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July 10, 1992

L-91-201
10 CFR 50.59

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

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

Pursuant to 10 CFR 50.59 (b)(2), the enclosed report contains a brief description and summary of the safety evaluation of Plant Changes/Modifications (PCMs) which were made, and are reportable, pursuant to 10 CFR 50.59. Included with the brief description of each PCM is a summary of the safety evaluation completed by Florida Power & Light Company for that PCM. This report includes PCMs completed between January 23, 1991, and January 22, 1992, and correlates with the information included in Revision 11 of the Updated Final Safety Analysis Report submitted under separate cover.

Should there be any questions on this information, please contact us.

Very truly yours,

D. A. Sager
Vice President
St. Lucie Plant

DAS/JJB/kw

Enclosure

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

DAS/PSL #733-92

9207160192

Plant Change/Modifications reportable pursuant to 10CFR50.59 for St. Lucie Unit 1

| Number | Supplement | Title |
|---|---|---|
| 439-191 | 0 | Intake Cooling Water System Flow Enhancement. |
| 252-191 | 0 | Containment Spray Vent Valve Installation. |
| 216-191 | 0-1 | Steam Generator Tube Stabilization with Flexible Stakes. |
| 160-191 | 0 | Removal of Turbine Runback. |
| 090-191 | 0 | Motor Operated Valve Arc Suppression Varistors. |
| 060-191 | 0 | BAM Tank Low Level Alarm Selector Switch Installation. |
| 403-190 | 0 | Removal of Acoustic Flow Monitor Recorder. |
| 005-190 | 0 | Fisher & Porter Indicating Controllers Replacement. |
| 312-189 | 0 | New Fuel Crane Hoist. |
| 009-189 | 0 | Replacement of Honeywell Fire Detection Panels. |
| 399-988 | 1 | Fuel Dispensing Facility. |
| 073-987 | 0-1 | Fisher & Porter Transmitter Replacement. |
| 020-187 | 0-2 | Replacement of Containment Level Monitoring System. |
| 142-186 | 0-2 | Spent Fuel Pool Rerack Platform Modification. |
| 045-986 | 5 | Installation of a Perimeter Security Barrier, Intrusion Detection and Surveillance System. |

## ABSTRACT

This Engineering Package (EP) provides the engineering to remove the 45° open mechanical stop on Temperature Control Valves I-TCV-14-4A and I-TCV-14-4B and replace orifices I-SO-21-1A and I-SO-21-1B. The purpose of these modifications is to increase the flow capacity of the Intake Cooling Water (ICW) system during accident system alignment, and to enhance the capability of the system to allow increased pressure drop through the strainer and heat exchanger.

This modification will allow the ICW system to pass 14,500 gpm of seawater with an inlet temperature of 95°F considering increased heat exchanger and strainer differential pressure drops. There will be no impact on normal operation since the temperature control valves downstream of the Turbine Cooling Water (TCW), Open Blowdown Cooling Water (OBCW) and Component Cooling Water (CCW) heat exchanger will still modulate flow as required.

The modifications considered in this EP affect the ICW system which is a safety related system. The ICW system is classified as quality Group C and Seismic Category I. Therefore, this modification is classified as Safety Related. The safety evaluation provided in Section 3.0 has shown that this EP does not constitute an unreviewed safety question. Implementation of this EP will have no adverse impact on plant safety or operation, does not require a change to the Plant Technical Specifications and does not reduce the margin of safety for any Technical Specification. Therefore, prior NRC approval is not required for implementation of this EP.

## SAFETY EVALUATION

The proposed change involves the removal of the 45° mechanical stop from valves I-TCV-14-4A and -4B, the recalibration of the valve controller to match the 90° stroke and the replacement of restriction orifices I-SO-21-1A and -1B with orifices having a larger flow area. The proposed change will enhance the flow passing capacity of the ICW system. This change is being made to accommodate for higher heat exchanger pressure drops and will enhance the heat removal capacity of the system in the accident configuration.

