RE: St. Lucie Plant
Docket No. 50-389
10 CFR 50.59 Report

St. Lucie Unit 2
Report of Changes Made to the Facility
Under the Provisions of
10 CFR 50.59
for the period March 26, 1993
to April 22, 1994

NOTE: The safety evaluations in this report are chronologically arranged starting with those implemented more recently. Please note that the level of detail of safety evaluations from earlier years do not reflect current practices.

Plant Change/Modifications reportable pursuant to 10 CFR 50.59 for St. Lucie Unit 2

Engineering Package (PC/M Number)	Title
001-294	Fuel Load For 80 Assembly Cycle 8
018-294	Fuel Transfer Tube Blind Flange Bolt Reduction
038-294	Pressurizer Upper Head Split Instrument Nozzle Modification
004-293	Pressurizer Safety Relief Valve Discharge Piping Modification
121-293	NRC Generic Letter 89-10 Motor Operated Valve Thrust Values and Actuator Modifications
132-293	CV-2 Relay Replacement
134-293	Removal of the Movable Incore Detector System
177-293	VOTES Force Sensor Mounting on Motor Operated Valves, VOTES 100 Force Sensor Mounting Procedure replacement (GL 89-10)
183-293	ICW/CCW Pump Control Circuits
212-293	Feedwater Control System Level Lead/Lag Module Deletion
215-293	Steam Generator Tube Plugging and Staking
049-292	VOTES Force Sensor Mounting on Motor Operated Valves (GL 89-10)
053-292	Generator Inadvertent Energization Circuitry
087-292	Place the CVCS Charging Pumps and Boric Acid Make-up Pumps on the EDG Zero Load Block
163-292	Turbine Trip Solenoid Valve Test Loop Addition
268-292	ICW Lube Water Piping Removal and CW Lube Water Piping Renovation
298-292	Install Individual Cell Equalizers
510-291	NRC Generic Letter 89-10 MOV Thrust Values
120-290	Extraction Feedwater Heater Replacement
173-290	MOV Limit Switch Modification
124-289	SGBS Containment Isolation Valves Replacement
134-289	Agastat Relays Series 7000 Change-out
290-289	Station Blackout Unit 1/2 Cross-tie
311-289	Obsolete Smoke Detector Replacement
281-288	Reactor Cavity Seal Ring Water Fill Modification
065-287	ITT Barton Differential Pressure Switch Replacement
196-285	Analog Display System Graphic Display Spares
053-284	Installation of Schnoor Springs on the 2A and 2B AFW Pumps
085-284	Reactor Vessel Head Shielding

Engineering Package 001-294

ABSTRACT

This engineering package (EP), prepared in accordance with QI Supplement 3.1-8, provides the reload core design of St. Lucie Unit 2 Cycle 8 developed by Florida Power & Light Co. The Cycle 8 core was designed to yield a cycle length of 11,676 EFPH +/- 250 EFPH, based upon a nominal Cycle 7 length of 10,475 EFPH. Cycle 7 achieved an EOC exposure of 10,602 EFPH. This increased cycle 7 exposure will require a coastdown at EOC 8 to meet the Cycle 8 target cycle length.

The primary design change to the core for Cycle 8 is the replacement of 80 irradiated assemblies with 80 fresh Region K (CE-4) fuel assemblies. The fuel is arranged in a low leakage pattern with no significant differences between the Cycle 8 loading pattern and the Cycle 7 design. The mechanical design of Region K is identical to that of Region J (Cycle 7) and Region H (Cycle 6) reload fuel.

The safety analysis of this design was performed by Asea Brown Boveri Combustion Engineering Nuclear Power, Inc. (ABB/CE), using the HERMITE code, and was independently reviewed by Florida Power and Light Co. It has been determined that the operation of the Cycle 8 reload core does not pose an unreviewed safety question and can be implemented with no changes to the St. Lucie Unit 2 Technical Specifications. Therefore, prior NRC approval is not required for implementation.

The implementation of this EP will not adversely impact plant safety or operation.

Revision 1

The purpose of this revision is to include data into the Original package that was not available at the time of the initial issue. This data is required to support initial startup, power ascension and beginning of cycle full power operation.

Engineering Package 001-294

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The Cycle 8 reload at St. Lucie Unit 2 does not change the overall configuration of the plant. The mode of operation of the plant remains unchanged. The implementation of the HERMITE code also does not change the overall configuration of the plant. Therefore, the probability of occurrence of an accident previously evaluated in the SAR is not changed.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The Cycle 8 design remains bounded by the SAR analyses. The consequences of accidents analyzed in the SAR remain unchanged. The HERMITE code has been shown to have no significant impacts on the results of setpoints analysis. Therefore, Cycle 8 reload does not increase the consequences of accidents previously evaluated in the SAR.

C. Does the proposed activity increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

Plant refueling is a normal plant operation. Probabilities of equipment failure during refueling have already been incorporated into the plant design basis. Implementation of the St. Lucie Unit 2, Cycle 8 reload does not increase the probability of equipment malfunction. Therefore, the probability of occurrence of any equipment malfunction important to safety previously evaluated in the SAR will not increase.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

Since Cycle 8 limiting design parameters remain bounded by the existing analyses, consequences of accidents resulting from malfunction of equipment remain unchanged. In addition, the HERMITE code results, which are used in the generation of setpoints for the analog protection system, have been shown to be more accurate than previous methods. Therefore, the Cycle 8 reload does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

Engineering Package 001-294

Unreviewed Safety Question (USQ) Determination (Cont.)

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

Fuel reload is a normal plant evolution. Possible accidents have already been postulated and analyzed in the SAR. The implementation of the HERMITE code for generation of setpoints results in more accurate results. Thus, no new accidents are created. Therefore, the Cycle 8 reload does not create the possibility of an accident of a different type than any previously evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

Fuel reload is a normal plant evolution. Possible equipment malfunctions have already been postulated and analyzed in the SAR. As stated above, the use of HERMITE is an improvement over previous methodology. Thus, no new equipment failures are created. Therefore, the Cycle 8 reload does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

Since the Cycle 8 design parameters and safety analyses remain bounded by the existing analyses, and the HERMITE code has been shown to be an improvement over previous methodology, the margin of safety as defined in the basis for any Technical Specification is not reduced.

Engineering Package 018-294

ABSTRACT

The St. Lucie Mechanical Maintenance department requested a reduction from the 44 bolts that must be installed and torqued on the fuel transfer tube blind flange at the end of each refueling outage. The installation currently requires 1.5 shifts and 1 man-rem of exposure. By reducing the number of bolts installed, it is expected that personnel exposure will be reduced.

Based on an analysis of the St. Lucie Unit 2 bolt stress and flange stress, the number of bolts required can be reduced to 8, with an increase in maximum torque from 150 ft-lbs to 200 ft-lbs. The analysis was performed consistent with the original design requirements for an ASME III Subsection NE, class MC (metal containment) vessel. The bolt and 0-ring specifications remain the same. Actual bolt stress is below 30% of allowable stress.

The blind flange retains two 0-rings which form part of the containment isolation boundary for the fuel transfer tube penetration (# 25), and is subjected to a BYPASS/Type C leakage test per the Technical Specifications. The Local Leak Rate Test will validate that the 0-rings are properly seated with reduced bolting. This modification does not affect the Fuel Transfer Tube Nozzle expansion bellows.

This Engineering Package (EP) justifies and documents a bolt reduction for the fuel transfer tube blind flange, from 44 bolts to 8 bolts, with respect to the bolt and flange design loads.

The modification considered in this EP is classified as Safety Related because the containment penetration performs the safety related function of maintaining containment integrity during emergency and accident conditions. The safety evaluation has shown that this EP does not constitute an unreviewed safety question as defined in 10 CFR 50.59 and implementation of this EP does not require a change to the Plant Technical Specifications. Therefore, prior NRC approval is not required for implementation of this EP. Implementation of this EP will have no adverse impact on plant safety or operation.

Engineering Package 018-294

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. May the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The fuel transfer tube blind flange is not the initiator of, or a contributor to, a previously analyzed accident. The calculations show that with a minimum of eight bolts, the bolt stresses remain within the allowable values and the leak-tight sealing capability of the blind flange has been demonstrated. The design function of the reactor containment is maintained. Therefore, the probability of an accident previously evaluated in the SAR will not be increased.

B. May the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The design function of the transfer tube blind flange has been shown to be unaffected. The containment isolation function has not been degraded in any way. Since the function of the bolted blind flange connection has been shown to be maintained, the reduction in the number of transfer tube cover bolts will not affect any assumptions previously made in evaluating the consequences of an accident described in the SAR

C. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed modification uses existing equipment, along with replacement components required by standard maintenance practice, such as replacement of the 0-rings or bolts. Implementing this modification will not increase the probability of a common mode failure. It has been demonstrated that the proposed change should cause no change of overall reliability of the component. Therefore, the proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

Engineering Package 018-294

Unreviewed Safety Question (USQ) Determination (Cont.)

The proposed modification has been designed to satisfy the original design requirements of the fuel transfer tube closure and associated containment isolation device. Implementation of the design modification does not increase the possibility of a common mode failure. It has been demonstrated that the proposed change should cause no deterioration of overall reliability of the component. Therefore, the proposed change does not increase the consequences of a malfunction of equipment important to safety as previously evaluated in the SAR.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The subject modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. No new failure modes or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification .

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not interact spatially nor functionally with any structure, system or component important to safety other than the fuel transfer tube closure itself. As demonstrated previously, no new failure modes or conditions have been created by this modification. Therefore, the possibility of a malfunction of equipment important to safety which is a different type than previously evaluated by the SAR is not created by this modification.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specification requirements and Technical Specification bases are not affected by this modification. It has been demonstrated that the subject modification does not degrade the overall reliability of the fuel transfer tube closure and does not affect the ability of the component to perform its intended safety function. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 038-294

ABSTRACT

This engineering package (EP) provides for the modification of four St. Lucie Unit 2 pressurizer upper head 1" instrument nozzles. The instrument nozzles shown as "A", 'B", "or, and "D" on PC/M drawing JPN-038-294-001 serve the instrument lines (1"-RC-130, -105, -104 and -107, respectively) for the pressurizer pressure and level instruments. The instrument nozzles are being modified due to the discovery of indications in the internal nozzle welds. Note: Evaluation of root cause and the as left conditions of the internal weld indications will be addressed as part of NCR 2-584. A modification of the original partial penetration weld/nozzle design will be required to reduce the susceptibility to primary water stress corrosion cracking (PWSCC), to reduce residual stress and for ALARA consideration. The modification moves the partial penetration weld joint to the pressurizer outside surface where previously the joint was on the inside surface. Exposure to personnel during installation and welding of the nozzle will be reduced by both the modified design and by automation of the welding processes. Also, the modification will separate the instrument nozzle in the penetration into two segments. The resulting gap will allow the primary borated water from the RCS to come in contact with the pressurizer base metal in the penetrations.

The pressurizer upper head instrument nozzles provide taps for sensor inputs to monitor the level and pressure of reactor coolant in the pressurizer. The sensors provide input to several protective systems which function to automatically trip the reactor in the event of predetermined abnormal condition. Additionally, they furnish input to monitoring systems and instrumentation.

The pressurizer and pressurizer instrument nozzles are Quality Group A components and ASME Section III Safety Class 1, therefore this EP is classified as Safety Related (Ref. 5.5 and Table 5.2-1 in Ref. 5.1).

A review of the changes to be implemented by this EP was performed in accordance with the requirements of 10CFR50.59. As indicated in the Safety Evaluation (Section II), this EP does not involve an unreviewed safety question, nor does it require a revision to the Plant Technical Specifications. This modification will have no effect on plant safety and operation. Prior NRC approval is not required for the implementation of this EP.

Supplement 0 of this engineering package was for the partial implementation.

Revision 1

This revision removes the hold points concerning stress analysis calculation and the use of Code Case 2142. The piping stress analysis and structural analysis have been completed with acceptable results. Code Case 2142 has been approved by the NRC for this modification. Isometric drawings have been added to address the removal and re-installation of piping connected to the modified nozzles. Restraint drawings have been added to document editorial changes and correct discrepancies between the stress analysis and restraint drawings. Editorial changes have been made to the safety evaluation but the technical evaluation and conclusion of the original safety evaluation have not changed. This revision does not affect, amend, or change the Technical Specifications.

Engineering Package 038-294

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. May the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The probability of occurrence of an accident previously evaluated in the SAR will not increase because the structural integrity of the reactor coolant system pressure boundary is not adversely affected. The new configuration meets the requirements of the ASME code of record. The pressurizer instrument nozzles will be of the same acceptable material that was previously installed in the pressurizer. Welding of the weld buildup pads and partial penetration J welds will be controlled by weld procedures which have been reviewed and approved by FPL. Code Case N-432, Repair Welding Using Automatic or Machine Gas Tungsten - Arc Welding (GTAW) Temperbead Technique, Section XI, Division 1, will be used for the weld buildup pad to allow relocating the partial penetration weld joint from inside the pressurizer to the outside surface. Non-destructive testing will be performed to verify acceptability of the modification. The splitting of the pressurizer nozzles will create a gap and expose the pressurizer base material to the reactor coolant system fluid which is a boric acid solution. Exposure of the pressurizer ferritic base material to borated water may result in corrosion of the material. The corrosion rate of the ferritic base material is estimated to be 3 mil per year. The 3 mil per year corrosion rate equates to a few hundred years of plant operation prior to reaching the ASME reinforcement limit. Therefore, for this particular case the rate of corrosion is acceptable as this is approximately an order of magnitude greater than the remaining St. Lucie Unit 2 licensed operational lifetime (Ref. 5.6). Therefore, the probability of an accident previously evaluated is not increased by this activity.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The new nozzle design does not change, degrade, or prevent actions described in, or assumed to occur in the mitigation of any accident described in the SAR. Therefore, this activity does not increase the consequences of accidents previously analyzed in the SAR.

C. Does the proposed activity increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

Engineering Package 038-294

Unreviewed Safety Question (USQ) Determination (Cont.)

The function of the pressurizer instrument nozzles are to provide taps for sensor input to monitor pressurizer level or pressure. The pressurizer pressure and level sensors provide input to overall protective systems which automatically trip the reactor in the event of predetermined abnormal conditions. The modification to the pressurizer instrument nozzles meets the ASME code of record and does not effect the operation of these protective sensors. There are no new failure modes or system interactions associated with this modification. Therefore, this activity does not increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

This PC/M provides a fully qualified, functionally equivalent design. There is no impact to the functionality of equipment important to safety. The complete failure of an instrument nozzle is bounded by the LOCA analyses in the SAR. Therefore, this activity does not increase the consequences of a malfunction of equipment important to safety which was previously evaluated in the SAR.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This PC/M provides a functionally equivalent design. A leaking or complete failure of a pressurizer instrument nozzle will result in a small break LOCA which is analyzed in the SAR. Failure of a nozzle will also result in the loss of the associated sensor output. Sensor failure is an analyzed event in the SAR. There are no new failure modes or system interactions associated with this modification. Therefore, there is no possibility that an accident may be created that is different from any already evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

As previously stated, no new failure modes or system interactions are introduced. Exposure of the pressurizer ferritic base material to borated water may result in corrosion of the material. The corrosion rate of the ferritic base material is estimated to be 3 mil. per year. The 3 mil. per year corrosion rate equates to a few hundred years of plant operation prior to reaching the ASME reinforcement limit. Therefore, for this particular case the rate of corrosion is acceptable as this is approximately an order of magnitude greater than the remaining St. Lucie Unit 2 licensed operational lifetime. Therefore, this activity does not create a different type of malfunction of equipment important to safety from any already evaluated in the SAR.

