabinets which have similar component functions, (i.e: bistable trip, isolation, two-out-of-four actuation logic, three-out-of-four block permissive logic, and automatic test functions). Therefore, the new modules are designed to be fully interchangeable with the existing ones and will be utilized for future replacements.

SAFETY EVALUATION:

The Plant Technical Specifications, Section 3/4.3.2 describe the Limiting Conditions for Operation and the surveillance requirements of the ESFAS instrumentation. For the purposes of the test, only Channel A for Containment Radiation-High, Channel A for Refueling Water Tank - Low, and the Automatic Test Inserter (ATI) module, will be utilized. As stated in the Technical Specifications, for both Containment Radiation-High and Refueling Water Tank-Low, two channels out of four are required for trip and three channels are required to be operable.

The probability of occurrence or the consequence of an accident of malfunction of equipment important to safety previously evaluated in the safety analysis report (FUSAR) will not be increased because one of four channels will be considered to be inoperable at any time. This leaves the channel functions in a two-out-of-three trip condition during the majority of the test, which is within the Plant Technical Specifications. For the short period of time that the function is in one-out-of-three logic (i.e. during bistable and isolation module changeout), the condition is more conservative than Plant Technical Specification limiting conditions.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report (FUSAR) will not be created because the trip logic conditions are those previously evaluated are specifically stated in the plant Technical Specifications. In addition, the cabinet circuit design separates and electrically isolates measurement channels from each other and safety channels from each other. Therefore, the operation of any other channel will not be adversely affected.

The margin of safety as defined in the basis for any Technical Specification is not reduced since the Technical Specifications allow for plant operations in a two-out-of-three trip logic for a specified time period.

ST. LUCIE UNIT 2
SAFETY EVALUATION FOR 2B CCW HEAT EXCHANGER FLANGE GASKET REV. 1

INTRODUCTION:

The purpose of this evaluation is to address the potentially degraded sealing capability of the tubesheet/channel flange gasket in the inlet and outlet water boxes of the 2B Component Cooling Water (CCW) Heat Exchanger (Ref. NCR #2-434, NCR # 8740-2694M) and to justify the use of the Arcor S-16/Arc-Thane joint coating system to enhance the sealing of the joint for Cycle 6 operation.

Repair welding performed to resolve NCR # 2-428 and NCR 2-434 slightly warped the tubesheet flanges at the 12 O'clock position, which may impact the ability of the existing gasket to seal the the subject flange joints. PC/M 350-290M REV. 2 provided for coating the tubesheet/channel flange joint with the Arcor S-16/Arc-Thane joint coating system to allow for thermal movements and prevent sea water from contacting the carbon steel channel flange. The Arcor S-16/Arc-Thane joint coating system will enhance the capability of the gasket/flange joint configuration to provide a proper seal, particularly in the area of the detected warpage.

SAFETY EVALUATION:

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. This condition does not alter the function of any existing components, and thus does not increase the possibility of their failure. The addition of the Arcor S-16/Arc-Thane system in the 2B CCW heat exchanger tubesheet/channel flange joint provides an enhancement to the existing joint configuration to ensure the joint is sealed and no leakage will occur.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since this condition doe not affect any accident initiating components. The 2B CCW heat exchanger is not an accident initiating component.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased since this condition does not have a detrimental effect on any safety related equipment or components.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created since this condition will not inhibit or otherwise adversely affect the operation of the CCW or ICW systems. The components of the condition are in compliance with the Safety Analysis Report requirements for the system elements.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this condition since this condition does not have a detrimental affect on any equipment required to mitigate the effects of an accident. Failure of the gasket in the warped area on the 2B CCW heat exchanger flange would allow only minor leakage compared to the total ICW flow rate. The leakage would be on the ICW outlet of the CCW heat exchanger and therefore would not affect the ability of the CCW heat exchanger to perform its safety related functions. Should leakage occur on the ICW inlet, sufficient margin is available to perform the safety related function as addressed in Section 4.0. Failed coating pieces would be carried through the system and be released into the discharge canal without affecting the operation of the system.

This condition does not reduce the margin of safety as defined in the basis for any technical specification since this condition is only to a joint configuration on the 2B CCW heat exchanger. No changes are being made to the system design, modes of operation or assumptions in the bases for the Technical Specification or the FUSAR.