The change affects both the ICW and CCW systems. Both these systems are classified as Safety Related by FSAR Subsection 9.2. The effect on the ICW system will be to allow more flow to be passed in the safety related portion of the system. The safety related function of the ICW system is to remove the post accident heat load from the CCW system. During shutdown modes, post accident heat loads are lower than during power operation. The change will not affect the ability of the ICW system to remove post accident heat loads since the change enhances the heat removal capacity of ICW by increasing its flow. The ICW component most directly affected by this change is the ICW pump. Calculations (Attachment 7.4) show the ICW pump will still be operating within its tested performance. Therefore there will be no effect on the operability of the ICW pump.

SAFETY EVALUATION (Continued)

The safety related function of the CCW system is to remove the post accident heat load from containment and reactor via the containment spray system and reactor containment fan coolers. The effect of this change will be to enhance heat removal from the CCW system. Thus the CCW system will meet its design basis requirements.

The increased velocity in the tubes of the CCW heat exchanger is the only other effect. This is not expected to be a concern since design basis calculation (Attachment 7.4) shows the flow with the TCV wide open would not exceed design limits.

The proposed change has no effect on the Technical Specifications. Technical Specification 3.7.4.1 requires the ICW system to be operable. This modification in no way changes the operability of the ICW system. The bases for 3.7.4.1 states that the ICW system is required to provide sufficient cooling. This modification supports this bases and enhances it by providing increased cooling capacity due to higher flow rates.

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.

The modifications have been evaluated under 10CFR50.59 and it has been determined that the modifications included in this Engineering Package do not involve an unreviewed safety question as demonstrated by the answer to the questions below:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report?

   This modification does not increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report. The Temperature Control Valves and Orifice modified by this EP are not considered in the initiation of any accident and do not affect any equipment considered as accident initiating components.

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

   The consequences of an accident previously evaluated in the Safety Analysis Report are not increased with the implementation of this PCM. After implementation of this PCM, the flow capacity of the ICW system in the accident alignment will increase. The system remains capable of delivering the minimum flow requirements for accident conditions. All components retain their functions and capabilities with the increased flow.

SAFETY EVALUATION (Continued)

3) 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 system flow capacity has been enhanced and all components remain capable of operating with the increased flow. The ICW pump flow remains within the tested range. The proposed modification does not increase the severity or possibility of pressure surges on startup of the ICW pumps. A review of the surge pressure study shows that the transient pressure surge is influenced primarily by the minimum position of the temperature control valves and the position of vacuum breakers within the system. This modification does not impact either of these design features.

4) Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report?

This modification does not increase the consequences of a malfunction of Nuclear Safety Related equipment previously evaluated in the Safety Analysis Report. The ICW system remains capable of performing its safety functions with the increased flow. No new failure modes are introduced by this modification. As stated above the consequences of a transient pressure surge have not been changed.

5) Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report?

This modification does not create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report. The ICW system remains within its capacity to function for safety related functions and no new failures are created. The ICW system, before and after the modification, is not considered as initiating any accident.

6) 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 create the possibility of a malfunction of nuclear safety related equipment of a different type than any previously evaluated in the Safety Analysis Report. As stated previously no new failure modes are introduced via this modification and all failures analyzed in the FSAR for the ICW system remain unchanged.

SAFETY EVALUATION (Continued)

7) 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 as defined in the basis for any technical specification at St Lucie - Unit 1. The only margin of safety that could be impacted is the bases for Technical Specification 3.7.4.1. which requires sufficient capacity from the ICW system to cool vital equipment. This modification, by increasing the flow capacity of the system, enhances this margin of safety by providing greater heat sink capabilities during accident conditions.

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 nor impact the Technical Specifications and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required.