Engineering Package 038-294

Unreviewed Safety Question (USQ) Determination (Cont.)

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specifications?

This activity does not change the design bases, functions or operations of any safety related equipment. There are no Technical Specification requirements nor their bases applicable to this modification. Therefore, this activity does not reduce the margin of safety as defined in the bases for the Technical Specifications.

Engineering Package 004-293

ABSTRACT

This Engineering Package (EP) provides design for modification of the discharge piping for pressurizer safety relief valves (PSRVs) V1200, V1201, and V1202. It is intended to reduce end loads on the valve outlet flanges to minimize potential PSRV seat leakage by modification of discharge piping supports. A pipe stress analysis has been performed and affected piping component and supports/restraints modified. NRC NUREG 0737 submittal has been reviewed and found to have not been affected.

Although the PSRV discharge piping serves no Safety Related functions, this EP is classified as Quality Related since the PSRV discharge piping is stress analyzed and seismically supported. The safety evaluation has shown that this EP does not constitute an unreviewed safety question and prior NRC approval is not required for implementation. The implementation of this EP does not require a change to the Plant Technical specifications and does not reduce the margin of safety for any Technical Specification.

The implementation of the EP will have no impact on plant safety or operation.

Revision 1

The design verification statement is changed to discuss the PCM statement "modifies one existing support". The support/restraint drawings affected by snubber settings were added to the Affected Drawing List.

Engineering Package 004-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The PSRV discharge piping is an auxiliary system which supports the PSRV design basis of overpressure protection for the Reactor Coolant System. The PSRV discharge piping is not an initiator of any accident previously evaluated in the SAR, and the piping and support integrity for conditions of normal operation, transients, and accidents remains valid. Thus, this proposed change does not increase the probability of the occurrence of any accident previously evaluated in the SAR.

B. Does the proposed chance increase the consequences of an accident previously evaluated in the SAR?

The PSRV discharge piping and support integrity for conditions of normal operation, transients, and accidents remains valid. Thus this proposed chance does not increase the consequences of any accident evaluated in the SAR.

C. Does the proposed change increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The modifications to the PSRV discharge piping supports are intended to result in reduced end loads on the PSRV outlet flanges, and thus this proposed change should reduce the probability of occurrence of leakage past the flanges of the PSRVs during normal operation. Therefore, this proposed change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed change increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The modifications to the PSRV discharge piping supports reduce the end loads on the PSRV outlet flanges, and thus this proposed change should reduce the leakage past the flanges of the PSRVs during normal operation. Therefore, this proposed change does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

Engineering Package 004-293

Unreviewed Safety Question (USQ) Determination (Cont.)

E. Does the proposed change create the possibility of an accident of a different type than any previously evaluated in the SAR?

The PSRV discharge piping is an auxiliary system which support the PSRV design basis of overpressure protection for the Reactor Coolant System. The PSRV discharge piping is not an initiator of any accident of a different type than previously evaluated in the SAR, and the piping and support integrity for conditions of normal operation, transients, and accidents remains valid. Therefore this proposed change does not create the possibility of an accident of a different type than any previously evaluated in the SAR.

F. Does the proposed chance create the possibility of a malfunction of equipment important to safety previously evaluated in the SAR?

The modifications to the PSRV discharge piping supports are to reduce the end loads on the PSRV outlet flanges, and thus this proposed chance should reduce the leakage past the flanges of the PSRVs during normal operation. The piping and support integrity for conditions of normal operation, transients, and accident remains valid. Therefore, this proposed change does not create the possibility of a malfunction of equipment important to safety previously evaluated in the SAR.

G. Does the proposed change reduce the margin of safety as defined in the basis for any Technical Specification?

The modifications to the PSRV discharge piping supports are intended to result in reduced end loads on the PSRV outlet flanges, and thus this proposed change should reduce the leakage past the flanges of the PSRVs during normal operation. The piping and support integrity for conditions of normal operation, transients, and accidents remains valid. This modification does not affect either the limiting conditions for operation, or the operability or surveillance requirements of the PSRVs, and this proposed modification should reduce the RCS leakage from the PSRVs which is a source of allowable Technical Specification leakage. Therefore, this proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification.

Engineering Package 121-293

ABSTRACT

NRC Generic Letter 89-10 requires that operating nuclear plants develop and implement a program to ensure that switch settings on all safety-related motor-operated valves (MOVs) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the plant's design basis.

Item a) of the Letter requires that the design basis for these MOVs be reviewed to determine the maximum system parameters (i.e. differential pressure, line pressure, flow) expected during both the opening and closing strokes for all postulated events for the valves within the scope of the Engineering Package. This has been completed and documented in FPL Calculations for the scope of valves in this EP.

Item b) of Generic Letter 89-10 requires that the licensee establish the correct MOV switch settings based on the previously determined maximum system parameters. All switches, including torque switches, torque switch bypass switches, position limit switches, position indication switches, thermal overloads, etc., have been considered. This design package provides the required thrust and torque values for 37 motor operated valves in addition to specific design information, as determined by calculation, necessary to replace actuator spring packs on certain MOVs. These MOVs are installed in the Reactor Coolant System, Chemical and Volume Control System, High and Low Pressure Safety Injection Systems, the Main Steam System and the Auxiliary Feedwater System. These systems are described in Sections 5.1, 9.3, 6.3, 10.3 and 10.5 of the St. Lucie Unit 2 SAR.

The valves in this PC/M were previously included in the scope of IE Bulletin 85-03. 28 of the 37 valves were also previously included in the scope of PC/M 510-291, but were not tested under that PC/M prior to as-building, and are therefore included herein.

Because the motor-operated valves associated with Generic Letter 89-10 are safety-related, or may affect safety-related systems, this engineering package has been classified as safety-related. A review of the changes to be implemented by the PC/M was performed against the requirements of 10CFR50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

Revision 1

Revision 1 to this PCM is issued to revise the actuator rating and minimum required torque for MV-08-14/15/16 and 17, correct typographical errors on drawings JPN-121-293-026 and 027, and revise drawings JPN-121-293-040 and 041 to reflect thrust values which include packing loads for all valves. This revision has no impact on the original conclusions of the safety evaluation.

Engineering Package 121-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serve.. .. determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not adversely affect any equipment whose ... .unction is postulated in the SAR to initiate an accident or prevent an accident from occurring. The modifications performed by the Engineering Package enhance the ability of the components to perform as intended during emergency and off normal conditions under maximum differential/line pressures.

Revising the switch setpoints or upgrading the structural integrity and performance of the equipment only serves to enhance the operation characteristics of the ... /s. As such, no new accident initiating events are created. Therefore, the modifica...s described in this Engineering Package do not increase the probability of valve failure, and thus the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

This modification does not adversely affect any structures, systems or components that function to deter the release of radioactivity or to provide post-accident shielding. The modifications performed by this Engineering Package do affect systems and components that are relied upon to mitigate accident consequences, and contain radioactive fluids. However, the modification improves the operational characteristics of the valves and improves the equipment's ability to function during an accident. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

C. Does the proposed a..vity increase the probability of occurrence of a malfunction of equ... nt important to safety previously evaluated in the SAR?

System operability is not being affected by the modifications to the MOVs identified in this Engineering Package. Valve operability will be enhanced by the prescribed modifications. The replacement of existing parts with similar parts, manufactured by the original equipment manufacturer for the specific purpose, and designed to meet all applicable regulatory and code requirements as the originals does not result in the

Engineering Package 121-293

Unreviewed Safety Question (USQ) Determination (Cont.)

introduction of any new failure mode or possible malfunction. Therefore, the modifications described in this Engineering Package do not increase the probability of the occurrence of malfunctions of any equipment.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

System operation is not affected by this modification. This modification does not interact spatially or functionally with any structure, system or component important to safety other than the valves and valve operators themselves. Although actuator and valve loadings may increase, the revised loads are within the published ratings for the components. Replacement components have been selected in accordance with the same design criteria as the original components. The modifications performed by this Engineering Package enhance the ability of the valves and valve operators to perform as intended during normal, off-normal, and emergency conditions under maximum differential pressures. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased by this modification.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. This modification provides increased design documentation, makes adjustments to components within their published operating range or makes replacements of equivalent parts. No new failure modes or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not interact spatially or functionally with any structure, system or component important to safety other than the valves and valve operators themselves. This modification does not alter the function or the design basis of any MOV. No new failure modes are created for the subject MOV's that can be postulated to cause a malfunction of equipment important to safety different than those previously analyzed in the SAR. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

Engineering Package 121-293

Unreviewed Safety Question (USQ) Determination (Cont.)

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specification requirements and Technical Specification Bases are not affected by this modification. The design bases of the valves and valve operators remains unchanged. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 132-293

ABSTRACT

This Engineering Package (EP) provides the design necessary to replace the existing Westinghouse type CV-2 undervoltage relays with the solid state relays manufactured by ABB, type 27N. These relays are a part of the undervoltage/degraded voltage (PSB-1) protective scheme. This modification was requested by the System Protection and Electrical Maintenance personnel to enhance the outage calibration activities. This EP will also replace the single 4.16 kV loss of voltage CV-2 relay with two ABB-27N relays connected for a coincident trip logic only one will be connected at this time due to the Technical Specification requirements. In addition, this EP will add to each of the existing 4.16 kV undervoltage protective relaying schemes a timer relay to provide adequate time for the test circuit to change state before re-arming the trip circuit. The total scope of the required modifications is as follows:

1. Replacement of twelve(12) CV-2 relays(6 in each of the PSB-1 cabinets) with the 27N relays, and their associated auxiliary relays with timing relays(four in each PSB-1 cabinet). Four(4) new test switches are also being added-two in each PSB-1 cabinet.

2. Replacement in each of the 4.16 kV safety switchgear (2A3 and 2B3) of the single loss of voltage CV-2 relays with two(2) 27N relays and a replacement of the two(2) auxiliary relays (one in each SWGR) in the test circuitry with Agastat EGPD relays with more contacts.

3. Addition of an Agastat E7022PA timing relay to the undervoltage/load shedding test circuitry in each of the safety 4.16 kV switchgear(2A3 and 2B3).

The undervoltage and degraded voltage protective relaying performs a safety related function. All the components and systems affected by this EP are safety related, therefore this PC/M is classified nuclear safety related.

Engineering Package 132-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

Relay setpoint changes and hardware modifications being implemented by this EP do not increase the probability of occurrence of an accident previously evaluated in the SAR, since the undervoltage protective relaying can not initiate an accident. The modifications will result in a more reliable undervoltage/degraded voltage protective scheme.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

All the new equipment being used is qualified for its intended function. The new setpoints have been determined to protect the Class 1E system equipment from possible damage due to unacceptably low voltage levels. Therefore, the proposed activity does not increase the consequences of an accident previously evaluated in the SAR.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The new equipment being installed by the proposed modification is qualified for its intended service. The new equipment (relays and switches) functions in the same manner as similar equipment already installed in the plant, and therefore it is as reliable as the existing equipment. Therefore, the proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

These proposed modifications change the hardware and the setpoints for the undervoltage/degraded voltage protective relays. The new setpoints were derived on basis of the latest system analysis utilizing approved and verified methodology. The new hardware being used has been qualified for its intended service and all the modifications were engineered to assure compliance with the original design basis. Therefore, the

Engineering Package 132-293

Unreviewed Safety Question (USQ) Determination (Cont.)

proposed activities do not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The overall protective functions of the existing undervoltage/degraded voltage protective relaying are not being changed, even though individual relay functions and setpoints were modified. The proposed modifications add/replace equipment qualified for its intended service. Changes have been engineered to assure compliance with the original design basis. New failure modes are not introduced by these modifications. Therefore, the proposed activity does not create the possibility of an accident of a different type than any previously evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The proposed modifications are engineered to the original design basis and utilize equipment qualified for its intended service. Failure modes have been evaluated and new failure modes are not created. Therefore, the proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The new relay setpoints meet the present Technical Specification requirements. Modifications supporting the setpoint changes are engineered to the original design basis and utilize components qualified for its intended service. Therefore, the proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

Engineering Package 134-293

ABSTRACT

This Engineering Package provides the engineering justification and the design details necessary for removal of the Unit 2 Movable Incore Detector System (MICDS) from the Incore Instrumentation System.

The Incore Instrumentation System is described in Section 7.7.1.1.8 of the St. Lucie Unit 2 FSAR and consists of 56 instrument assemblies which are inserted into selected fuel assemblies. Each of the instrument assemblies (fixed instrumentation) contains four (4) self powered rhodium detectors to monitor the neutron flux distribution within the reactor core. In addition to the Fixed Incore Detector System (FICDS), the original design included a Movable Incore Detector System (MICDS). The MICDS consists of two movable detectors (fission chamber type) and associated hardware to position either probe at any location within the 56 fixed incore instrument assemblies. The MICDS was provided as a backup to the fixed detector system to offer a neutron flux map independent of the FICDS.

Per Technical Specification 3.3.3.2, the MICDS may be used as an alternative means of declaring an incore detector location operable. However, the MICDS has never been used at St. Lucie to assist in meeting the operability requirements for the incore detector locations as specified in TS 3.3.3.2. In addition, the absence of correlation data between fission chambers and the Rhodium detectors precludes use of the MICDS as an analytical or calibration tool.

The major components of the MICDS are two (2) cable assemblies (movable fission chamber detectors), two (2) drive assemblies (detector positioning devices), and two (2) transfer assemblies (fuel bundle selecting devices). Each cable assembly consists of a hollow core helically wound cable with a detector tip that functions to place the detector at predetermined positions and transmit back the electrical signal. Each drive assembly functions to push or pull the cable assembly via guide tubes, thereby moving the detector into or out of the core. Each transfer assembly functions to route the cable assembly into any one of the 28 possible paths. The drive assemblies are physically located on the reactor vessel missile shield. The other equipment is located near the reactor vessel head.

Each refueling outage, the tubing out of the drive machines is disconnected and the drive machines are lifted out of the way by crane. Additionally in the past, all 56 head connections would have to be disconnected at the head flanges to allow lifting of the closure head lift rig. This work is an ALARA concern in addition to being critical path outage work. The above mentioned burden and the cost of maintaining the MICDS is not justified by any safety significance, operational enhancement, or other necessity. During the spring refueling outage of 1992, FPL evaluated the need for the MICDS under the 10 CFR 50.59 evaluation contained in PC/M 120-292 (Ref. III.5.3), but retained the MICDS option by maintaining critical equipment in place. The fixed incore instrument assemblies were however changed out. The new assembly design does not have a Dry Calibration Tube, which was used by the MICDS, but was replaced with a Central Member Assembly. The burden associated with maintaining the redundant MICDS has caused FPL to re-evaluate its necessity, and concludes there is no justification for continuing to retain the system. Removal of the MICDS will delete the use of the movable incore detectors as an alternative means of determining operability of an incore location, but will also eliminate the burden.