ST. LUCIE UNIT 2
LEAK REPAIR OF VALVE 2I-V08111

INTRODUCTION:

Valve 2I-V08111 is leaking at the valve body to bonnet gasket connection as identified by NCR 2-335. Leak Repairs, Inc. sealant will be injected to stop the leakage by the drill and tap method.

The purpose of this evaluation is to evaluate the temporary installation of the 1/8" injection valve for injection of the sealant, and installation of a pipe plug subsequent to injection. The valve bonnet will be drilled in the area of the bolt circle, at an angle to intersect the gasket sealing surface. The valve shall be permanently repaired during the next refueling outage or outage of sufficient duration.

SAFETY EVALUATION:

The use of this method to repair the subject valves will have no impact on plant safety or operation. A review of the plant Technical Specifications and the Safety Analysis Report has shown that there are no unreviewed safety questions or Technical Specification changes involved.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased. Failure of the injection seal and plug is comparable to a gasket failure and is therefore encompassed by the original design bases and accident analysis described in FUSAR Section 15.1.6.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this repair. FUSAR Section 15.1.6 discusses the main steam line break, which results in the maximum steam generator blowdown rate through 6.36 sq ft (36" line between the steam generator nozzle and the flow venturi). Total failure of this gasket/sealant would not approach this scenario.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased. The valve is required to maintain pressure boundary. Its ability to do so will not be affected by this repair since the bolt loadings are not affected by the injection of sealant. The gasket and/or sealant does not perform a safety function. The sealant will be limited to the volume of the gasket area void and therefore, will not adversely affect operation of the valve or any components in the safety related portion of the main steam system.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created. The proposed repair does not provide a new mode of normal or emergency plant operation.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created. Limiting the injection of sealant to just interacting outside the valve pressure boundary precludes a malfunction of a different type. Leakage of sealant into the main steam system is precluded by limiting sealant injection volume to the volume of the gasket area void, and the method of injection, thus downstream components will not be adversely affected.

Chemistry limits are not altered and no other change is proposed to the plant design, modes of operation or assumptions in the basis for the Technical Specifications or Safety Analysis. Therefore, this repair does not reduce the margin of safety as defined in the basis for any Technical Specification.

ST. LUCIE UNIT 2
SAFETY EVALUATION FOR I-FCV-23-3 AND 5 LEAK REPAIR

INTRODUCTION:

PSL-2 Blowdown Containment Isolation Valves I-FCV-23-3, 5 have had a history of leakage problems. Numerous leak repairs have been performed on these Steam Generator Blowdown system containment valves on both units.

Steam Generator Blowdown is utilized to control steam generator secondary side water chemistry, monitor secondary side radioactivity for any primary to secondary leakage, reduce the steam generator blowdown contaminants to an acceptable level prior to discharge to the environment, and provide blowdown system containment isolation capability. The portion of the piping and valves at the containment penetrations are seismic to ensure containment integrity following a containment isolation signal.

The purpose of this evaluation is to provide a method for temporarily repairing body to bonnet leaks on valves I-FCV-23-3 and 5. The method of repair will be sealant injection. The valves shall be replaced or permanently repaired during the next scheduled outage or other suitable time period. These valves are nonisolable and normally open.

SAFETY EVALUATION:

The use of this method to repair the subject valves will have no impact on plant safety or operation. The failure of any component in the Steam Generator Blowdown System does not affect safe shutdown of the plant.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased. These valves are not accident initiating components. These valves serve to isolate containment in the event of an accident.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this repair. The valve is required for maintaining containment isolation, and its ability to do so will not be affected.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created. The proposed repair does not provide a new mode of normal or emergency plant operation. By limiting the injection of sealant to just interacting outside the valve pressure boundary precludes a malfunction of a different type.

Chemistry limits are not altered and no other change is proposed to the plant design, modes of operation or assumptions in the basis for the Technical Specifications or Safety Analysis. Therefore, this repair does not reduce the margin of safety as defined in the basis for any technical specification.

ST. LUCIE UNIT 2
INSTALLATION OF BLIND FLANGE ON OUTLET OF PURGE
EXHAUST VALVE FCV-25-6

INTRODUCTION:

The containment isolation system provides the means of isolating fluid systems that pass through containment penetrations such that any radioactivity that may be released to the containment atmosphere following a postulated Design Basis Accident (DBA) is confined.