PC/M 252-191, Supplement 0

## ABSTRACT

Engineering Evaluation (EE) JPN-PSL-SEMP-91-029, Rev. 0," Engineering Evaluation of Shutdown Cooling System Transient Response", states air in the Containment Spray (CS) header is causing pressure transients in the Shutdown Cooling (SDC) piping when the Low Pressure Safety Injection (LPSI) pumps are operated. As shown in various design documents, the CS header has no means of being vented. The EE recommends that vent valves be installed on the CS header upstream of the containment isolation valves. As valves 1-FCV-07-1A and 1-FCV-07-1B are normally closed and isolate the containment, the above mentioned vents need to be located at high points of the headers upstream of these isolation valves.

This Engineering Package (EP) provides the specific design information necessary to install one 3/4" vent in each CS header immediately up stream of valves 1-FCV-07-1A and 1-FCV-07-1B.

The CS system performs a safety related function, as described in FSAR, Section 6.2.2. As such, this EP has been classified as Safety Related. This EP does not have any adverse impact on plant safety and/or operation. Based on a Failure Mode and Effects Analysis and a review of the changes to be implemented by this EP against the requirements of 10CFR50.59, 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 modification is not required.

## SAFETY EVALUATION

With respect to Title 10 of the Codes of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety questions: (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 Final Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced. The modification here in does not involve an unreviewed safety question because of the following reasons:

1) 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 previously evaluated because the operation and functionality of the CS System has not been changed by this modification. The vents are not used during power operation or involved in any way with any safety function of this system. Based on this, the probability of occurrence of an analyzed accident remains unchanged.

SAFETY EVALUATION (Continued)

2) 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 these vents meet all regulatory requirements specified in the FSAR. Their operating and pressure retaining characteristics were shown to be acceptable. In all operational modes they are a passive pressure boundary component. The vents do not increase the radiological doses of an accident.

3) 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 this system's function and performance remain unchanged by the addition of these vents. Furthermore these vents do not directly or indirectly affect equipment important to safety. These vents do not degrade the reliability or increase challanges, directly or indirectly, for equipment important to safety.

4) 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 vents perform no active nuclear safety related or equipment protection function. The vent valves are administratively controlled to prevent their being left open and are designed to be physically strong enough to preclude being broken off. In the extreme unlikely event that the vent fails, the worse case scenario is the loss of one of the two redundant headers, which has been analyzed in the FSAR. Therefore, their operability will have no affect on the consequences of a malfunction of equipment important to safety.

5) 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 vents are not accident initiating devices. The vent valves are only operated when filling the SDC system and serve no active or controlling function.

SAFETY EVALUATION (Continued)

6) 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 method of operation of the CS system has not changed. The addition of the vents has been analyzed to maintain the seismic and pressure boundary integrity of the CS headers.

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

The proposed activity does not reduce the margin of safety as defined in the bases for any Technical Specification because T.S. Section 3/4.6.2 requires that both loops of the CS be operable, all valves in the CS System be properly aligned and that the pumps and initiation system be tested. The Technical Specification bases ensures that adequate containment cooling and depressurization capability exists during a LOCA and is not affected by this EP.

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 Specification. Prior NRC approval for the implementation of this modification is not required.

PC/M 216-191, Supplement 0-1

## ABSTRACT

The purpose of this Engineering Package is to provide documentation to perform the maintenance practice of installing steam generator flexible tube stakes ("stakes"). Stakes will be installed in those Steam Generator (S/G) tubes exhibiting circumferential cracking near the tube expansion transition region (i.e. at or near the top of the tubesheet). In-Service Inspection (via eddy current testing) will determine when circumferential cracking exists and when installation of stakes are necessary. The stakes are designed to prevent tube to tube contact, should the crack progress and the tube sever. All staked tubes are to be plugged in accordance with PC/M 108-191M.

This PC/M is classified as Safety Related, since the stakes may be relied upon to prevent a severed tube from contacting other S/G tubes, which are part of the RCS pressure boundary.

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 for implementation is not required.