Engineering Package 134-293

ABSTRACT (Cont.)

The MICDS performs no safety related functions and is presently disconnected from the reactor. The MICDS however is included in the Technical Specifications, therefore, this Engineering Package is classified as Quality Related.

A safety evaluation of these modifications has been performed in accordance with 10 CFR 50.59. This evaluation concludes that elimination of the MICDS does not involve an unreviewed safety question and has no adverse effect on plant safety or operation. However, the MICDS option is discussed in Technical Specification 3.3.3.2. FPL received License Amendment #64, which removed the MICDS option from the Technical Specification.

Engineering Package 134-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The MICDS is a passive system designed only as an option for monitoring the local neutron flux within the core by performance of the flux mapping process. The lack of this passive system can not increase the frequency of occurrence of a neutron flux/power anomaly since the incore detectors are not accident initiators and they are being removed. Therefore, the proposed activity does not increase the probability of occurrence of an accident previously evaluated in the SAR.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The MICDS and the FICDS are considered subsystems of the Incore Instrumentation System. The MICDS is a backup system to the FICDS, and is not relied upon to provide any mitigating protection. The FICDS will continue to be used to adequately map the core for core performance information. Neither the MICDS, nor the FICDS are relied upon for safety in any FSAR Chapter 15 accident analysis. The excore detectors, which are unaffected by this proposed change, perform all accident mitigation functions for core protection. Therefore, the proposed activity does not increase the consequences of an accident previously evaluated in the SAR.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

Since the MICDS has no adverse interaction with any safety systems, removal of the system will not increase the challenges or the likelihood of failure for any equipment important to safety. Therefore, the proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

Engineering Package 134-293

Unreviewed Safety Question (USQ) Determination (Cont.)

Since the MICDS has no interaction with any safety systems, removal of the system will not cause any equipment important to safety to operate outside their specified design limits. Therefore, the proposed activity does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This system informs operators of the local neutron flux within the reactor core and serves no controlling function. Removal of the MICDS does not introduce any accident initiators. Therefore, the proposed activity does not create the possibility of an accident of a different type than any previously evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The MICDS has no interaction with any safety systems and will not produce any new failure types since equipment important to safety will continue to perform their safety functions and be unaffected by the absence of the MICDS. Therefore, the proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The subject TS (TS 3.3.3.2) provides the criteria for determining what represents an adequate flux map (percentage of incore locations operable and core symmetry requirements). The TS bases states, "the operability of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core". Additionally, the TS currently states an operable incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of three operable rhodium detectors or an operable movable incore detector capable of mapping the location. The basis for the TS is unaffected since deletion of the MICDS only removes the flexibility offered by an optional method of determining operability of the detector locations, and does not reduce the amount of monitoring locations available. Therefore, the proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification and a Proposed License Amendment (PLA) was processed and submitted to the NRC to delete the MICDS as an alternative method of meeting operability.

Engineering Package 177-293

ABSTRACT

NRC Generic Letter 89-10 requires that operating nuclear plants develop and implement a program to ensure that control switch settings on all safety-related motor-operated valves (MOV's) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. To comply with the requirements of the Generic Letter, torque switches in the opening and closing direction are set through the use of diagnostic equipment monitoring valve and actuator performance. The Liberty Technology Center, Inc. specifically developed the Valve Operation Test and Evaluation System (VOTES) to provide a simple-to-use and accurate method of testing and diagnosing the operation of motor operated valves. One principal feature of the system is the force measurement concept. Since the valve stem forces cause equal and opposite yoke reaction forces, measurement of yoke deflection is an accurate indicator of stem force over the entire valve stroke. VOTES diagnostic equipment to be used to evaluate MOV's senses the reactionary force produced in the yoke via a force sensor (i.e. strain gauge). This Engineering Package provides the details necessary to mount the force sensors on the yokes of motor operated valves that were not completed under PC/M 049-292.

Because the motor-operated valves associated with Generic Letter 89-10 are safety-related, or may affect safety-related systems, this engineering package has been classified as Safety Related. These changes have no adverse effect on plant operation or safety. A review of the proposed modification to be implemented by the PC/M was performed against the requirements of 10CFR50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

This PC/M is being issued to replace the VOTES 100 Force Sensor Mounting Procedure from B&W Nuclear Service Company with the St. Lucie Electrical Maintenance Procedure No. 0940078 Rev. 0. Furthermore, this EP will allow the work be performed on these valves which were not completed by PC/M 049-292 Rev. 0.

The addition of the VOTES Force Sensors to the affected valves shall be documented in the Engineering Notes field of The Total Equipment Data Base (TEDB).

This PC/M does not effect, amend, nor change the original Safety Evaluation, the Technical Specifications nor the Technical Design bases.

Engineering Package 177-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not adversely affect any equipment whose malfunction is postulated in the SAR to initiate an accident or prevent an accident from occurring. The modifications performed by this Engineering Package do not affect the ability of the components to perform as intended during normal, emergency and off-normal conditions under maximum differential pressures.

Therefore, the modifications described in this Engineering Package do not increase the probability of valve failure, and thus the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

This modification does not affect any structures, systems or components that function to deter the release of radioactivity or to provide post-accident shielding. The modifications performed by this Engineering Package do affect systems and components that are relied upon to mitigate accident consequences, and contain radioactive fluids. However, the modifications performed do not affect the operational characteristics of the valves and do not affect the equipments ability to function during an accident. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

System operability is not being affected by the modifications to the MOV's identified in this Engineering Package. Valve operability will not be affected by the prescribed modifications. Seismic Qualification of the motor operated valve has not been affected by the modification. In addition, no new failure modes are created as a result of this modification, as this modification serves only to provide a non-intrusive means of measuring performance.

Engineering Package 177-293

Unreviewed Safety Question (USQ) Determination (Cont.)

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

System operation is not affected by this modification. This modification does not adversely interact spatially or functionally with any structure, system or component important to safety. The modifications performed by this Engineering Package do not affect the ability of the valves and valve operators to perform their mitigating functions as intended during emergency and off normal conditions under maximum differential pressures. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. This modification provides design documentation of a permanent modification to plant components. No new failure modes or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR. Additionally, component design parameters and system interfaces remain unchanged based on this modification. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not adversely interact spatially or functionally with any structure, system or component important to safety. This modification does not alter the function or the design basis of any MOV. No new failure modes are created for the subject MOV's that can be postulated to cause a malfunction of equipment important to safety different than those previously analyzed in the SAR. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specification requirements and Technical Specification Bases are not affected by this modification. The design bases of the valves and valve operators remains unchanged. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 183-293

ABSTRACT

This modification is an alternative to the presently utilized procedural controls to prevent undesirable ICW and CCW pump start conditions when operating the "C" pump as a replacement for either the "A" or the "B" pump. The existing control switches and their associated HFA latching relays will be replaced with control switches having a "pull-to-lock" feature and slip contacts. These switches will perform the latching relay memory function by their slip contacts and will allow pump lockout (trip) in their "pull-to-lock" position. Presently, racking out the switch gear breakers or pulling their control power fuses is required to disable the pump starting ability.

This modification includes the safety related ICW and CCW pump control circuits, therefore, this EP is classified as Safety Related.

A safety evaluation of these modifications has been performed in accordance with 10CFR50.59. This evaluation concludes that implementation of this EP does not involve an unreviewed safety question nor does it require a change to the Technical Specifications and it has no adverse effect on plant safety. Therefore, prior NRC approval is not required.

Engineering Package 183-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not affect any equipment whose malfunction is postulated in the FSAR to initiate an accident or prevent an accident from occurring. The modifications performed by this Engineering Package enhance the ability of the pumps to perform as intended during emergency and off normal conditions. Therefore, the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

This modification does not affect any structures,systems nor components that function to detect the release of radioactivity or to provide post accident shielding. This modification does not affect the functional operation of the pumps, however, it does enhance the ability to respond to a Station Blackout event. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The operability of the pumps in this modification is being enhanced by providing operation from the Control Room of functions presently requiring local actions. Since the reliability of the pump control circuit is being improved by this proposed change, it will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The operability of these systems is being enhanced by this modification. Since the pumps will continue to perform and satisfy their design basis requirements, the proposed change will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

Engineering Package 183-293

Unreviewed Safety Question (USQ) Determination (Cont.)

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. Since this modification does not affect any equipment whose malfunction is postulated in the SAR to initiate an accident or prevent an accident from occurring, it will not create the possibility of an accident of a different type than any previously evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification improves the reliability of the system by eliminating several relays, including HFA relays. It also replaces the SBM control switches with SBM switches having a "pull-to-lock" feature which would have the same malfunction characteristics as the original switches. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

This modification does not affect the operational requirements for the cooling systems as defined-by the Technical Specifications Sections 3/4 7.3 and 3/4 7.4. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 212-293

ABSTRACT

During the July 1992 post-refueling startup of Unit 2, a steam generator level lead/lag compensation module in the feed water control system (FWCS) was found to be operating improperly. Since there were no spare modules in stock, operation of the plant was evaluated with the lead/lag circuit bypassed (i.e., "jumpered out"). The evaluation concluded that level lead/lag compensation (particularly with the existing setpoints) provides only a relatively minor role in limiting the maximum and minimum steam generator levels which would be experienced during the applicable design basis operational transients. The evaluation further concluded that the FWCS would provide sufficiently stable steam generator level control without level lead/lag compensation.

The malfunctioning lead/lag module was temporarily jumpered out and the ensuing operating cycle proceeded without incidents. Since the level response without lead/lag compensation was found by analysis to be acceptable for all of the design basis load maneuvers, the compensation modules (one in the FWCS of each steam generator) are being physically removed from the control loop.

The physical removal of these modules does not impact any safety related structure, system, or component. They are located within RTGB 202 and RTGB 203, respectively, adjacent to other non-safety related feed water control equipment. In keeping with this, the potential for interaction with safety related equipment is limited. However, the existing modules were seismically installed for defense in depth, consistent with the general design requirements at the time. In light of the seismic design, the modifications performed by this Engineering Package (EP) are classified as Quality Related.

In addition to removing the two level lead/lag modules, this EP rewires the level transmitter current loop so that the I/I isolators (LY-9011, LY-9021) are included in the test circuit associated with test jack J3. A non-essential resistor is also removed from the loop.

The modifications performed by this EP do not adversely affect plant safety or operation. The EP demonstrates that the modifications do not constitute an unreviewed safety question and that no changes to the technical specifications are required.

Engineering Package 212-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The changes proposed by this engineering package are confined to the level transmitter current loops of the feed water control system (FWCS). The failure of a component in the current loop can cause a reduction or loss of feed water to an operating steam generator. The loss of feed water to one steam generator, however, is bounded by the plant safety analyses. That is, a loss of feed water to both steam generators is assumed as part of the plant safety analyses, to conservatively evaluate the plant response to a series of events characterized by a decrease in secondary system heat removal.

The failure of a component in the current loop could also overfeed an operating steam generator and challenge secondary system integrity. The FWCS receives a non-safety related input signal from the protection channels to prevent an overfill condition from occurring. Since this protection feature is performed by a control grade system, it is not analyzed in the SAR. Nonetheless, the ability of the FWCS to protect secondary side integrity is not diminished by the proposed circuit modifications.

The probability of underfeeding or overfeeding an operating steam generator due to a malfunction of a FWCS component is not increased by the proposed changes because the lead/lag module and the surplus resistor are each contributors to these events. Removing these two non-essential components from the loop reduces the likelihood of either event occurring. Although removal of the lead/lag module slows the initial response of the system, the change is only slight and a large margin remains to reactor trip or high level override.

Since the functions of the accident mitigation features (i.e., the low level reactor trip, high level override, and the high level turbine trip) are not affected by the proposed changes, the probability of an accident is not increased.

Engineering Package 212-293

Unreviewed Safety Question (USQ) Determination (Cont.)

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The circuit modifications performed by this EP eliminates two nonessential components from the steam generator level transmitter current loops (LT-9011, LT-9021). These loops are part of the FWCS which maintains steam generator level during steady state and transient load conditions. Since no functional changes are made to the FWCS or any interfacing system, the plant response to an accident is not altered by this EP. The consequences of a circuit failure are bounded by the loss of feed water event analyzed in SAR.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The FWCS does not directly affect operability of any safety related equipment. The components, however, are seismically supported so that they do not indirectly affect operability of safety related equipment located in the surrounding area. The modifications performed by this EP do not diminish the capability of the FWCS components to withstand a seismic event. The common support bracket that wraps around the steam generator level and feed water flow lead/lag modules is retained in the modified design. Despite removal of the level lead/lag unit, it maintains a compressive load against the feed water module to offset the prying effects on the fasteners at the base of the module during a seismic event.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed changes do not alter the actions of the feed water system during an accident. Any equipment malfunction which causes a rapid increase in steam generator level would still be terminated by the high level override and turbine trip functions. Any malfunction which causes a low steam generator level would still be terminated by the reactor trip and AFW initiation circuits.

Interaction with other safety systems is precluded by maintaining a lateral brace around the feed water flow lead/lag module. The space vacated by removal of the neighboring steam generator level lead/lag module does not diminish the compressive load applied to the feed water flow module.

Since the FWCS does not directly, or indirectly, affect operation of any accident mitigation systems, or systems which contain radioactive fluids, the consequences of an accident are not increased by this EP.

Engineering Package 212-293

Unreviewed Safety Question (USQ) Determination (Cont.)

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The changes proposed by this engineering package are confined to the level transmitter current loops of the feed water control system (FWCS). The failure of a component in the current LOOP can cause a reduction in the supply of feed water to an operating steam generator or result in an overfill condition. The loss of feed water to one steam generator, however, is bounded by the plant safety analyses. That is, a loss of feed water to both steam generators is assumed as part of the plant safety analyses, to conservatively evaluate the plant response to a series of events characterized by a decrease in secondary system heat removal. Steam generator overfill protection is an equipment protection feature. The proposed modification does not adversely affect the high level override or turbine trip functions which are designed to mitigate steam generator overfill conditions.

Interaction with other safety systems is precluded by maintaining a lateral brace around the feed water flow lead/lag module. Since there is no functional or spatial interaction with other systems or components, no new accident scenarios are created.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The changes proposed by this engineering package are confined to the level transmitter current loops of the feed water control system (FWCS). The failure of a component in the current loop can cause a reduction in the supply of feed water to an operating steam generator or result in an overfill condition. Both of these malfunctions are addressed in the plant safety analysis. A reduction in feed water is bounded by the loss of feed water analysis. The potential effects of steam generator overfill are bounded by the main steam line break analysis.