Table 6.2-52 and Fig. 6.2-69 of the St. Lucie Unit 2 FUSAR shows isolation valves FCV-25-4, -5 and -6 are installed in series in the 48" containment purge exhaust systems and are located inside containment, in the annulus, and outside the shield wall, respectively. Valves FCV-25-4 and -5 provide a double isolation for the containment penetration. Valves FCV-25-5 and -6 provide a double isolation for the shield wall penetration. FUSAR, Table 6.2-53 shows these valves are normally closed. Valves FCV-25-4 and -5 are also listed as Containment Isolation Signal (CIS) valves in the St. Lucie Unit 2 Technical Specification, Table 3.6-2 which is in effect for modes 1, 2, 3 and 4.

SAFETY EVALUATION:

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the containment purge exhaust system is neither required to function following a postulated Design Basis Accident nor is it required for the operational design of any system. Containment purge during normal plant operation is performed by the 8" containment purge system.

The margin of safety as defined in the bases for any Technical Specification is not reduced by this change because after implementing this change, the blind flange of valve FCV-25-6 provides a second isolation boundary and replaces the isolation function of FCV-25-5 which has repeatedly leaked during LLRT. This change is within the action items stated in the St. Lucie Unit 2 Technical Specification, Section 3.6.3.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the containment purge exhaust system is not a safety related system and is neither required to operate during normal plant operation nor after a Design Basis Accident. However, it is required to isolate the Containment after the Accident. The installation of a temporary blind flange on the exhaust side of valve FCV-25-6 will provide a second isolation boundary and replace the isolation function of FCV-25-5. The piping associated with valve FCV-25-6 and Penetration P-10, the weld between the closure plate and shield building anchor plant ring and penetration sleeve were evaluated for the additional seismic and dead weight loads of the newly designed flange and existing valve FCV-25-6 associated piping. They were found to be adequate for the additional loads. Application of sealant in the valve packing of valves FCV-25-5 and 6 will not adversely affect the normal function of the valves and will enhance the ability of the valves to perform its design function.

ST. LUCIE UNIT 2
SAFETY EVALUATION FOR WIDE RANGE
CONTAINMENT LEVEL CHANNEL MODIFICATION

INTRODUCTION:

This Safety Evaluation allows temporary modification of the wide range containment level monitoring instrument Channel L-07-13A. This alteration is necessary as sensor 11 of LE-07-13A is no longer operational. In order to compensate for this condition, the electronics of LT-07-13A are to be modified. This temporary circuit alteration will provide for the proper operation of channel L-07-13A.

This evaluation documents the acceptability of the level transmitter's circuitry alteration. The modification will not adversely affect the operation or the existing qualification of the containment level system.

SAFETY EVALUATION:

This temporary modification will have no impact on plant safety or operation. A review of the Plant Technical Specifications and the Safety Analysis Report has shown that this change is not an unreviewed safety question and does not require a change to the Plant Technical Specifications.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased. The failure of this system is not considered an initiating event in any accident scenario. The wide range containment water level monitoring loops are utilized solely for post accident monitoring purposes.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FUSAR has not been increased with this circuitry modification. The channel still provides monitoring of containment water level during an analyzed accident. The modification does not result in an increase in probability of a malfunction.

The consequences of an accident previously evaluated in the FUSAR have not been increased by this modification. The wide range containment level loops will continue to monitor the water level in containment during an analyzed accident. The modified channel L-07-13A will continue to provide post accident monitoring capabilities with the exception of decreased resolution between sensor 11 (approximately elevation 18'3") and sensor 12 (approximately elevation 20'2"). Water level in this range will be indicated as the elevation of sensor 10 (approximately elevation 16'5"). The redundant channel L-07-13B will continue to provide post accident monitoring capabilities and will provide correct resolution between sensors 11 and 12. Possible inoperability of channel L-07-13B between sensors 11 and 12 will not adversely impact any operator actions associated with accident mitigation as no actions or decision points are anticipated to occur based on a containment water level condition between and the elevations of sensors 11 and 12.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased. Because the channel will continue to monitor post accident containment water level as described in the previous paragraph, the consequences of a malfunction have not been changed.