**It is noted that implementation of this PC/M is contingent upon Facility Review Group approval of Revision 1 to PC/M 108-191M (S/G Tube Plugging). PC/M 108-191M Revision 1 will provide the documentation to permit the installation of the current ABB/Combustion Engineering welded S/G tube plugs.**

This PC/M provides the documentation to install stakes as a maintenance practice; the S/G instruction manual will be revised to provide the drawings necessary to perform this maintenance on an "as-needed" basis.

<u>SUPPLEMENT NO. 1</u>

Supplement No. 1 to this Engineering Package provides revised drawings of the S/G flexible tube stakes ("stakes"). Neither the design, the installation, nor the performance of the stakes is adversely affected by these drawing revisions. Supplement No. 1 does not effect, amend nor change the original Safety Evaluation, the Technical Specification or the Technical Specification bases.

PC/M 216-191, Supplement 0-1

SAFETY EVALUATION

Since the primary function is to prevent the staked tube from contacting other tubes, it has the potential to affect the RCS pressure boundary, this Engineering Package is classified as Safety Related. An FSAR Change Package (Attachment 7.6) for Section 5.5.1.3a is included to delineate installation of stakes in plugged S/G tubes which have circumferential cracks near the expanded region just above the S/G tubesheet.

The addition of stakes does not alter the operation, performance nor function of the S/G. Since it will be installed in tubes which are being plugged, it does not reduce the available S/G heat transfer area insofar as S/G plugging limits and asymmetry between S/G's is addressed by PC/M 108-191M. The stakes are being installed in a tube with circumferential cracks to prevent tube-to-tube contact should the staked tube sever. S/G operation with the tube stakes is bounded by the Technical Specifications (Reference 6.2). The licensing basis has not been altered by this modification.

A review of the potential failure modes for the S/G, as discussed in FSAR Chapter 5, was performed and is documented in Table 3-1. It was determined that no new failure modes are introduced, nor the probability of existing failure modes are increased by this modification.

The modifications associated with this Engineering Package do not affect any Limiting Condition for Operation (LCO) nor any Technical Specification Basis. The LCO's (and associated basis) reviewed were 3/4.4.5 and 3/4.7.2. Therefore this modification is bounded by the Technical Specifications (Reference 6.2).

With respect to Title 10 of the Code of Federal Regulations, Part 50.59(a)(2), 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 may be created; or (iii) if the margin of safety as defined in the basis of any technical specification is reduced. The modification provided by this Engineering Package does not involve an unreviewed safety question because each of the following seven (7) questions are answered with a negative response.

SAFETY EVALUATION (Continued)

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR)?

The function of the stake is to prevent the staked and plugged tube from contacting another tube should the former sever. Only those tubes with a circumferential crack near the expansion transition region will be staked. The installation of a stake does not increase the probability of a plugged tube severing, nor does it increase the probability of a S/G tube plug failure. Therefore this modification does not increase the probability of occurrence of an accident previously analyzed in the SAR.

2) Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

Accidents previously evaluated include S/G tube rupture (FSAR Section 15.4.4). S/G tube stakes will be installed in plugged tubes (isolated from primary systems) with the design criteria of preventing tube to tube contact in the event the plugged tube severs, under design conditions. Since the stakes will be installed in non-active tubes the consequences of a S/G tube rupture are not affected by this modification.

Other accidents previously analyzed include Main Steam Line Break (FSAR 15.4.6); Main Feedwater System (FSAR 15.2.10), and Control Malfunction.

The stake is designed to perform its function under design conditions, i.e. stabilize the staked tube under all conditions the S/G is subjected to. Therefore, this modification does not increase the consequences of an accident previously evaluated in the SAR.

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

Since the S/G tube stakes will be installed in plugged tubes (inactive) it can not affect active tubes or other components in the S/G unless its S/G tube degrades and severs. Should this event occur, the S/G tube stake will be retained within both ends of the S/G tube and not contact other S/G tubes or S/G internals, under design conditions.