Interaction with nearby safety-related equipment is precluded by maintaining a lateral brace around the feed water flow lead/lag module. Since there is no functional or spatial interaction with other systems or components, no new failure modes are created. As a result, no new equipment failures are postulated.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The circuit modifications performed by this EP eliminates two nonessential components from the steam generator level transmitter current loops (LT-9011, LT-9021). These loops are part of the FWCS which maintains steam generator level during steady state and transient load conditions. The level transmitters are not safety-related so their operation is not governed by plant technical specifications. They are not credited in the basis of any technical specification either since they are part of a "control grade" system. Based on the above, the margin of safety defined by the technical specifications is not changed by this EP.

Engineering Package 215-293

ABSTRACT

The purpose of this Engineering Package is to provide documentation to perform the maintenance practice of installing current ABB/CE designed steam generator tube plugs and short flexible tube stakes ("stakes").

The tube plugs will isolate degraded or leaking steam generator tubes to preclude or eliminate primary to secondary leakage.

Stakes will be installed in those Steam Generator (S/G) tubes exhibiting circumferential indications near the tube expansion transition region (i.e. at or near the top of the tubesheet). This issue has been identified at other Combustion Engineering designed nuclear power plants (Maine Yankee and Millstone Point 2) including St. Lucie Unit 1 in 1990. Circumferential indications are attributed to stress corrosion; this phenomenon is discussed in NRC Information Notice 90-49. In-Service Inspection (via eddy current testing such as rotating pancake coil ECT) will determine when circumferential indications exist and when installation of stakes are necessary. The stakes are designed to prevent tube to tube contact, should the indications progress and the tube sever.

The steam generator tube plugs form a part of the RCS Class 1 pressure boundary. Although stakes are not part of the primary or secondary pressure boundary, they indirectly protect active tubes of the S/Gs which perform a safety related function. Therefore, this modification is classified as Safety Related.

A safety evaluation was performed in accordance with 10 CFR 50.59. The evaluation concluded the implementation of this Engineering Package does not involve an unreviewed safety question, does not constitute a change to Technical Specifications, nor reduce the margin of safety of any Technical Specifications, and does not have an adverse effect on plant safety, security or operation. Therefore, prior NRC approval is not required for implementation of this EP. Implementation of this EP will have no adverse impact on plant safety or operation.

The existing plant drawings for the tube plugs are not the design currently being used by the vendor (ABB/Combustion Engineering). This EP updates plant documents (drawings and the instruction manual) to install stakes as a maintenance practice with the latest mechanical and welded tube plug design. The S/G instruction manuals will be revised to provide the drawings necessary to perform tube plugging and staking on an "as-needed" basis.

Engineering Package 215-293

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The stakes and/or plugs will be installed in tubes that will no longer be in service to remove heat from the RCS. The functions of these components are to protect the RCS pressure boundary, minimize primary to secondary leakage and prevent multiple tube damage. Analysis has demonstrated that the equipment can withstand the design loads it may be subjected to and not interact with active tubes in the S/G. Therefore, this modification does not increase the probability of occurrence of an accident previously analyzed in the SAR.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

Accidents previously evaluated include S/G tube rupture (FSAR Sections 15.6.2.1.4 and 15.6.5.3) and Main Steam Line Break (FSAR Section 15.1.6). The plugs are installed to isolate potentially leaking tubes. The S/G tube stakes will be installed in the plugged tubes (isolated from primary system) preventing tube to tube contact in the event the plugged tube severs. Because the components are designed not to interact with active tubes, this modification does not increase the consequences of an accident previously evaluated in the SAR.

C. Does the modification increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

Since S/G tube stakes are installed in inactive tubes, they do not affect active tubes or other components in the S/G. Analysis determined that should the staked tube degrade or sever, the S/G tube stake will be retained within both ends of the tube and not contact other S/G tubes or internals. Therefore, this modification does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

Engineering Package 215-293

Unreviewed Safety Question (USQ) Determination (Cont.)

D. Does the modification increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The S/G tube stakes are installed in tubes which will be plugged and isolated from active tubes and S/G internals. The design and installation of the plug and/or stake minimizes the probability of a S/G tube or RCS component failure. Failure of a S/G support component such as a stake is not considered credible. The stake will remain in the plugged tube and will not interact with other S/G internals. Since the modification minimizes the probability of a plugged tube from contacting other tubes, the proposed modification does not increase the consequences of a malfunction of equipment important to safety.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The stake is a S/G support component and is installed in a S/G tube that has a circumferential indication near the expansion transition region above the tubesheet. After staking, the tube is plugged. Since the stakes are isolated from the RCS and the failure of a support component is not considered credible, the stake by itself, does not create the possibility of an accident of a different type than any previously evaluated in the SAR. Additionally the stake is provided to minimize the possibility of a severed tube from damaging other tubes.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The stakes are isolated from the RCS by the tube plugs. It is designed to remain in place in a severed tube, not affect the wear of the staked tube and not increase the probability of failure of its associated S/G tube plug. Therefore, this modification does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any technical specification?

This modification does not reduce the margin of safety defined in the basis of any Technical Specification because the S/G tube stakes are installed in plugged tubes and therefore do not affect the plugging limit. It also does not affect the capability to detect imperfections and does not promote RCS leakage. Therefore, the proposed modification does not reduce the margin of safety as defined in the basis for any technical specification.

Engineering Package 049-292

ABSTRACT

NRC Generic Letter 89-10 requires that operating nuclear plants develop and implement a program to ensure that control switch settings on all safety-related motor-operated valves (MOV's) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. To comply with the requirements of the Generic Letter, torque switches in the opening and closing direction are set through the use of diagnostic equipment monitoring valve and actuator performance. The Liberty Technology Center, Inc. specifically developed the Valve Operation Test and Evaluation System (VOTES) to provide a simple-to-use and accurate method of testing and diagnosing the operation of motor operated valves. One principal feature of the system is the force measurement concept. Since the valve stem forces cause equal and opposite yoke reaction forces, measurement of yoke deflection is an accurate indicator of stem force over the entire valve stroke. VOTES diagnostic equipment to be used to evaluate MOV's senses the reactionary force produced in the yoke via a force sensor (i.e. strain gauge). This Engineering Package provides the details necessary to mount the force sensors on the yokes of motor operated valves within the Generic Letter Program.

Because the motor-operated valves associated with Generic Letter 89-10 are safety-related, or may affect safety-related systems, this engineering package has been classified as Safety Related. These changes have no adverse effect on plant operation or safety. A review of the proposed modification to be implemented by the PC/M was performed against the requirements of 10CFR50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 049-292

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not adversely affect any equipment whose malfunction is postulated in the SAR to initiate an accident or prevent an accident from occurring. The modifications performed by this Engineering Package do not affect the ability of the components to perform as intended during normal, emergency and off-normal conditions under maximum differential pressures.

Therefore, the modifications described in this Engineering Package do not increase the probability of valve failure, and thus the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

This modification does not affect any structures, systems or components that function to deter the release of radioactivity or to provide post-accident shielding. The modifications performed by this Engineering Package do affect systems and components that are relied upon to mitigate accident consequences, and contain radioactive fluids. However, the modifications performed do not affect the operational characteristics of the valves and do not affect the equipments ability to function during an accident. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

System operability is not being affected by the modifications to the MOV's identified in this Engineering Package. Valve operability will not be affected by the prescribed modifications. Seismic Qualification of the motor operated valve has not been affected by the modification. In addition, no new failure modes are created as a result of this modification, as this modification serves only to provide a non-intrusive means of measuring performance.

Engineering Package 049-292

Unreviewed Safety Question (USQ) Determination (Cont.)

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

System operation is not affected by this modification. This modification does not adversely interact spatially or functionally with any structure, system or component important to safety. The modifications performed by this Engineering Package do not affect the ability of the valves and valve operators to perform their mitigating functions as intended during emergency and off normal conditions under maximum differential pressures. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. This modification provides design documentation of a permanent modification to plant components. No new failure modes or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR. Additionally, component design parameters and system interfaces remain unchanged based on this modification. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not adversely interact spatially or functionally with any structure, system or component important to safety. This modification does not alter the function or the design basis of any MOV. No new failure modes are created for the subject MOV's that can be postulated to cause a malfunction of equipment important to safety different than those previously analyzed in the SAR. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specification requirements and Technical Specification Bases are not affected by this modification. The design bases of the valves and valve operators remains unchanged. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 053-292

ABSTRACT

The existing St. Lucie Unit No. 2 turbine generator relay protection scheme does not preclude certain inadvertent non-synchronized connections to the power system. Such events could result in extensive damage to the main generator and/or turbine.

This Engineering Package (EP) encompasses the engineering/design details for the installation of new relaying and control equipment for the St. Lucie Unit No. 2 main Generator as follows:

1. Protection against inadvertent non-synchronized connection to the power system.

2. Revision of the tripping logic of the under-frequency relays.

3. Addition of a synchrocheck relay to supervise closing of the generator breakers.

4. All interconnecting cabling and raceway for the above equipment, as required.

The relays/systems affected by this EP perform no Nuclear Safety Related function. However, since it involves modifications to the RTGB 201 which contains Nuclear Safety Related equipment, this EP has been classified Quality Related.

The safety evaluation of this EP has shown that the implementation of this PCM does not constitute an unreviewed safety question as defined in 10CFR50.59 and does not require a change in the Plant Technical Specifications. This PCM has no adverse impact on plant safety or operation; thus this PCM can be implemented without prior NRC approval.

Engineering Package 053-292

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not increase the probability of occurrence of an accident previously evaluated in the safety analysis report since the inadvertent energization relaying is manually isolated (key switch) with the plant in modes 1 and 2. No accidents evaluated in the SAR involve any equipment/systems modified by this PC/M. The relaying modifications enhance main generator protection with no impact on existing accident analyses.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the Safety Analysis Report?

The equipment modified or added by this PC/M will not prevent safety-related equipment from performing their intended functions. As generator protection has no bearing on any accidents previously analyzed in the SAR, the implementation of these modifications cannot increase the consequences of an accident previously evaluated.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report?

The addition or modification of equipment by this PC/M will not prevent safety-related equipment from performing their intended functions. As mentioned above, the implementation of these modifications cannot increase the probability of occurrence of a malfunction of equipment previously evaluated in the SAR since generator protection is not addressed therein.

D. Does the proposed change increase the consequences of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report?

Modifications performed by this PC/M will not prevent safety-related equipment from performing their intended functions. Since generator protection instrumentation is non-safety related and has no effect on Nuclear Safety Related equipment, the implementation of these modifications cannot increase the consequences of a malfunction of equipment previously evaluated in the SAR.

Engineering Package 053-292

Unreviewed Safety Question (USQ) Determination (Cont.)

E. Does the proposed activity increase the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report?

The equipment added/modified by this EP is not required during an accident condition nor will it prevent safety related equipment from performing their functions. This modification does not affect any safety related equipment. A failure can only cause turbine trip (modes 1 and 2), which is analyzed in SAR Section 15.2, or a loss of Off-Site Power, which is analyzed in SAR Section 15.3. Therefore. the possibility of an accident of a different type than any evaluated previously in the SAR is not created.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The possibility of a malfunction of equipment of a different type than evaluated previously is not created. The equipment added/ modified by this EP does not interface with any Safety Related equipment and cannot prevent nuclear safety related equipment from performing their design basis functions. This modification does not affect any safety related equipment.

G. Does the proposed activity reduce the margin of safety as defined in the bases for any Technical Specification?

The margin of safety as defined in the basis for any Technical Specification is not reduced by this modification since the equipment added/modified by this EP does not form the basis of any Technical Specification. Also, no Plant Technical Specification system availability or surveillance requirement is affected by the implementation of this PC/M.

Engineering Package 087-292

ABSTRACT

This Engineering Package covers modifications to the control circuits of the 2A, 2B and 2C CVCS Charging pumps, and the 2A and 2B Boric Acid Makeup pumps (Bumps). Specifically, this PC/M eliminates the five (5) minute time delay relays for these pumps such that they are loaded onto the Emergency Diesel Generator (EDG) zero load block. Each pump control circuit has a five minute time delay relay which loads them on the EDG, following a Loss of Offsite Power (LOOP) event. Eliminating the time delay enables these pumps to be manually started upon EDG breaker closure following a LOOP, if the pumps were not running; or, the pumps will automatically start if they were running prior to the LOOP.

This EP is being prepared as requested by the St. Lucie Operations department. It will enhance their ability to comply with the Emergency Operating Procedures regarding emergency boration, as directed in the Standard Post Trip Actions. This modification would also make the St. Lucie plants more similar, since these pumps on Unit l (Except for Charging Pump lC) are already loaded on the EDG zero time load block.

The actual modification requires the following:

1) Eliminate the breaker trip on under-voltage and breaker close permissive. Eliminate the 5 minute time delay relay for load sequencing on the Emergency Diesel Generator following a Loss of Off-site Power (LOOP), for Charging pumps 2A, 2B and 2C.

2) Elimination of the 5 minute time delay relay for the Boric Acid Makeup pumps 2A and 2B, and deletion of the manual and automatic reset following a LOOP or LOOP/SIAS condition.

Note that automatic actuation of these pumps (i.e., following a SIAS) is not adversely affected.

Post modification testing will be required to verify that the sequencing time of the pumps has been properly implemented, that the control logic with respect to Safety Injection Actuation Signal and LOOP is functioning properly, and that other automatic functions have not been adversely affected.

Because the charging pumps, BAMPS and EDG's are required to perform a safety function for the plant to achieve safe shutdown or mitigate the consequences of an accident, the safety classification associated with this modification is Safety Related. A review of the modification to be implemented by this PC/M was performed against the requirements of 10CFR50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 087-292

ABSTRACT (Cont.)

SUPPLEMENT 1:

Supplement 1 to this Engineering Package was issued to restore the control circuits of the 2A, 2B and 2C CVCS Charging Pumps and the 2A and 2B Boric Acid Make-up Pumps to their original configuration, i.e. a five minute time delay. This supplement was intended to restore the circuitry to its original condition until such time that the 0.1 second time delay could be successfully implemented.

The modification presented by this supplement returned the control circuits to their original design configuration. Therefore, this configuration has been previously reviewed against the requirements of 10CFR50.59 and it was concluded that this modification does not constitute an unreviewed safety question and does not require a change to plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

SUPPLEMENT 2:

Supplement 2 to this Engineering Package is being issued to modify the control circuits of the 2A, 2B and 2C CVCS Charging Pumps and the 2A and 2B Boric Acid Make-up Pumps to place these pumps on the EDG zero load block. Specifically, this supplement provides the following:

1) The modifications to the Charging Pump 2A, 2B and 2C are the same.

Charging Pumps circuitry modifications consist of the elimination of the bus stripping signal to the breaker trip and breaker close permissive and the deletion of the 5 minute time delay relay (sequencer relay) and the elimination of the manual reset following a Loss Of Offsite Power (LOOP). The modifications require lifting wiring associated with the breaker trip and close functions and the installation of jumper wiring to eliminate the timing permissive in the closing circuit.