The possibility of a malfunction of a different type than any evaluated previously in the FUSAR has not been created as this temporary modification does not introduce any new failure modes to the post accident containment level monitoring system.

The proposed activity does not reduce the margin of safety as defined in the basis for any technical specifications as the modified channel will continue to provide the necessary monitoring function of post accident containment water level as required by the Plant Technical Specifications.

ST. LUCIE UNIT 2
INSTALLATION OF BLIND FLANGE ON FCV-25-5

INTRODUCTION:

The containment isolation system provides the means of isolating fluid systems that pass through containment penetration such that any radioactivity that may be released to the containment atmosphere following a postulated design basis accident (DBA) is confined. Isolation valve FCV-25-5 has experienced repeated leak problems during local leak rate testing (LLRT). To correct this problem, a specially designed blind flange will be installed on the exhaust side of valve FCV-25-6 to provide a second isolation boundary and replace the isolation function of valve FCV-25-5.

SAFETY EVALUATION:

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The containment purge exhaust system is not a safety related system and is neither required to operate during normal operation nor after a design basis accident. The installation of a temporary blind flange on the exhaust side of valve FCV-25-6 will provide a second isolation boundary and replace the isolation function of FCV-25-5.

The possibility for an accident or malfunction of a different type that any evaluated previously in the safety analysis report is not created. Because the containment purge exhaust system is neither required to function following a DBA nor is it required for the operational design of any system. Containment purge during normal plant operation is performed by the 8" containment purge system. The margin of safety as defined in the bases for any technical specification is not reduced by this change. This change is within the action items stated in the St. Lucie Unit 2 Technical Specification, Section 3.6.3.

The temporary blind flange on the outlet of purge valve FCV-25-6 is acceptable in that it replaces the isolation function of valve FCV 2-5. The enhancement of valve packing on valves FCV-25-5 and 6 provide additional assurance for the leak tightness.

ST. LUCIE UNIT 2
INSTALLATION OF BLIND FLANGE ON FCV-25-6

INTRODUCTION:

The containment isolation system provides the means of isolating fluid systems that pass through containment penetrations such that any radioactivity that may be released to the containment atmosphere following a postulated Design Basis Accident (DBA) is confined.

Table 6.2-52 and Fig. 6.2-69 of the St. Lucie Unit 2 FUSAR shows isolation valves FCV-25-4, -5 and -6 are installed in series in the 48" containment purge exhaust systems and are located inside containment, in the annulus, and outside the shield wall, respectively. Valves FCV-25-4 and -5 provide a double isolation for the containment penetration. Valves FCV 25-5 and -6 provide a double isolation for the shield wall penetration. FUSAR, Table 6.2-53 shows these valves are normally closed. Valves FCV-25-4 and -5 are also listed as Containment Isolation Signal (CIS) valve in the St. Lucie Unit 2 Technical Specification, Table 3.6-2 which is in effect for modes, 2, 3 and 4.

SAFETY EVALUATION:

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the containment purge exhaust system is neither required to function following a postulated Design Basis Accident nor is it required for the operational design of any system. Containment purge during normal plant operation is performed by the 8" containment purge system.

The margin of safety as defined in the bases for any Technical Specification is not reduced by this change because after implementing this change, the blind flange of valve FCV-25-6 provides a second isolation boundary and replaces the isolation function of FCV-25-5 which has repeatedly leaked during LLRT. This change is within the action items stated in the St. Lucie Unit 2 Technical Specification, Section 3.6.3.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The containment purge exhaust system is not a safety related system and is neither required to operate during normal plant operation nor after a Design Basis Accident. However, it is required to isolate the Containment after the Accident. The installation of a temporary blind flange on the exhaust side of valve FCV-25-5. The piping associated with valve FCV-25-6 and Penetration P-10, the weld between the closure plate and shield building anchor plant ring and penetration sleeve were evaluated for the additional seismic and dead weight loads of the newly designed flange and existing valve FCV-25-6 associated piping. They were found to be adequate for the additional loads. Application of sealant in the valve packing of valves FCV-25-5 and 6 will not adversely affect the normal function of the valves and will enhance the ability of the valves to perform its design function.