SAFETY EVALUATION (Continued)

4) 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 plugged tubes, isolated from active tubes and S/G internals. It is not considered credible to postulate the failure of a S/G support component such as a stake. Therefore, the stake will remain in the plugged tube and will not interact with other S/G internals. Since the purpose of this modification is to minimize the probability of a plugged tube from contacting other tubes, the S/G tube stakes do not increase the consequences of a malfunction of equipment important to safety.

5) 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 crack near the expansion transition region above the tubesheet and the tube is subsequently plugged. Since the stakes are isolated from the RCS and the failure of a support component is not considered credible, the stake by itself cannot 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.

6) 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. Should the tube in which it is installed continue to degrade and sever, it is designed to prevent tube-to-tube contact. It is designed to remain in place in a severed tube and to have minimal effect on the wear of the staked tube. It does 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.

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

The Technical Specification (Reference 6.2) basis reviewed were 3/4.4.5, 3/4.4.6.2 and 3/4.7.2. 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, nor affect the capability to detect imperfections, nor does it inhibit or promote RCS leakage.

SAFETY EVALUATION (Continued)

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

PC/M 160-191, Supplement 0

## ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to delete the Main Turbine Runback feature. The modification will leave the Turbine Runback logic unchanged. It will no longer be initiated. Turbine Runback occurs whenever there is a loss of a Steam Generator Feed Pump (SGFP) above 60% power or the loss of both Heater Drain Pumps (HDP) above 92% power. Deleting the Turbine Runback feature will be accomplished by lifting the Turbine Runback leads in the Turbine Digital Electronic Hydraulic Control (DEH) Cabinet, Sequence of Events Cabinet, and at Reactor Turbine Generator Boards (RTGB) 101 & 102. Cables 10712F, 10712J and 10717Z will be spared and all drawings will be revised accordingly. In addition the Control Room Turbine Runback Annunciator Window D27 will be rendered inoperative and will be spared.

The function of the Turbine Runback is to run the Turbine/Generator back at a predetermined rate upon loss of a SGFP or both HDP's until Turbine/Generator output decreases to 60% & 92% respectively as measured by turbine first stage (impulse) pressure. During a Turbine Runback the Main Governor valves throttle the steam flow until the load matches the setpoint of 60% or 92% load depending on the initiating event. During this event the Turbine/Generator RPM remains constant.

St. Lucie Unit 1 experienced a Reactor trip from 100% power on June 14, 1987 & June 30, 1988, due to a Turbine Runback which was caused by the loss of the 1B SGFP (Ref. 6.7 and 6.8). During both events a turbine runback was automatically initiated to approximately 60% power. In less than 30 seconds into the transient the Reactor Protection System initiated a Reactor trip on a high pressurizer pressure signal.

The purpose of removing the Turbine Runback feature is to minimize the effects of a partial loss of feedwater transient on the Plant and to provide the plant operators with additional time to restore 100% feedwater flow before a Reactor trip occurs.

A Thermal Hydraulics Analysis (Ref. 6.9) of a loss of SGFP transient was performed with and without the Turbine Runback feature. This analysis demonstrates that by removing the Turbine Runback feature a Reactor trip could be avoided, provided the plant operators restore 100% feedwater flow within 110 seconds into the event. If full feedwater flow is not restored within the 110 seconds, a reactor trip will occur on low steam generator levels. This transient will not challenge the Pressurizer Power Operated Relief Valves (PORV's) or the Main Steam Safety Valves (MSSV's) which lifted in the two SGFP loss events mentioned earlier.

Although the Main Turbine, Turbine Controls and the Turbine Runback feature do not perform a safety function per FSAR Section 7.7, this EP is classified as Quality Related because it requires work to be performed in the Control Room.

ABSTRACT (Continued)

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.

## SAFETY EVALUATION

This Engineering Package (EP) provides the engineering and design necessary to delete the Main Turbine Runback feature. Turbine Runback occurs whenever there is a loss of a Steam Generator Feed Pump (SGFP) above 60% power or the loss of both Heater Drain Pumps (HDP) above 92% power.