The control circuitry modification will be implemented in conjunction with a modification to Operating Procedure OP 2-0210020 Rev 22, to require that during Normal Plant Operation one charging pump will be running with the switch in START, a second charging pump in AUTO and the third charging pump in STOP. The Charging Selector switch will be set to the START and AUTO pumps. Under abnormal circumstances, two charging pumps may be in START with the third charging pump in STOP and the Charging Selector is selected to the START selected pumps.

The modification to the Charging Pumps control circuitry allows the Charging pump selected in START to automatically restart following a LOOP, and the Charging pump selected in AUTO (standby) to automatically restart following a LOOP, if the pump was running before the LOOP.

2) The modifications to the Boric Acid Makeup Pumps (BAMPS) 2A and 2B consists of the elimination of the 5 minute time delay relay (sequencer relay) and the elimination of the manual and automatic reset following a Loss Of Offsite Power or LOOP/LOCA.

The modification to the BAMPS 2A and 2B control circuitry allows these pumps to automatically restart following a LOOP, if the pump was running before the LOOP.

Engineering Package 087-292

ABSTRACT (Cont.)

Prior to this modification operator action was required to manually reset the circuit and restart the BAMPs following a LOOP. The control circuitry modification also eliminates the Automatic Reset for LOOP/SIAS.

This configuration was reviewed against the requirements of 10CFR50.59 and it was concluded that this modification does not constitute an unreviewed safety question and does not require a change to plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 087-292

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not affect any equipment whose malfunction is postulated in the SAR to initiate an accident or prevent an accident from occurring. The modifications performed by this Engineering Package enhance the ability of the components to perform as intended during emergency and off-normal conditions. Eliminating the five (5) minute time delay and allowing automatic start of the running pumps after LOOP and automatic start of the pumps in START and AUTO after LOOP/SIAS will enhance the ability of the CVCS to respond to off-normal and emergency conditions. There are no new failure modes associated with this modification, therefore the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

B. Does the proposed change increase the consequences of an accident previously evaluated in the SAR?

This modification does not affect any structures, systems or components that function to detect the release of radioactivity or to provide post-accident shielding. The modifications performed by this Engineering Package do affect systems and components that are relied upon to mitigate accident consequences. However the modification performed improve the operational characteristics of the Chemical and Volume Control System by ensuring that borated water is delivered to the RCS with a minimal time delay. Several transients in FSAR Chapter 6 (Engineered Safety Features) and Chapter 15 (Accident Analysis) were evaluated with respect to the reduced response time for CVCS charging pump flow for events involving a LOOP. It was determined that reducing the charging flow response time from its existing Technical Specification value of 330 seconds to the new value, which will range from zero seconds to approximately 180 seconds, will not adversely affect the conclusions of the Safety Analyses.

In addition, the modification will allow the automatic start of the BAM and charging pumps that were running prior to a LOOP and will allow the operator to emergency borate by manually starting the BAM and charging pumps in AUTO upon EDG breaker closure following a LOOP. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

Engineering Package 087-292

Unreviewed Safety Question (USQ) Determination (Cont.)

C. Does the proposed change increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The operability of the CVCS is being enhanced by this modification, by ensuring that borated water can be delivered to the RCS with minimal time delay. The systems affected by this modification are the CVCS and the Emergency Diesel Generator. These systems have been reviewed and it has been determined that there are no new failure modes created by this modification. The loading profile of the EDG has changed, however, it has been concluded that the new load profile is acceptable. Since the CVCS and EDG will continue to perform and satisfy their design basis requirements, the proposed change will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed change increase the consequences of a malfunction of equipment important to Safety previously evaluated in the SAR?

The operability of the CVCS is being enhanced by this modification, by ensuring that borated water can be delivered to the RCS with minimal time delay. The systems affected by this modification are the CVCS and the Emergency Diesel Generator. These systems have been reviewed and it has been determined that there are no new failure modes created by this modification. The loading profile of the EDG has changed, however, it has been concluded that the new load profile is acceptable. Therefore, the ability of the CVCS to mitigate the consequences of an accident has been enhanced. The consequences of a malfunction of equipment important to safety has not been altered by this modification. Since the CVCS and EDG will continue to perform and satisfy their design basis requirements, the proposed change will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

E. Does the proposed changes create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. The modifications provide for a change to the control logic of the Charging and Boric Acid Makeup pumps. The modification to the control logic enhances the operational characteristics of the CVCS. The effect on the Emergency Power system has been reviewed. The EDG's have sufficient margin to handle the loading of the pumps in the new load block designated by this modification. No new failure modes, or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR as a result of this modification. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification.

Engineering Package 087-292

Unreviewed Safety Question (USQ) Determination (Cont.)

F. Does the proposed change create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not interact spaciously or functionally with any structure, system or component important to safety other than the CVCS and EDG's. No new failure modes are created for the subject systems or equipment that can be postulated to cause a malfunction of equipment important to safety different than those previously analyzed in the SAR. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

G. Does the proposed change reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specifications requirements and Technical Specification Bases are not affected by this modification. The modifications provided by this EP affect systems and components required by the Technical Specifications. Table 3.3-5 of the Technical Specifications "Engineered Safety Features Response Times" lists the response time for charging flow to be less than or equal to 330 seconds, including EDG starting and sequence loading delays, or less than or equal to 180 seconds, not including EDG starting and sequence loading delays. The footnote to this table indicates that response time includes the movement of valves and the attainment of pump discharge pressure.

This modification deletes the (5) minutes time delay relays from the charging pumps and deletes the charging pumps breaker trip and closing permissive on under voltage, thus placing the charging pumps on the EDG zero load block. These modifications enable the running charging pump to automatically start upon EDG breaker closure following a LOOP and the charging pump in AUTO to re-start if it was running before the LOOP. Following a LOOP with SIAS the Charging pumps in START and AUTO will start automatically upon EDG breaker closure. The above modifications enhance the capability of the charging system to satisfy the 330 seconds requirement; reducing the time delay will have no effect on satisfying the requirement of 180 seconds.. Therefore, the existing requirement for charging flow stated in the Engineered Safety Features Response times remains valid.

This modification does not affect the operational requirements for the charging pumps or BAMPS as defined by the Technical Specifications. The Technical Specifications for the Boric Acid makeup pumps only defines when the pumps must be operational. There are no limits associated with time delays. The boration flow paths are defined in the Technical Specifications Sections 3/4.1.2, 3/4.3.2, 3/4.5.2. The boration flow paths ensure that negative reactivity control is available for normal or emergency conditions during each mode of facility operation. The requirements for a boration flow path are defined in the Technical Specifications and the FSAR chapters 6 and 15 for Loss of Coolant Accident events (LOCA) and non-LOCA events. These safety functions are not adversely affected by this modification

Engineering Package 087-292

Unreviewed Safety Question (USQ) Determination (Cont.)

This modification does not affect the operational requirements for the EDG's as defined by the Technical Specifications Sections 3/4.8.1, and 3/4.8.3. The EDG's provide the emergency on-site power for off-normal and emergency conditions. The requirements for EDG performance are defined in the Technical Specifications and FSAR Chapter 8. Calculations have been performed to determine that the EDG safety functions are not adversely affected by this modification.

Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 163-292

ABSTRACT

The purpose of this Engineering Package (EP) is to provide a means of testing the turbine trip solenoid valves (20-1/OPC, 20-2/OPC and 20/ET) with the plant on line. This will be accomplished by the installation of a test block under each of the turbine trip solenoid valves which provides an inlet and outlet isolation valve for each solenoid valve. This test block will allow for periodic exercising, testing, and replacement of the these solenoid valves with the plant on line. Each SOV can be isolated individually from the emergency trip header. A local test station with key lock test switches will be provided and testing will be performed locally at the turbine front standard.

The isolation valves will be provided with locking devices to ensure the isolation valves are locked open during normal operation. The testing will be performed under strict administrative control to prevent any interference with the turbine trip function. The turbine trip function is not a safety related function but due to the potential for interference with the turbine trip function and the resulting challenges to safety systems, this EP has been classified as Quality Related. The safety evaluation shows that this PCM does not constitute an unreviewed safety question nor require a change to the Technical Specifications. Therefore, prior approval by the NRC is not required.

Revision 1

This PC/M is being revised to modify the SOV test block to the configuration used on the St. Lucie Unit 1, to add an EH test header, to provide unique sets of locks and keys for the test block inlet and outlet isolation valves, and to replace the existing obsolete Parker Hanifin SOVs with direct replacement Sterling Hydraulic SOVs. A brief description of each of these modifications is provided below.

1. The test block was modified to reflect the design used on St. Lucie Unit 1, i.e., remove the pressure gauge and orifice from the test block and relocate them to the EH test header at each SOV. The existing pressure gauge port and orifice flow path will be plugged.

2. A separate EH Test Header has been added to supply an alternate source of EH fluid at normal operating pressure to each SOV. The EH test header allows the SOV test to be accomplished while the SOV is completely isolated from the emergency trip header. The alternate EH test header will minimized the potential for pressure perturbations within the turbine trip header during and after SOV testing.

3. The replacement SOV is supplied by Westinghouse under part number 807J949002. The existing, obsolete SOV design was supplied by Westinghouse under part number 877A848001. Although there are minor differences in size, weight, and configuration, the new valve is being provided as a direct replacement which will fit in the same location and will meet the same performance requirements as the existing valve.

Engineering Package 163-292

ABSTRACT (Cont.)

4. A separate unique set of locks and keys will be provided for each test block inlet and outlet isolation valves. Unique locks and keys will minimize the potential of human error while performing the testing.

Engineering Package 163-292

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

SAR Section 15.2.1.2, Limiting Reactor Coolant System Pressure Event-Isolation of Turbine, evaluates plant response to an isolation of turbine at 102% power, and SAR Section 15.2.2.1, Limiting Offsite Dose Event Isolation of Turbine With a Stuck Open Main Steam Safety Valve, evaluates response to isolation of turbine at 20% power.

Although the proposed changes result in a slight increase in the probability of an initiating event, turbine trip due to circuit failure, the changes also provide compensating effects through the addition of turbine control system test and monitoring functions which enhance the capability to assess the operability of a protective system prior to the system being required to operate, and through modification of control operation to provide assurance that a manual turbine trip will be maintained. The proposed changes therefore do not increase the probability of occurrence of an accident previously evaluated as defined in 10CFR50.59.

B. Does the proposed change increase the consequences of an accident previously evaluated in the SAR?

This modification does not affect any structures, systems or components that function to deter the release of radioactivity or to provide post-accident shielding. The modifications performed by this Engineering Package only effect the turbine trip function testing and does not affect components that are relied upon to mitigate accident consequences, and contain radioactive fluids. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

Engineering Package 163-292

Unreviewed Safety Question (USQ) Determination (Cont.)

C. Does the proposed change increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

System operability is not being affected by the modifications as identified in this Engineering Package. Turbine trip reliability will be enhanced by the prescribed modifications. In addition, no new failure modes are created as a result of this modification, as this modification serves to provide additional testing capabilities.

Addition of SOV test blocks, EH test header, and associated SOV test switches, serve only to add the capability to test the turbine trip solenoids which will enhance the reliability of the turbine trip function. As such, no new accident initiating events are created. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR has not increased by this modification.

D. Does the proposed change increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

System operation is not affected by this modification. This modification does not interact spatially or functionally with any Safety Related structure, system or component. The modifications performed by this Engineering Package only effect the turbine trip function testing and does not affect components that are relied upon to mitigate accident consequences, and contain radioactive fluids. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification.

E. Does the proposed change create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. This modification provides increased test capabilities for the turbine trip function. No new failure modes or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification.

F. Does the proposed change create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not interact spatially or functionally with any structure, system or component important to safety other than the trip solenoid valves. This modification does not alter the function or the design basis of the turbine trip function or turbine trip solenoid valves. No new failure modes are created that can be postulated to cause a

Engineering Package 163-292

Unreviewed Safety Question (USQ) Determination (Cont.)

malfunction of equipment important to safety different than those previously analyzed in the SAR. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

G. Does the proposed change reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specification requirements and Technical Specification Bases are not affected by this modification. The design bases of the turbine trip function remains unchanged. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

The modifications performed by this EP enhance the ability to test the turbine trip function and therefore increases the reliability of the trip function.

Engineering Package 268-292

ABSTRACT

This PC/M is for the redesign of the aluminum bronze/monel lube water system piping from the Intake Cooling Water (ICW) System header to the ICW Pump seals and bearings and Circulating Water (CW) pump seals and bearings. This task must be performed after the completion of the ICW Self-Lubrication Modifications, PC/M 043-287 for the 2A pump and PC/M 178-290 for the 2B and 2C pumps. PC/M 117-292M, which installed isolation valve 21-SH21500, has not yet been closed. Since the existing ICW lube water piping provides lube water to the four CW pumps, an alternate means of supplying lube water to the CW Pumps will be required while this modification is being implemented. The proposed solution of tying in the 3" lube water piping from the non-essential header of the ICW system to the presently installed lube water header of the CW Pumps will be used. The proposed plan is to isolate one train at a time and supply lube water to the CW pump seals and bearings via the other train. Even so, a backup source of lube water will be made available during the installation of this modification. The new piping added to the system will be green thread, fiberglass piping.

In addition to the above, the following items are included in this Engineering Package (EP):

1. The ICW Lube Water System piping will be downgraded to Non Safety Related.
2. The CW Lube Water System will be supplied by the ICW non essential header.
3. All ICW Lube Water piping and pipe supports will be removed or abandoned in place.
4. The Safety Related SIAS signal and the wiring to the motor operators to MOVs I-MV-21-4A and 4B will be removed even though the valves will remain in place.
5. Any spared equipment that can not be removed while the unit is operating will be completed during a future outage under PC/M 287-292M.
6. All instrumentation associated with this PC/M will be removed and all annunciators will be spared out.

This PC/M has been classified as Safety Related because it is removing equipment that is Safety Related even though this equipment no longer serves a Safety Related function. After this PC/M has been completed the systems and equipment involved with this package, that is, the CW lube water system and the piping supplying lube water from the ICW system will be non-safety related. A safety evaluation of these modifications has been performed in accordance with 10 CFR 50.59. This evaluation concludes that implementation does not involve an unreviewed safety question nor a change to Technical Specifications. Additionally, it has no adverse effect on plant safety or operation. Therefore, prior NRC approval is not required.

Engineering Package 268-292

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The proposed activity does not increase the probability of occurrence of an accident because the modifications to the ICW and CW Lube Water Systems do not affect any accident that was previously evaluated in the SAR.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The proposed activity does not increase the consequences of an accident because the modifications to the ICW and CW Lube Water Systems have no affect on the performance of the systems and thus have no affect on mitigating the consequences of an accident.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety because the modifications to the ICW and CW Lube Water Systems are designed to ensure that safety related equipment is not adversely impacted. The modifications were evaluated for seismic excitations as required to assure continued functioning of safety related equipment.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed activity does not increase the consequences of a malfunction of equipment important to safety because the modifications to the ICW and CW Lube Water Systems are designed to ensure that safety related equipment is not adversely impacted.