ST. LUCIE UNIT 2
SAFETY EVALUATION FOR 2B CCW HEAT EXCHANGER FLANGE GASKET

INTRODUCTION:

The purpose of this evaluation is to address the potentially degraded sealing capability of the tubesheet/channel flange gasket in the outlet water box of the 2B Component Cooling Water (CCW) Heat Exchanger (Ref. NCR #8740-2694M) and to justify the use of the Arcor S-16/Arc-Thane joint coating system to enhance the sealing of the joint for Cycle 6 operation.

Repair welding performed to resolve NCR # 2-428 slightly warped the tubesheet flange at the 12 O'clock position, which may impact the ability of the existing gasket to seal at the subject flange joint. PC/M 350-290M Rev. 2 provided for coating the tubesheet/channel flange joint with the Arcor S-16/Arc-Thane joint coating system to allow for thermal movements and prevent sea water from contacting the carbon steel channel flange. The Arcor S-16/Arc-Than joint coating system will enhance the capability of the gasket/flange joint configuration to provide a proper seal, particularly in the area of the detected warpage.

SAFETY EVALUATION:

This condition will have no impact on plant safety or operation. A review of the plant Technical Specifications and the Safety Analysis Report has shown that there are no unreviewed safety questions or Technical Specification changes involved since this condition does not hinder or change the operation of any components or systems.

The possibility of a malfunction of a different type than any evaluated previously in the safety analysis report has not been created since this condition will not inhibit or otherwise adversely affect the operation of the CCW or ICW systems. The components of the condition are in compliance with the Safety Analysis Report requirements for the system elements.

The possibility of an accident of a different type than any evaluated previously in the safety analysis report has not been created since this condition does not add or affect any equipment capable of initiating an accident. This condition only affects the 2B CCW heat exchanger flange joint.

The consequences of a malfunction of equipment important to safety previously evaluated in the FUSAR have not been increased since this condition does not have a detrimental effect on any safety related equipment or components.

The probability of occurrence of a malfunction of equipment important to safety analysis report has not been increased. This condition does not alter the function of any existing components, and thus does not increase the possibility of their failure.

The probability of occurrence of an accident previously evaluated in the FUSAR has not been increased since this condition does not affect any accident initiating components. The 2B CCW heat exchanger is not an accident initiating component.

This condition does not reduce the margin of safety as defined in the basis for any technical specification since this condition is only to a joint configuration on the 2B CCW heat exchanger. No changes are being made to the system design, modes of operation or assumptions in the bases for the Technical Specification or the FUSAR.

ST. LUCIE UNITS 1 & 2
EVALUATION OF DELETION OF SPECIFIC INSTRUMENT
REFERENCES FROM THE EMERGENCY ACTION LEVELS

INTRODUCTION:

An evaluation of the potential effects of deleting specific instrument readings from the St. Lucie Plant Emergency Plan has been conducted in accordance with the requirements of 10 CFR 50.59 and 10 CFR 50.54(q.)

The proposed deletion was determined to require review in accordance with the requirements of 10 CFR 50.59 as it involved a change to a requirement specifically referenced in the St. Lucie Unit 2 Safety Evaluation Report, NUREG-0843, dated October, 1981. Section 13.3.2.4, "Emergency Classification System," specifically references NUREG-0654, and the preparation of EALs which use"...specific instrumentation, parameters and equipment status."

SAFETY EVALUATION:

Deletion of the specific instrument references in the Emergency Action Level (EAL) Tables in the St. Lucie Plant Emergency Plan does not impact the probability of occurrence of an accident previously evaluated as the EAL tables are not assumed to play any role in event initiation in the St. Lucie Units 1 and 2 Updated Final Safety Analysis Report (FUSARs). Deletion of the specific instrument references in the Emergency Plan does not impact the probability of the malfunction of any equipment important to safety as the EAL tables do not affect equipment operation in any way.

The potential consequences of an accident are not affected as the proposed change does not increase any hazard to the health and safety of the general public in any way. The existing emergency operating guidelines technology is based upon CEN-152, Revision 3, "Combustion Engineering Emergency Operating Guidelines." The accident mitigation strategy employed in CEN-152 is based upon a systematic approach to plant operations based on a hierarchy of operational protective actions. These actions are directed at minimizing the consequences of an event and, once fulfilled, ensure proper control of the event in progress. Such actions are termed "safety functions", and are defined as conditions or actions that prevent core damage or minimize radiation release to the public. Fulfillment of a complete set of safety functions ensures proper operator control of the event and, therefore, ensures that the health and safety of the public is not threatened. As no changes are being made to any of the operational parameters affecting the ability to meet any safety function in any emergency operating procedure, the existing analyzed potential dose rates at the site boundary will not be affected by the proposed change to the Emergency Action Level Table in the Emergency Plan.