The purpose of removing the Turbine Runback feature is to minimize the effects of a partial loss of feedwater transient on the plant and to provide the plant operator with additional time to restore 100% feedwater flow before a Reactor trip occurs.

There are no licensing requirements impacted by this modification.

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. The modification included in this engineering package does not involve an unreviewed safety question because of the following reasons:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in SAR?

   The proposed activity does not increase the probability of occurrence of an accident previously evaluated because the functionality of the Main Turbine, Turbine Controls or Reactor Protection trip signals have not been changed by this modification. Based on this, the probability of occurrence of an analyzed accident remains unchanged.

SAFETY EVALUATION (Continued)

2) 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 due to the deletion of the Main Turbine Runback feature because the Main Turbine, Turbine Controls, or Runback feature do not serve a Safety Related function.

3) 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 a malfunction of equipment important to safety because deletion of the Turbine Runback feature does not adversely affect Safety Related equipment. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety.

4) 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 deletion of the Turbine Runback feature does not adversely affect Safety Related equipment. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety.

5) 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 during the loss of a SGFP transient without Turbine Runback the heat removal from the Reactor Coolant System (RCS) by the secondary side of the Plant is maintained and does not challenge the Reactor Protection System. The Turbine Runback feature is not Safety Related and the deletion of the Turbine Runback does not affect any Safety Related signals required to initiate a Reactor trip. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety. For these reasons, the proposed activity does not create the possibility of an accident of a different type than previously described in the FSAR.

SAFETY EVALUATION (Continued)

6) 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 deletion of the Turbine Runback does not adversely affect Safety Related equipment or signals required to initiate a Reactor trip. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety.

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

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification, because deletion of the Turbine Runback feature has been shown by analysis not to effect the basis of the Technical Specification.

The foregoing discussions constitute, per 10 CFR 50.59(b), the written safety evaluation which provides the basis that this modification does not involve an unreviewed safety question, nor a change to the Plant Technical Specifications. As such, prior NRC approval for the implementation of this PC/M is not required.

PC/M 090-191, Supplement 0

## ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to add discharge resistors across the shunt field of 125 VDC motor operated valves (MOVs) at St. Lucie Unit 1. The addition of discharge resistors is based on recommendations which state that upon MOV de-energization, high voltage transients are created which can cause insulation degradation and reduced motor life. This concern is addressed in the NRC Information Notice 88-72.

The safety evaluation of this EP has determined that this PCM does not constitute an unreviewed safety question as defined in 10 CFR 50.59 and does not require a change in the plant Technical Specifications. This PCM has no adverse impact on plant safety or operation. Therefore, this PCM can be implemented without prior NRC approval.

This EP involves modification of Nuclear Safety Related MOVs; and is therefore classified as Nuclear Safety Related.

## SAFETY EVALUATION

This Engineering Package provides the documentation necessary to install discharge resistors on the DC motor operated valves (MOVs) located at St. Lucie Unit 1. The following MOVs are affected by this EP:

| Tag Number | MOV Description |
|---|---|
| MV-08-3 | AFWP 1C Turbine Steam Valve |
| MV-08-13 | Steam Generator 1A to AFWP1C Turbine |
| MV-08-14 | Steam Generator 1B to AFWP 1C Turbine |
| MV-09-11 | AFWP 1C Discharge to Steam Generator 1A |
| MV-09-12 | AFWP 1C Discharge to Steam Generator 1B |

Presently, high voltage transients are induced during 125 VDC MOV de-energization that can potentially cause motor insulation degradation and reduced motor life, as addressed in NRC Notice 88-72. The use of discharge resistors will protect the DC MOVs against the potential effects of voltage transients and will not adversely affect operation of the MOV nor the AFW system.