Engineering Package 268-292

Unreviewed Safety Question (USQ) Determination (Cont.)

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The proposed activity does not create the possibility of an accident of a different type than any previously evaluated because the modifications to the ICW and CW Lube Water Systems do not involve any accident initiating scenarios. The ICW Lube Water System has been deleted because all ICW pumps are now self lubricated. The lube water supply to the CW Pumps has been moved to the non-essential header of the ICW System and it is protected by an isolation valve should the non safety, non-essential ICW header require isolation in an emergency.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The proposed activity does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated because the modifications to the ICW and CW Lube Water Systems are designed to ensure that safety related equipment is not adversely impacted.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The proposed activity does not reduce the margin of safety as defined in the bases for any Technical Specification because the modifications to the ICW and CW Lube Water Systems do not affect any safety margins as discussed in the bases of any Technical Specification. No margin of safety is affected by this modification.

Engineering Package 298-292

ABSTRACT

The 125 Vdc batteries 2A & 2B are part of the Class lE subsection and the 125 Vdc batteries 2C & 2D are part of the Non-Class lE subsection of the 125 Vdc distribution system at St. Lucie Unit 2. Each battery consists of sixty (60) cells to provide the required nominal output voltage of 125V. Vendor maintenance procedures for these 125 Vdc batteries require that the difference between individual cell voltage and average cell voltage be maintained within certain limits. Due to small variations in intercell resistance values, individual cells were not charged equally.

The Individual Cell Equalizers (ICE) were installed across the individual battery cell terminals of the 2D non-nuclear safety battery on a trial basis. This installation has improved the individual battery cell performance by minimizing the individual cell voltage excursions.

Engineering Package 298-292

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

Installation of Individual Cell Equalizers (ICE) devices on the Class IE safety-related batteries (2A & 2B) does not affect the initiation of an accident evaluated in the SAR nor increase the probability of occurrence. The operability of the Class IE safety-related batteries (2A & 2B) is maintained even with the ICE devices coupled directly across each battery cell. Therefore, the probability of an accident previously evaluated in the SAR would not be increased.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The consequences of an accident previously evaluated in the SAR are not increased by the installation of ICE devices on the Class IE safety-related batteries (2A & 2B). Installation of the ICE devices will not change, degrade or prevent system functions described in, or assumed to occur in the mitigation of any SAR accident. This proposed activity has no impact on the LOOP/LOCA analysis and the radiological consequences of an accident evaluated in the SAR will not be increased. Therefore, the consequences of an accident previously evaluated in the SAR would not be increased.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR has not been increased because the proposed activity will not result in new performance requirements being imposed on any system or components such that any design criteria will be exceeded. The Class IE safety-related batteries (2A & 2B) functional requirements are unchanged, therefore, no new probability of a malfunction has been imposed.

Thus, the probability of a malfunction of equipment important to safety previously evaluated in the FSAR would not be increased.

Engineering Package 298-292

Unreviewed Safety Question (USQ) Determination (Cont.)

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

Installation of the ICE devices does not change, degrade, or prevent actions described in, or assumed to occur in the mitigation of any SAR accident. Therefore, ICE device installation will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The proposed activity to install ICE devices on Class lE safety related batteries (2A & 2B) at St. Lucie Unit 2 does not introduce failure modes of a different type than any previously analyzed in the SAR. The system configuration and the design basis of the 125 Vdc system has not been changed or affected.

Therefore, the proposed activity does not create the possibility than an accident may be created that is different from any already evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

The installation of ICE devices on the Class lE safety-related batteries (2A & 2B) has been evaluated and does not impact the structural integrity or performance capability of the 125 Vdc system. The possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR is not increased by this proposed activity.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The installation of the ICE devices does not reduce the margin of safety for any PSL-2 Technical Specification.

Engineering Package 510-291

ABSTRACT

NRC Generic Letter 89-10 requires that operating nuclear plants develop and implement a program to ensure that switch settings on all safety related motor-operated valves (MOV's) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. Item a) of the Generic Letter requires tnat the design basis for these MOV's be reviewed to determine the maximum differential pressure expected during both opening and closing strokes for all postulated events. This has been completed and documented in FPL Calculation PSL-2FJM-91-046, Revision 1, and FPL Engineering Evaluation JPN-PSL-SEMP-91-048, Revision 0.

Item b) of Generic Letter 89-10 requires that the licensee establish the correct MOV switch settings based on the previously determined maximum differential pressure. All switches, including torque switches, torque bypass switches, position limit, position indication, overloads, etc., shall be considered. This design package provides the overall switch setting guidelines for fifty-eight (58) motor operated valves, in addition to specific design information, as determined by calculation, necessary to replace actuator spring packs and set both the open and close torque switches to meet the requirements of Generic Letter 89-10 for the valves identified herein.

Because the motor-operated valves associated with Generic Letter 89-10 are safety-related, or may affect safety-related systems, this engineering package has been classified as Safety Related. A review of the switch setting changes to be implemented by the PC/M was performed against the requirements of 10CFR50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval for the implementation of this PC/M is not required.

Supplement 1 of this Engineering Package is issued to reflect revised thrust values for numerous valves. The revised thrust values resulted from detailed system reviews performed for the valves, or from new or revised vendor information. Two (2) valves were added to the scope of the modification increasing the total to sixty (60) valves. In addition, the methodology for the open torque switch is revised due to a change in the diagnostic equipment which has been selected for use. This supplement revises the safety evaluation to reflect the revised design information. However, the original conclusions of the safety evaluation, that the change does not constitute an unreviewed safety question nor require a change to the plant Technical Specifications remains unchanged. Therefore, prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 510-291

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

This modification does not affect any equipment whose malfunction is postulated in the SAR to initiate an accident or prevent an accident from occurring. The modifications performed by this Engineering Package enhance the ability of the components to perform as intended during emergency and off-normal conditions under maximum differential pressures.

Replacement of the actuator spring packs and revising the thrust or torque values for the MOV operators only serve to enhance the operational characteristics of the MOV's. As such, no new accident initiating events are created. Therefore, the modifications described in this Engineering Package do not increase the probability of valve failure, and thus the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

B. Does the proposed change increase the consequences of an accident previously evaluated in the SAR?

This modification does not affect any structures, systems or components that function to deter the release of radioactivity or to provide post-accident shielding. The modifications performed by this Engineering Package do affect systems and components that are relied upon to mitigate accident consequences, and contain radioactive fluids. However the modification performed improves the operational characteristics of the valves and improves the equipments ability to function during an accident. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

Engineering Package 510-291

Unreviewed Safety Question (USQ) Determination (Cont.)

C. Does the proposed change increase the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

System operability is not being affected by the modifications to the MOV's identified in this Engineering Package. Valve operability will be enhanced by the prescribed modifications. In addition, no new failure modes are created as a result of this modification, as this modification serves to provide additional design documentation, or replace existing parts.

Replacement of the spring packs and revising the thrust or torque values for the MOV operators only serve to enhance the operational characteristics of the MOV's. As such, no new accident initiating events are created. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR has not increased by this modification.

D. Does the proposed change create malfunction of equipment in evaluated in the SAR?

System operation is not affected by this modification. This modification does not interact spatially or functionally with any structure, system or component important to safety other than the valves and valve operators themselves. Although actuator and valve loadings may increase, the revised loads are within the published ratings for the components. Replacement components have been selected in accordance with the same design criteria as the original components. The modifications performed by this Engineering Package enhance the ability of the valves and valve operators to perform as intended during emergency and off normal conditions under maximum differential pressures. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

This modification does not change the function or design bases of any structure, system or component important to safety as described in the SAR. This modification provides increased design documentation, makes adjustments to components within their published operating range or makes replacements of equivalent parts. No new failure modes or conditions are created that can be postulated to cause an accident different than those previously analyzed in the SAR. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created by this modification.

Engineering Package 510-291

Unreviewed Safety Question (USQ) Determination (Cont.)

F. Does the proposed change create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

This modification does not interact spatially or functionally with any structure, system or component important to safety other than the valves and valve operators themselves. This modification does not alter the function or the design basis of any MOV. No new failure modes are created for the subject MOV's that can be postulated to cause a malfunction of equipment important to safety different than those previously analyzed in the SAR. Therefore, the possibility of a malfunction of equipment important to safety which is of a different type than any previously evaluated in the SAR is not created by this modification.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The Technical Specification requirements and Technical Specification Bases are not affected by this modification. The design bases of the valves and valve operators remains unchanged. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced by this modification.

Engineering Package 120-290

ABSTRACT

This Engineering Package (EP) provides details for the replacement of the 5A High Pressure Feedwater Heater (HPFWH) and some of its associated valving.

After plugging an excessive number of tubes, a tube from the St. Lucie Unit 2 HPFWH was metallurgically examined to characterize Eddy Current Test results and determine the degradation mechanism. The examination showed evidence of chloride induced stress-corrosion cracking (SCC), a corrosion mechanism in which a combination of a susceptible alloy, sustained tensile stress and a corrosive environment lead to cracking of a metal. Further analysis verified the tube material to be type 304 stainless steel, which was as specified in the design specification.

A detailed review of the system operating history was performed and showed that normal chemistry values were exceeded in only a few instances, and rapid plant response resulted in short transient durations. Furthermore, only 5A HPFWH has a history of tube failures and these tubes were plugged after only one month of commercial operation. In conclusion, it is not possible to determine if the chlorides were introduced during fabrication or during MSR tube replacement (or some other maintenance activity). However, they are probably not the result of normal operations.

The tube material selected for the replacement heater is type 316L stainless steel which is more resistant to SCC than type 304 (or 304L) due to the addition of molybdenum, and other elements, to its chemistry.

This EP involves the non-nuclear safety related, non seismic portion of the feed water system as identified in the Unit 2 FSAR, section 10.4.7, and is therefore classified as Non-Nuclear Safety Related.

The safety evaluation of this package has shown that the implementation of this PCM does not constitute an Unreviewed Safety Question and requires no revision to the Unit 2 Technical Specifications. Prior NRC approval is not required for implementation. This PCM has no adverse impact on plant safety and operation, or the Plant Technical Specifications.

Engineering Package 120-290

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of Safety as defined in the bases for any technical specification is reduced.

This Engineering Package will implement the replacement of the 5A HPFWH and 25 globe valves. The Feedwater system from the condenser hotwell to the Feedwater isolation valves is non-nuclear safety related and not required for safe shutdown of the plant (SAR, Sec. 10.4.7).

This Engineering Package (EP) provides details for the replacement of the 5A, High Pressure Feedwater Heater (HPFWH) and some of its associated valving.

After many tubes required plugging, a tube from the St. Lucie Unit 2 HPFWH was metallurgically examined to characterize Eddy Current Test results and determine the degradation mechanism. The examination showed evidence of chloride induced stress-corrosion cracking (SCC), a corrosion mechanism in which a combination of a susceptible alloy, sustained tensile stress and a corrosive environment lead to cracking of a metal. Further analysis verified the tube material to be type 304 stainless steel, which was as specified in the design specification. A review of the system operating history showed that normal average chemistry values were exceeded three times since 1985 due to condenser tube leaks.

The tube material selected for the replacement heater is type 316L stainless steel which is more resistant to SCC than type 304 (or 304L) due to the addition of molybdenum, and other elements, to its chemistry.

This EP will be performed on the non-nuclear safety related, non seismic portion of the feedwater system as identified in the Unit 2 SAR, subsection 10.4.7, and is therefore classified as Non-Nuclear Safety Related.

This modification does not involve an unreviewed safety question. The following are the bases for this conclusion.

(i) 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. This EP replaces equipment which is non safety and does not interact with any equipment important to safety. The replacement equipment is similar to equipment being replaced, performs the same function and is built and installed to the same standards as the original equipment. Therefore the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety has not increased.

Engineering Package 120-290

Unreviewed Safety Question (USQ) Determination (Cont.)

(ii) The possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report is not created because the replacement components perform the same function and operate in the same manner, therefore the components involved with this modification introduce no new type of accidents and do not interact with any safety related equipment.

(iii) The margin of safety as defined in the bases for any Technical Specification is not reduced since the components being replaced do not form the bases of any Technical Specification.

The implementation of this PC/M does not require a change to the plant Technical Specifications, nor does it create an Unreviewed Safety Question.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an Unreviewed Safety Question and prior Nuclear Regulatory Commission approval for the implementation of this PC/M is not required.

Engineering Package 173-290

ABSTRACT

This Engineering Package (EP) covers modifications to the motor operated valves (MOV's) identified in NRC Generic Letter 89-10, which were not previously modified by PC/M 006-287 in response to IE Bulletin 85-03.

This EP will provide the engineering and design details required to implement the close to open torque bypass switch and closed position indication wiring modifications for the motor operated valves.

The MOV's in some of these systems are required for plant safe shutdown and classified as Class 1E, are seismically qualified and perform a safety related function. Therefore, this PC/M is considered Nuclear Safety related.

This EP does not constitute an unreviewed safety question since the modifications described above were reviewed in accordance with 10CFR50.59 and will not have an adverse impact on plant operations or safety related equipment.

The implementation of this PC/M does not require a change to the plant Technical Specifications.

The safety evaluation demonstrates that this EP does not involve an unreviewed safety question and therefore, prior Commission approval for the implementation of this PC/M is not required.

Engineering Package 173-290

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This EP provides the engineering and design details required to install additional rotors and/or internal limit switch wiring changes to MOV's. Recommended action to satisfy item "b" of NRC Generic Letter 89-10 increases the closed to open torque bypass switch settings which impact the closed position indicating light. Increasing the number of rotors from two to four will allow the limit switch for the closed position indicating light to be located on a rotor other than that used for the torque bypass switch. Motor operated valves that have four rotors will only require internal wiring changes. The addition of the new rotors does not affect the existing equipment qualifications.

The implementation of this EP increases the availability of the MOV's during safe shutdown conditions and improves the MOV position indication provided to the Control Room operators.

Most of the motor operated valves that are to be modified by this EP perform safety related functions. The others are defined in the generic letter as position-changeable MOV's that are designed for operation under design-bases conditions.

Based on the preceding, the following conclusions can be made:

(i) 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, since the modifications to the MOV's enhances the operability of the equipment. The addition of rotors and/or internal wiring changes to the valves will prevent the possibility of inaccurate remote closed position indication resulting from the increased bypass limit switch settings.

(ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type than any previously evaluated. This modification alters accident mitigating equipment to enhance their operation. There was no introduction of any new failure mode for the equipment.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The safety function that is controlled by the various applicable Technical Specifications is maintained by this change. The proposed design ensures that the MOV's will function as assumed during an accident. Thus, the margin of safety provided by the Technical Specifications is preserved.

Engineering Package 173-290

Unreviewed Safety Question (USQ) Determination (Cont.)