Therefore, the proposed change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

There are no facets of the Emergency Plan EAL Table which play any role in the operation of plant equipment, or in the determination of general plant operational guidelines; hence, a change to this table cannot be assumed to be an initiating factor in any accident analysis. Changes to the EAL Table

will therefore not involve the possibility of creation of an accident or malfunction of a different type than any previously evaluated in the safety analysis report.

The EAL table is not referenced in the basis for any Technical Specification for either St. Lucie Unit 1 or St. Lucie Unit 2. Therefore, changing the EAL table will not reduce the margin of safety as defined in the basis for any technical specification.

ST. LUCIE UNITS 1 & 2
NUREG-0737 ITEM II.D.1 PRESSURIZER SRV AND PORV
DISCHARGE PIPING STRESS ANALYSIS

INTRODUCTION:

Item II.D.1 of NUREG-0737 required that licensees examine the functional performance capabilities of PWR pressurizer safety, relief and block valves and verify the integrity of the associated piping systems for normal, transient and accident conditions. The NRC issued the staff's Safety Evaluation (SE) and associated Technical Evaluation Reports (TER's) for the St. Lucie Units 1 and 2 response to Item II.D.1 of NUREG-0737 on May 11, 1990. The NRC's SE concluded that FPL met all of the II.D.1 requirements with the exception of those items identified in Section 5.2 of the TER's. The majority of these items were resolved via the Reference 1 transmittal. The purpose of this transmittal is to resolve the remaining open item (Item 8 of the TER's) associated with Item II.D.1.

Item 8 of the NRC TER's identified that the original St. Lucie Units 1 and 2 piping stress analyses did not follow the thermal-hydraulic modeling recommendations presented in EPRI document NP-2479, "RELAP5 Calculation of SRV Piping Loads". To resolve this issue, new RELAP5 models were developed for both St. Lucie Units 1 and 2 incorporating the EPRI modeling recommendations. These RELAP5 results were then utilized in inputs into revised piping stress analyses.

SAFETY EVALUATION:

1. The St. Lucie Unit 1 piping satisfies all of the applicable requirements of ANSI B31.7 (1969) and B31.1 (1967). All the associated St. Lucie Unit 1 piping supports (with the exception of support #RC-005-34B) are acceptable for the revised design loads. Support #RC-005-34B has been shown to be operable/functional for its revised design loads for the remainder of the present operating cycle (Cycle 10). The operability evaluation for support #RC-005-34B can be found in the attached evaluation JPN-PSL-SEMJ-90-055, Revision 0. Support #RC-005-34B will require modification during the 1991 St. Lucie Unit 1 refueling outage.

2. The St. Lucie Unit 2 piping satisfies all of the applicable requirements of ASME Section III (through summer 1973 addenda) and B31.1 (1967). Additionally, all associated St. Lucie Unit 2 piping supports have been shown to be acceptable for the revised design load.

3. The St. Lucie Units 1 and 2 pressurizer nozzles have been evaluated for the revised nozzle loads and these revised loads produce stresses that remain within code allowables under all loading conditions.

4. The St. Lucie Units 1 and 2 pressurizer safety valves and PORV's calculated discharge flange bending moments remain within those values obtained through the EPRI valve test program or values previously approved by the NRC. Therefore, operability of the safety valves and PORV's is demonstrated.

ST. LUCIE UNITS 1 & 2
OCEAN INTAKE SYSTEM VELOCITY CAP PANEL REMOVAL

INTRODUCTION:

Precast concrete panels (approximately 12' x 19' x 1'-3" thick) for the St. Lucie Plant ocean intake pipeline velocity caps have experienced structural failures. One of the center panels for the 16 foot diameter pipeline velocity cap and one panel for the southernmost 12 foot diameter pipeline velocity cap has collapsed and broken apart. It is likely panel debris has fallen into the pipelines. The failures were documented in NCR's 1 - 324 and 1 - 328, which were initially dispositioned with engineering operability assessments concluding that the plant could be operated safely with the failed velocity cap panels. An evaluation was also prepared concluding that total failure of the velocity caps would have no adverse impact to nuclear plant safety.

SAFETY EVALUATION:

A detailed, systematic plan for mobilization, rigging, lifting and demobilization has been developed for removal of six (total) concrete panels from the two impaired velocity caps. One panel from the damaged 12 foot diameter pipeline velocity cap will be removed using a pedestal mounted crane on board an anchored barge.

The probability of occurrence of an accident previously evaluated in the safety analysis report is not increased. The probability of losing the primary source of cooling water will not increase for the panel removal operation.

The possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the safety analysis report is not created.

The velocity cap structures and ocean intake pipelines have no safety related function. In addition, sufficient cooling water will continue to be available for shutdown cooling by using the second, independent source of water (Big Mud Creek) if the primary source of water is not available. Therefore, modifications to the velocity caps cannot cause, contribute to, or become factors in a new type of safety-related equipment malfunction.

The margin of safety as defined in the basis for any Technical Specifications is not reduced.

Limitations on the minimum water level for the ultimate heat sink will not be affected.

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

ST. LUCIE UNITS 1 & 2
IMPLEMENTATION OF A PERMANENT PRIMARY CHEMISTRY CONTROL
PROGRAM WITH MODIFIED LITHIUM CONCENTRATIONS

INTRODUCTION:

Primary coolant chemistry is controlled during various plant operations and evolutions to: minimize corrosion rate of materials in contact with the reactor coolant, minimize excessive fouling at heat transfer surfaces, and minimize reactor plant radiation levels in component/access areas where maintenance may be required. To achieve these goals, critical parameters for control include both oxygen and pH. Dissolved oxygen is scavenged by hydrazine in the coolant while pH is maintained within a prescribed range via lithium (Li) in the form of lithium hydroxide. The ideal pH range (6.9 to 7.4) has been specified in conjunction with 0.2 to 2.4 ppm Li to provide a wide margin between the upper operational limit and the threshold for attack in the event any concentrating phenomena exist. This range reduces the corrosion and results in less dissolved corrosion products circulating in the reactor coolant. When properly coordinated, high lithium/pH promotes the deposition of corrosion products on cooler surfaces (e.g., steam generator downcorner area) rather than at hotter surfaces (e.g., fuel rods and core areas). The overall irradiated crud load would be reduced by preventing crud from depositing on the fuel surfaces. Although crud load in the steam generators would increase, the net effect is to minimize the actuation of corrosion products by reducing their core residence time.

SAFETY EVALUATION:

The revised primary chemistry program as described in this safety evaluation does not result in an unreviewed safety question or reduction in the margin of safety as defined by the St. Lucie Technical Specifications, nor have an adverse affect on plant safety or operation. Since this safety evaluation addresses chemistry limitations which provide corrosion protection to ensure the structural integrity of the fuel and of the RCS, this safety evaluation is classified as safety-related.

The modified lithium coolant chemistry program proposed for St. Lucie Units 1 and 2 beginning in Cycles 11 and 6, respectively, will result in a zircaloy corrosion rate that is lower than the corrosion rate from the use of the elevated lithium coolant chemistry program in Cycles 9 and 5.

Since fuel rod performance parameters will not change, exposure of the cladding to greater than 2.4 ppm and up to 3.5 ppm lithium, as part of the modified lithium coolant chemistry program, will not result in fuel corrosion or mechanical behavior either greater than, or different from , that previously considered in the input to any safety analysis. Therefore, the consequences of previously analyzed accidents are not increased. As the fuel is operated in the same manner as in previous cycles, the probability of an accident or malfunction of equipment also remains unchanged.

Implementation of this modified lithium program has been recently evaluated by the FPL Nuclear Fuel Group and the NSSS manufacturer, ABB-CE, without any negative effects. Structures, systems and components continue to meet original design criteria and limits in compliance with the FUSAR.

The proposed modified lithium program will not increase the incident of stress corrosion of PWSCC of the components wetted by primary coolant since testing performed on mill annealed steam generator tubing has not shown a correlation to PWSCC and high lithium chemistry, nor has any PWSCC observed at other nuclear plants due to elevated lithium chemistry.