The implementation of this PC/M does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

Engineering Package 124-289

ABSTRACT

This Engineering Package (EP) provides design for the replacement of four (4) existing Steam Generator Blowdown System (SGBS) Containment Isolation valves 2I-FCV-23-3, 4, 5 and 6. Valves 2I-FCV-23-3 & 5, located outside Containment, perform containment isolation function. Valves 2I-FCV-23-4 & 6, located inside containment, provide protection to the equipment in the penetration area and RAB from a SGBS line break. The original manufacturer of the existing valves and associated actuators, has gone out of business. Therefore, spare parts are not available from the original equipment manufacturer and the valves, associated actuators and accessories are being replaced with Fisher Controls models. The replacement valves will perform the same function and in a similar manner as the existing valves.

The modification considered in this EP is classified as Nuclear Safety Related because the SGBS Containment Isolation valves perform a safety related function.

Design details are provided for installation of new valves, associated actuators, accessories and modifications to existing valve supports, instrument air lines, and conduit.

The safety evaluation has shown that this EP does not constitute an unreviewed safety question as defined in 10CFR50.59 and implementation of this EP does not require a change to the Plant Technical Specifications. Therefore, prior NRC approval is not required for implementation of this EP.

The implementation of this EP will have no adverse impact on plant safety or operation.

Supplement 1:

Revision 1 of this EP provides revised design for 2" socket weld (SW) ends for valves 2I-FCV-23-4 and -6, and 3" butt weld (BW) ends for valves 2I-FCV-23-3 and -5. The valve body size was increased to sustain the seismic load of the larger actuator required to operate the replacement valves. Revision 1 also provides the EQ documentation required to remove the hold point stated in Revision 0 of the EP. Instrument sensing lines, conduit and cable for other equip men: not associated with the replacement of the subject valves will be relocated to accommodate the larger valves and actuators. The safety evaluation has been reviewed and expanded to incorporate the seven question format, and the conclusion that no unreviewed safety questions have been created remains valid.

Engineering Package 124-289

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The proposed activity does not increase the probability of occurrence of an accident because the replacement valves perform the same isolation function as the current valves to the same criteria.

The replacement valves are designed to the same requirements as the existing valves including the requirements of ASME Section III, 1974. The new valves 21-FCV-23-3 and 2I-FCV-23-5 are deigned to close within 5 (five) seconds upon CIAS and/or high radiation signal. The valve electrical components are environmentally qualified to IEEE 323-1974 requirements and seismically qualified to IEEE 344-1975. Therefore, the new valves will perform the same function in a similar manner as the existing valves. The modified pipe supports are adequate for the new loads in accordance with the requirements of FSAR Section 3.9.3.4.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The proposed activity does not increase the consequences of an accident. The valves perform the same function and are designed and qualified to the same codes and standards.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety, because the replacement valves are expected to be more dependable than the current valves. The replacement of valves will not affect other equipment, since they function in the same manner to isolate SGBS line rupture effects. The new valves meet applicable code requirements and the supports were reanalyzed.

Engineering Package 124-289

Unreviewed Safety Question (USQ) Determination (Cont.)

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The valves and pipes for the system are not addressed in the FMEA (SAR Table 6.2-5) for the system. The proposed activity does not increase the consequences of a malfunction of equipment important to safety, because the valves are supplied with air flow control valves which will be set to open slowly to avoid flow induced vibration in the downstream piping. The valves will isolate within 5 seconds similar to the current valves. Globe valves form a better seat than the current gate valves, assuring isolation from containment accidents and SGBS line rupture.

E. Does the proposed activity create the possibility of an accident of a different type than previously evaluated in the SAR?

The proposed activity does not create any possibility of an accident of a different type, because the valves are Seismic Category I and redundant. No new failure modes are created.

F. Does the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the SAR?

The proposed activity does not create the possibility of a different type of malfunction of equipment important to safety. This EP does not modify the intended operation or test requirements of the system because the replacement valves are designed to the same requirements of ASME Section III, 1974 Edition as the existing valves. The valves are tested to the requirements of ASME Section III 1986 Edition. A code reconciliation was performed as required by ASME Section XI, paragraph IWA 7210 to verify that use of the revised code is acceptable. The new valves are designed to close within 5 (five) seconds and are environmentally qualified to IEEE 323-1974 requirements. Therefore, the new valves will perform the same function in the similar manner as the existing valves.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any technical specification?

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification, because the valves are built and installed to the same requirements as the current valves. Technical Specification 3.9.4 and 3.6 will not be affected by this change.

Engineering Package 134-289

ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to change out all commercial grade Agastat Series 7000 Relays used in safety related applications at St Lucie Unit 2 with nuclear grade Agastat Series E7000 Relays. This Engineering Package involves the following tasks:

1) The replacement of two Agastat Model 7032PBB Relays tagged TD3/1601 and TD3/1611 which are located, one each, in the Diesel Generator 2A and 2B Control Panels, respectively. The Model 7032PBB relay performs both an On-Delay and Off-Delay function but is not available from the manufacturer, Amerace, with qualification for Class IE service. Therefore, each Model 7032PBB relay will be replaced with two Class IE qualified Agastat Relays, one each of Models E7012PB (On-Delay) and E7022PB (Off-Delay). All drawings affected by this modification will be revised accordingly.

2) The replacement of all commercial grade Agastat Series 7000 Relays used in safety related application with nuclear grade Agastat Series E7000 Relays. The commercial grade Series 7000 relays currently installed in Nuclear Safety Related applications were provided by equipment manufacturers as components of larger assemblies which were qualified as a unit (i.e. the Reactor Turbine Generator Board (RTGBs) and Heating and Ventilation Control Board (HVCBs). This replacement is being implemented to facilitate future relay replacements and to address qualified life concerns associated with the Series 7000 Relays. All drawings affected by this modification will be revised accordingly.

3) All vendor drawings, Instruction Manuals and like documents will be updated to remove reference to relay model numbers to facilitate future replacements. The St. Lucie Unit 2 Total Equipment Data Base (TEDB) and Bill of Materials (Ref. drawing 2998-B-325) will be updated to reference the correct relay model numbers, and these two documents will be the only permanent plant documents which will contain this information.

This Engineering Package is classified as Nuclear Safety Related since the Agastat Relays being replaced are used in safety related applications and since this package involves modification of the 2A and 2B Diesel Generator Air Start Controls circuitry.

A safety evaluation of this modification has been performed in accordance with 10 CFR 50.59. This evaluation indicates that implementation of this Engineering Package does not involve an unreviewed safety question nor a change to Plant Technical Specifications and has no detrimental effect on plant safety or operation. Therefore, prior NRC approval for implementation of this modification is not required.

Engineering Package 134-289

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The proposed activity does not increase the probability of occurrence of an accident since the control circuit modifications required for the replacement of the Agastat Model 7032PBB Relays with Agastat Model E7012PB & E7022PB Relays do not change or affect the original design intended for the Diesel Generators Air Start System. Furthermore, the replacement of the existing Agastat Relays will not affect the overall performance and operation of the Diesel Generators Air Start System. Finally, this modification will not result in an increase in the probability of a previously evaluated accident within a frequency class or result in a change from a less frequent class to a more frequent class.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

The proposed activity does not increase the consequences of an accident since the control circuit modifications required for the replacement of the Agastat Model 7032PBB Relays with Agastat Model E7012PB & E7022PB Relays do not change, degrade or prevent actions described or assumed for any accident as discussed in the FSAR. In addition, the replacement of the existing Agastat Relays does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed activity does not increase the probability of occurrence of a malfunction of equipment important to safety because the control circuit modifications required for the replacement of the Agastat Model 7032PBB Relays with Agastat Model E7012PB & E7022PB Relays do not affect the original design basis for the Diesel Generators Air Start System. Furthermore, the replacement relays meet all seismic and environmental qualification requirements. Finally, this modification does not, either directly or indirectly, degrade the performance of any safety system below the original system design basis.

Engineering Package 134-289

Unreviewed Safety Question (USQ) Determination (Cont.)

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

The proposed activity does not increase the consequences of a malfunction of equipment important to safety because the change of the Air Start Control circuitry does not change the original design intent for the Diesel Generators Air Start System. This modification does not increase the probability of radiological hazards nor does it adversely affect the seismic integrity of the Diesel Generator Control Panel 2A & 2B.

E. Does the proposed activity create the possibility of an accident of a different type than previously evaluated in the SAR?

The proposed activity does not create any possibility of an accident of a different type than previously evaluated in the SAR since the circuit modifications of this EP do not change the intended design function of the Diesel Generators Air Start Motors start circuitry. The replacement relays and modified circuits will perform the same timing functions as the existing relay. A review of the Failure Mode for the Diesel Generator Air Start Systems has been performed and it has been concluded that no new failure modes are created by implementation of this modification. In addition, the overall seismic integrity of the Diesel Generator Control Panel 2A & 2B will not be degraded by this modification.

F. Does the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the SAR?

The proposed activity does not create the possibility of a different type of malfunction of equipment important to safety because the change in the Diesel Generator Air Start control circuitry does not introduce any new Failure Modes. The replacement of Agastat Model 7032PBB Relays with Agastat Model E7012PB and E7022PB Relays will not change the intended design bases of the original Diesel Generator Air Start Systems. The replacement of the Safety Related Relays will not affect the reliability or performance of the Diesel Generators Air Start System.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any technical specification?

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification since the changes to the Diesel Generator Control Systems does not affect the original design bases for the Systems. Also, the replacement of the Agastat Relays are of component level which is not included in the basis of the Technical Specification for the Diesel Generator Air Start System or any Technical Specification.

Engineering Package 290-289

ABSTRACT

In 1988 the Nuclear Regulatory Commission (NRC) amended its regulation to 10 CFR Part 50 by adding a new section 10CFR50.63, Station Blackout and added Regulatory Guide 1.155, Station Blackout. The Rules require light-water cooled nuclear power plants to be able to withstand a complete loss of AC power for a specified duration and to achieve safe shutdown (hot standby) during that period. The Rules also require the Licensee who had operating plants as of July 21, 1988 to submit certain information regarding the plant's capability to withstand a Station Blackout to the NRC by April 17, 1989. A study was prepared and it recommended that a tie, connecting the existing safety swing 4.16 kV switch gears (lAB & 2AB) of St. Lucie Units 1 & 2, be designed to allow powering of one train of each units emergency busses from one available emergency diesel-generator set.

This EP encompasses the engineering/design details for the installation of the St. Lucie Unit 2 portion of the cross-tie, including construction of raceway, repulling of cables, modifications to cubicles in 4.kV Switchgears 2AB, 2A3, and 2B3, and the necessary control features in preparation of a future Station Blackout tie to be completed in PC/M 028-190.

This tie will be utilized in case of a unit blackout, to transfer power for unit shutdown to the blacked out unit's essential equipment, from the available emergency diesel generator set.

These modifications involve Nuclear Safety Related equipment and functions, therefore, this EP shall be classified as Nuclear Safety Related.

This Engineering Package does not constitute an unreviewed safety question since the modifications described above were reviewed in accordance with 10CFR50.59 and were determined to have no adverse impact on plant operation or safety related equipment. The implementation of this PCM may require a change to the Plant Technical Specifications, depending upon final NRC ruling on Station Blackout. NRC approval of the FPL submittal outlining the proposed blackout resolution is required prior to implementation of the complete blackout resolution scheme. This PC/M can be implemented without prior NRC approval, as it will not, in itself, affect any of the plant safety systems or functions.

Engineering Package 290-289

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of Safety as defined in the bases for any technical specification is reduced.

This Engineering Package provides the engineering and design details for the installation of the St. Lucie Unit 2 portion of the cable cross-tie connecting the existing safety related swing 4.k Switchgears lAB and 2AB in preparation of the Station Blackout tie to be completed in PC/M 028-190.

This Station Blackout tie will be utilized in case of a unit blackout, to transfer power to achieve safe shutdown (hot standby) conditions, from any available Emergency Diesel Generator

Modification to the 4.k switch gears (2AB, 2A3 and 2B3) will include replacement of spare current transformers, over current relays and ammeter scales and removal of spare ground detection current transformer and relays. The Instrumentation and Control portion of this modification will consist of the installation of a key operated control switch and indicating lights on RTGB 201. Utilization of existing spare EDG breaker contacts through an isolator prevents any adverse impact on the plant.

These modifications involve Nuclear Safety Related equipment and functions, therefore, this EP has been classified as Safety Related.

Based on the preceding, the following conclusions can be made:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased by this modification. This EP modifies cubicles in the 4.k switch gear and adds control and indication in preparation of a future Station Blackout tie. The equipment added or modified by this EP will be non-functional after implementation of this package and will not affect any normal plant operations. Therefore, the implementation of this package cannot increase the probability of occurrence or the consequences of an accident or malfunction of equipment.

(ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type other than any previously evaluated. The added and modified equipment remains non-functional after implementation of this EP and will not affect any normal plant operations. No new failure modes have been introduced by this modification. The equipment added/modified by this EP will not prevent safety related equipment from performing their functions.

Engineering Package 290-289

Unreviewed Safety Question (USQ) Determination (Cont.)

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The modified equipment will continue to perform in the same manner since the modification involves spare breakers and non-functioning controls.

The implementation of this PC/M does not require a change to plant Technical Specifications.

However, during implementation of this modification, care should be taken to maintain the minimum operable electric power sources, per Plant Technical Specifications 3/4.8.1, "Electric Power Systems: A.C. Sources", and 3/4.8.3 "Electric Power Systems: Onsite Power Distribution". Also during implementation, fire barriers will be breached in the control room floor under RTGB 201, SAB section and the provisions of the plant Technical Specifications Section 3/4.7.12 shall apply until integrity of the fire barrier is restored.

The foregoing constitutes, per 10 CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 311-289

ABSTRACT

This Engineering Package provides the engineering and design details required to implement the replacement of the existing ionization smoke and duct detectors and the existing Main Fire Alarm panels. The smoke detectors and panels are part of the fire detection system.

The existing detectors are divided into two groups: the originals (installed 9 years ago) which are obsolete; and their replacements (installed as the originals failed) which are no longer manufactured. To ensure the reliability of the fire detection system, new ionization smoke detectors will be installed.

The existing panels are obsolete and spare parts are no longer available. The replacement panels represent the current industry standard in hardware and software. The new detectors and panels are compatible with the existing plant fire detection system computer.

The fire detection system, which is part of the fire protection system, is non-safety related, but is provided in areas that contain or present a fire hazard to equipment essential to safe plant shutdown. Therefore, this Engineering Package (EP) is classified as Quality Related.

This EP was reviewed in accordance with 10CFR50.59 and was found not to constitute an unreviewed safety question. The modifications described above have no adverse impact on plant operations or safety and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval is not required for the implementation of this EP.

Engineering Package 311-289

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; (II) if a possibility for an accident or malfunction of a different type than any evaluated in the Safety Analysis Report may be created; or (III) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This EP provides the engineering and design details required to replace the existing ionization smoke and duct detectors and Main Fire Alarm panels with new equipment. The existing equipment is either obsolete or no longer manufactured. The new detectors and panels are compatible with the existing plant fire detection system computer.

The implementation of this EP will improve the reliability of the fire detection system by replacing obsolete equipment. This ensures the availability of the individual detectors to detect a fire and that spare parts will be obtainable in case of equipment failure.

Fire detection systems are provided in areas that contain or present a fire exposure to equipment essential to safe plant shutdown. Therefore, this EP has been classified as Quality Related.

Based on the preceding, the following conclusions can be made:

(1) 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 by these modifications. The replacement of the obsolete detectors and panels enhances the operability of the equipment and the fire detection system. The new detectors and panels have the same characteristics as the existing equipment. The possible failure of this equipment will not prevent safety related equipment from performing their intended functions. The detectors and panels are not required during an accident condition. Therefore, the implementation of these modifications cannot increase the probability of occurrence or the consequences of an accident or malfunction of equipment.

(ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type other than any previously evaluated. The detectors and panels are not required during an accident condition nor will they prevent safety related equipment from performing their intended function. This modification does not affect any safety related equipment.

Engineering Package 311-289

Unreviewed Safety Question (USQ) Determination (Cont.)

(iii) The margin of safety as defined in the bases for any Technical Specification is not reduced by this modification. The functions of the fire detection system that are controlled by the applicable Technical Specifications 3/4 - 3.3.3.7 are maintained by this change. The implementation of this PC/M does not require a change to plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 281-288

ABSTRACT

An Engineering Package is required to modify the Reactor Cavity Seal Ring (seal ring) so that it can be filled with water during reactor refueling. The water-filled ring girder will be more effective as a shield against radiation exposure. The water will be removed from the seal ring and discharged to the floor drains of the reactor Building at the conclusion of the refueling operation.

The modifications included in this EP include providing pipe plugs on each section of the Reactor Cavity Seal Ring toroid to permit filling and draining of water and providing 2 inch diameter holes through the seal plate to access the plugs. Any required structural modifications to the seal ring, its supports and lifting lugs are also included.

The Reactor Cavity Seal Ring is non-nuclear safety related and son-seismic, based on the St. Lucie Unit 2 FSAR and the fact that its use is for refueling operations to allow filling the Reactor Cavity with water. This EP is classified Quality Related to obtain added assurance that the Reactor Cavity Seal Ring will not adversely affect the Reactor Vessel, since the seal ring is lifted over and installed over the open Reactor Vessel.

This modification does not involve an unreviewed safety question, has no effect on plant safety and operation, and does not involve change to any plant Technical Specification, based on a 10CFR50.59 review. Therefore, prior NRC approval is not required for implementation of this Engineering Package.

Engineering Package 281-288

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety analysis Report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this EP are limited to the Reactor Cavity Seal Ring which is non-safety-related. The modifications consist of adding penetrations and pipe plugs to the seal ring to allow it to be filled with water. This water provides additional radiation shielding while the seal ring is in place. The water will be removed at the conclusion of the outage. The seal ring lifting lugs are modified to meet the requirements of Subsection 3.8.4 of the St. Lucie Unit 2 SAR since the lifted load has increased. No safety-related equipment is affected. This modification is classified as quality Related to obtain additional control during the engineering, design and installation to assure that the Reactor Cavity Seal Ring will perform its intended function and will not adversely affect the reactor vessel. Based on the above, the following provides the justification that an unreviewed safety question does not exist:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased. The modifications included in this EP have been reviewed aid it has been determined that they will not adversely affect safety-related equipment. Since the Reactor Cavity Seal Ring is not considered by the SAR in determining the probability of accidents, possible types of accidents, or in the evaluation of consequences of accidents, it can be concluded that the probability of occurrence or the consequences of accidents previously addressed in the SAR remain unchanged.

ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated. The seal ring will be filled with water only during outages when the plant is shut down. The structural members affected by this modification have been analyzed for the increased load and have been found to be acceptable. The sealing portion of the seal ring has not changed. Therefore, the possibility of an accident of a different type has not been created.

iii) This modification does not change the margin of safety as defined in the basis for any Technical Specification. The installation of the seal ring is not specifically addressed in the Technical Specifications. Furthermore, the sealing portion of the seal ring has not been changed and structural effects have been reviewed and found to be acceptable.

The implementation of this change does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PCM is not required.

Engineering Package 065-287

ABSTRACT

This modification replaces ITT Barton Model 227, 288A, 580 and 580-1 differential pressure switches used in the Intake Cooling Water System with Meriam Instrument Model 1226-2 differential pressure switches in order to provide more reliable switches and better means of replacing internal parts.

Some of the differential pressure switches are used to measure differential pressure across the Screen Wash Pump strainers and the strainers on the ICW inlet to the CCW and TCW heat exchangers. Each switch provides local indication and is wired into a control room annunciator which alarms on high differential pressure.

Other differential pressure switches are used to measure ICW outlet flows from the CCW and TCW heat exchangers by measuring the differential pressure across flow elements. Each switch provides local indication and the instruments which measure the flows from the CCW heat exchangers are wired into control room annunciators which alarm on low flow.

The remaining pressure switches are used to measure the lubricating water pressure to the ICW pumps. Each switch provides local indication and is wired into a control room annunciator which alarms on low pressure.

Engineering Package 065-287

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of Safety as defined in the bases for any technical specification is reduced.

This Engineering Package involves the replacement of existing ITT Barton differential pressure switches with new differential pressure switches by Meriam.

These switches provide alarm and indicating functions in the Nuclear Safety Related Intake Cooling Water System. Switches which connect to seismically designed piping are seismically qualified and seismically mounted.

The replacement Meriam differential pressure switches for the Intake Cooling Water Pump lubricating water pressure function have less repeatability accuracy than the instrument accuracy stated in FSAR Table 9.2-3. It has been determined that the change from 0.5% to 1.0% in the accuracy for switches which only input to alarms has no impact on Nuclear Safety.

These modifications have been evaluated under 10CFR50.59 and it has been determined that this EP does not involve an unreviewed safety question. The following are the bases for this conclusion.

(i) 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 because the new switches are functionally a one-for-one replacement for existing equipment and cannot fail in any manner which would inhibit the fulfillment of any safety related functions.

(ii) The possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report is not created because the new pressure switches introduce no new type of accidents and cannot cause malfunctions of any safety related equipment.

(iii) This modification does not reduce the margin of safety as defined in the base for any Technical Specification because it does not affect any of the bases in the Technical Specifications.

The implementation of this PC/M does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Nuclear Regulatory Commission approval for the implementation of this PC/M is not required.

Engineering Package 196-285

ABSTRACT

This Engineering Package covers the spare replacement of the Analog Display System (ADS) Video display monitor in the RTG Board Section 204 with Ramtek Model GM-720. The existing video display monitor in the RTGB is a Conrac Model 5211 which is no longer manufactured. The ADS monitors the vertical positions and movements of the 91 Control Elements Assemblies (CEA), utilizing the signals from reed switch position transmitters. The CRT in the Control Room provides the operator with one of the two continuous video graphic displays for CEA positions. The CEA position system is Non-Safety Related (see FSAR Section 7.5.1). However, the associated mounting assembly in the RTGB must be seismically qualified, mandating the PCM to be classified as "Quality Related."

This EP was reviewed in accordance with 10CFR50.59 and was found not to constitute an unreviewed safety question. The modifications described above have no adverse impact on plant operations or safety and do not require a change to the plant Technical Specifications. Therefore, prior NRC approval is not required for the implementation of this EP.

Engineering Package 196-285

Unreviewed Safety Question (USQ) Determination

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; (ii) if a possibility for an accident or malfunction of a different type than any evaluated in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

(i) This modification described in the PCM replaces the existing CRT monitor associated with the Analog Display System. The Analog Display System monitors the vertical positions and movements of the 91 Control Elements Assemblies (CEA), utilizing the signals from reed switch position transmitters. The CRT in the Control Room provides the operator with one of the two continuous video graphic displays for CEA positions. No modification to the system is initiated by this PCM since it uses a one for one replacement.

(ii) The failure of this component to function would not affect the safe shutdown of the unit since it is not required to shutdown the reactor, cool the core, or cool another safety system in the reactor containment (after an accident); nor is it part of any system that reduces radioactive release during an accident. The housing is required to withstand loadings induced by design basis earthquake. Therefore, this PCM is classified "Qualified Related."

The modifications to the RTGB-204 is analyzed as to maintain the seismic integrity of the equipment.

(iii) The implementation of this PCM does not require a change to the plant's Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PC/M is not required.

Engineering Package 053-284

ABSTRACT

This PCM installs Schnoor springs on the 2A and 2B AFW pumps. During plant startup, the AFW pump vendor, Ingersoll-Rand, indicated that the balancing drum could potentially rub against a stationary part of casting when either of the AFW pumps are started from idle. Once a pump is running, however, a one mil clearance is established. When the 2C AFW pump was returned to the vendor's shop for some rework, the vendor deemed it beneficial to add a Schnoor spring to the thrust bearing housing. This spring exerts a force on the shaft assembly, pushing it towards the thrust bearing, and thereby preventing the rubbing of the balancing drum against the casing.

The AFW pumps provide an alternative source of feedwater to the Steam Generators in the event that main feedwater is not available. This system is fully described in FSAR Section 10.4.9. This function will not be altered in any way, although this modification will increase reliability by preventing any possible rubbing of internal parts.

A safety evaluation of this modification has been performed in accordance with 10 CFR 50.59. This evaluation indicates that implementation of this Engineering Package does not involve an unreviewed safety question nor a change to Plant Technical Specifications and has no detrimental effect on plant safety or operation. Therefore, prior NRC approval for implementation of this modification is not required.

Engineering Package 053-284

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. The probability of occurrence of an accident previously evaluated in the SAR has not been affected since the probability of failure of the AFW pumps does not affect the probability of occurrence of a SAR design basis accident.

B. The consequences of an accident previously evaluated in the SAR have not been changed since this modification does not affect the operability of any equipment required to mitigate the effects of an accident.

C. The probability of malfunction of equipment important to safety previously evaluated in the SAR has not been adversely affected since the vendor, Ingersoll-Rand states: "the bevel springs furnished by Ingersoll-Rand for installation in the bearing housings of the subject units are suitable for the intended service, and will not have any adverse affect on the structural integrity or performance of these units as they were originally qualified." This modification does not affect any other safety related equipment.

D. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR have not been affected for the same reason given in C above.

E. The possibility of an accident of a different type than analyzed in the SAR has not been created since this modification does not provide for a common mode failure which would alter the availability of redundant components provided to mitigate the consequences of a single active failure within the AFWS. Also, failure of any component associated with this modification does not result in an uncontrolled release of radioactive material.

F. The possibility of equipment malfunction of a different type than analyzed in the SAR has not been increased for the same reasons given in C. Additionally, this modification adds material which is already installed on the 2C AFW pump per vendor direction.

G. The margin of safety as defined in in the bases for any Technical Specification is not affected by this modification since the components involved in this modification are not included in the bases of any Technical Specifications.

Engineering Package 085-284

ABSTRACT

This Engineering Package provides the necessary details for the installation of a radiation shielding system around the reactor head. The system consists of temporary shielding blankets and a permanent support structure. The temporary shielding blankets will be installed before removal of the reactor vessel head and remain in place during the refueling outage until the reactor vessel head is returned. This will provide gamma dose reduction for individuals working around the reactor vessel flange area.

The Reactor Vessel Head Shielding does not perform any Nuclear Safety-Related function. The permanent support structure will be mounted directly to the reactor head lift rig and will be in the vicinity of safety related equipment and systems. Since failure of the structure could impact safety-related functions, this Engineering Package has been classified as Quality Related.

A safety evaluation of this modification has been performed in accordance with 10 CFR 50.59. This evaluation indicates that implementation of this Engineering Package does not involve an unreviewed safety question nor a change to the Plant Technical Specifications and has no detrimental effect on plant safety or operation. Therefore, prior NRC approval for implementation of this modification is not required.

REVISION NO. 1, SUPPLEMENT NO. 1

During initial installation of the shielding support system, an interference was encountered between the trolley track and the stud detensioner hoist system mounted on the reactor head lift rig, and the modification was not implemented. Revision 1, Supplement 1 to this PC/M incorporates the design modifications required to eliminate the interference and permit the installation of the shielding support system; the basic change involves the reduction of the radius of the trolley track beam by 4-1/4".

The Westinghouse report (Reference 6.8) documenting the structural and seismic adequacy of this modification has been revised. However, the revision to this document does not affect the conclusions of the Design Analysis or the Safety Evaluation. A report addressing the Heavy Lift Evaluation has been reviewed, and it has been determined that the modifications to the shielding support system included in Revision 1, Supplement 1 of this PC/M do not affect the analysis or conclusions of the report, and therefore do not affect the conclusions of the Design Analysis or the Safety Evaluation. Additionally, this revision does not affect, amend, or change the Plant Technical Specifications. Therefore, prior NRC approval for implementation of this modification is not required.

Engineering Package 085-284

Unreviewed Safety Question (USQ) Determination

As defined in 10 CFR 50.59 an unreviewed safety question exists: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report (SAR) may be increased; or (ii) if a possibility of an accident of malfunction of a different type than any previously evaluated in the SAR may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

In accordance with 10 CFR 50.59, the following evaluation serves to determine whether this modification constitutes an unreviewed safety question:

A. Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the SAR?

The only safety related system or component which the permanently attached structure (indirectly) interfaces with is the reactor vessel. The only effect it has on the reactor vessel is a slight load on the lift lugs. It may be concluded without analysis that the small stress at the lift lugs will have a very negligible effect on the massive reactor vessel nozzles and supports. It is therefore concluded that this change will not increase the probability of occurrence of any accident previously evaluated in the SAR.

B. Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

This change does not directly modify or compromise any safety related component or system. This addition has also been shown to be structurally sound, which means it will not break loose or damage safety related equipment. Therefore, there is no reasonable way to postulate that the proposed modification will increase the consequences of any accident previously evaluated in the SAR.

C. Does the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR?

Since the head shield is designed to safely withstand the effects of an SSE, it does not compromise any safety related equipment. Therefore, the proposed modification will not increase the probability of occurrence of any malfunction of equipment important to safety previously evaluated in the SAR.

D. Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

Engineering Package 085-284

Unreviewed Safety Question (USQ) Determination (Cont.)

Similar to Question 2, this change does not alter or affect safety related equipment, and it will not increase the consequences of any malfunction of equipment important to safety previously evaluated in the SAR.

E. Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the SAR?

The only possible additional accident associated with a minor addition to the reactor vessel head would be LOCA activated missile generation. Since this part in no way intersects the pressure boundary and does not interface with items which may be postulated as missiles (i.e., CRDM'S) and cannot itself become a missile, it is concluded that this change is not involved with missile generation. Therefore, the proposed modification does not create the possibility of any accident of a different type than any previously evaluated in the SAR.

F. Does the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR?

Since the head shield system is non-functional during plant operation, is stationary, and is designed to meet SSE requirements, malfunctions of any type (including different types of malfunction of equipment important to safety than any previously evaluated in the SAR) are not anticipated.

G. Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

The margin of safety as defined in the bases for any Technical Specification is not affected by this modification since the components involved in this modification are not included in the bases of any Technical Specifications. Therefore, the proposed modification does not reduce the margin of safety as defined in the bases for any Technical Specification.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question nor a change to the Plant Technical Specifications, and prior NRC approval for the implementation of this modification is not required.