The proposed increase in lithium level will not create a malfunction or a different failure mechanism than previously evaluated, since the corrosion rates will not increase for the plant components which contact the primary coolant or its letdown.

The margin of safety as described in the basis of any Technical Specification is not reduced because no changes in any safety analysis input or assumptions are required as a result of the proposed changes; nor are any changes to analysis methodology necessary to describe fuel rod behavior. As no inputs, assumptions, or methods have changed, the results of the precious safety analysis remain unchanged.

ST. LUCIE UNITS 1 & 2
BORIC ACID SECONDARY WATER CHEMISTRY

INTRODUCTION:

Two common causes of steam generator degradation are intergranular attack (IGA) and tube denting. Both are the result of corrosion mechanisms induced by the presence of impurities. These impurities become concentrated to detrimental levels in sludge piles and crevice regions. The use of boric acid on the secondary side, however, has been shown to mitigate the effects of the impurities.

The JNS chemistry staff has conducted a safety evaluation in accordance with JNS-QI-3.0 which implements the requirements of 10 CFR 50.59. This safety evaluation addresses the addition of boric acid to the PSL secondary systems, in particular, to PSL-1 as an inhibitor to the ongoing intergranular attack (IGA) of the steam generator allow 600 tuubing.

SAFETY EVALUATION:

The Limiting Condition for Operation (LCO) 3.4.5 Steam Generators, addresses the operability requirements of the steam generators. The associated surveillance requirements reflect the details of the tube inspection program. LCO 3.4.7, under the heading "Chemistry" refers only to reactor coolant chemistry. The lone chemistry related Technical Specification associated with the Secondary System is LCO 3.7.1.4. This LCO specifies a limit for specific activity in the secondary coolant system. Therefore, the proposed change does not affect this LCO.

Based upon this review, the addition of boric acid to the secondary cycle will not require a change to the Technical Specifications incorporated into the referenced licenses for St. Lucie Units 1 and 2 as there is no directly applicable Technical Specification. The corrosion rates on secondary system materials are low. Due to the low concentrations of boric acid throughout the secondary system, corrosion of external components as a result of leakage from the system will take significantly longer to occur and will be negligible. The consequences of potential accidents are bounded by the analyses of the FUSAR. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety will not be increased.

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

LCO 3.4.5 (Steam Generators) addresses the operability requirements of the steam generators. The basis of the associated surveillance requirements for inspection is to ensure that the structural integrity of the steam generator tubing is maintained such that primary to secondary leakage from both steam generators does not exceed 1.0 gpm. Accordingly, the plant is expected to be operated such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking.

The proposed change has been shown to mitigate the effects of corrosive impurities present within the steam generator. Therefore, the margin of safety as described in the basis for any Technical Specification is not reduced.

ST. LUCIE UNITS 1 & 2
SAFETY EVALUATION OF ROSEMOUNT SERIES 1153 & 1154
TRANSMITTER SUSCEPTIBLE TO OIL LEAKS

INTRODUCTION:

In the engineering evaluation for the use of Rosemount Model 1153 and 1154 transmitters at St. Lucie Units 1 & 2, certain batches of the Rosemount 1153 and 1154 series transmitters have been identified by Rosemount as being susceptible to a loss of oil in the sealed sensing cell.

SAFETY EVALUATION:

Out of the 42 suspect batch transmitters, 27 are in the storeroom, 10 are installed, 4 have been removed from service and 1 has not been located. The transmitters in the storeroom have been placed on QC hold. The one that has not been located has been verified not to be installed in any safety related applications at St. Lucie.

A total of 20 suspect batch transmitters were found installed in St. Lucie Units 1 & 2 based on Rosemounts February 7, 1989 and December 22, 1989 notification. Out of these 20, 11 transmitters have been replaced. There are 9 remaining suspect batch transmitters in Units 1 & 2 which will remain installed until replacements are available. There is no suspect batch transmitter installed in any safety related application including RPS, ESFAS, and AFAS out of these remaining 9 transmitters.

There is no conclusive evidence that the suspect lot of Rosemount transmitters that are installed at S. Lucie would fail to operate as designed. Failure of these transmitters would not preclude the safe shutdown of the Plant.