This modification installs a discharge resistor in the local DC starter boxes across the shunt field winding of each 125 VDC MOV. This circuit configuration will allow induced voltage transients to dissipate across the fixed resistance load of the discharge resistor. The addition of discharge resistors has been evaluated to have no adverse affect to the safety-related MOVs. This EP does not affect (1) designed reactor coolant pressure boundary; (2) capability to shut down and maintain the reactor in a safety shutdown condition; (3) the capability to prevent or mitigate the consequences of accidents with potential exposures comparable to 10 CFR 100 levels.

SAFETY EVALUATION (Continued)

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. The modification included in this Engineering Package does not involve an unreviewed safety question for the following reasons:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR)?

   This modification does not adversely affect any equipment whose malfunction is postulated in the SAR. This modification adds a discharge resistor to the AFW MOVs and does not change the function, nor operation of the MOVs. This modification does not circumvent the valves' safety functions. This modification does not affect MOV pressure boundaries. Therefore, the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

2) Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

   This modification does not affect the function or operation of the AFW System MOVs, nor does it affect other systems and components that are relied upon to mitigate accidents. Therefore, the consequences of an accident previously evaluated in the SAR in not increased by this modification.

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

   This modification adds discharge resistors across the shunt field winding of each 125 VDC MOV to protect against potential high voltage transients that may occur during de-energization of a MOV. However, a malfunction of the discharge resistor will result in an open circuit across the MOV shunt field winding, which in turn will be electrically equivalent to the existing MOVs. The reliability of the MOV motors will be increased by this change. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR has not increased by this modification.

SAFETY EVALUATION (Continued)

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

The addition of the discharge resistors will not alter the original safety function of the MOVs. The discharge resistors have an open circuit failure mode and do not affect the existing performance of the MOV and therefore, the implementation of this EP would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

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

This modification adds protection against high voltage transient that may cause motor insulation degradation and reduced motor life. Protection is added by installing a discharge resistor across each DC MOV shunt field winding. This change does not alter the function or operation of a MOV, nor create any new failure mode or conditions which could 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 EP.

6) 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 modification does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated. The credible failure mode of the resistor is an open circuit. Therefore, a modified MOV circuit with a failed discharge resistor is electrically identical to the existing MOV circuit. The addition of the resistor does not create a new failure mode of the modified MOVs.

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

The modification does not reduce the margin of safety as defined in the bases for any Technical Specification. Adding a discharge resistor to an MOV as per this modification enhances the reliability of the MOV. The installation of each discharge resistor will increase the battery loading approximately 0.083 amperes at 125 VDC. This increase loading occurs only during the time of MOV operation (several seconds). The 125 VDC system calculations were reviewed and determined that the small increased loading has a negligible affect on the 125 VDC System. The addition of a discharge resistor does not adversely affect the MOVs performance. The existing margin of safety as defined in the basis for any Technical Specification remains unchanged after the implementation of this modification.

PC/M 090-191, Supplement 0

SAFETY EVALUATION (Continued)

This modification does not reduce the margin of safety as defined in the basis for any Technical Specification. The implementation of this EP does not require a change to the plant Technical Specifications. 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.

PC/M 060-191, Supplement 0


ABSTRACT

This Engineering Package provides for modification to the borated water level annunciation circuitry for the 1A and 1B Boric Acid Makeup (BAM) Tanks. The modification involves the addition of a low level alarm selector switch which provides three options. The new switch allows enabling of the low level alarm circuity for the 1A Tank (1B not enabled), or enabling of the low level alarm circuitry for the 1B Tank (1A not enabled), or enabling of the low level alarm circuitry for both the 1A and the 1B Tanks.

The purpose of this modification is to eliminate undesired annunciation (i.e., nuisance alarms and continuously lit windows) in the main control room from low borated water levels in the BAM Tanks. Current plant practice during normal operation involves use of one BAM Tank in combination with the Refueling Water Tank to satisfy the borated water source requirements of Technical Specification 3.1.2.8. Operating in this mode, the other BAM Tank is not required to be operable and as a result its water level frequently drops low enough to cause the undesired annunciation. The elimination of this undesired annunciation is consistent with the NUREG 0700 "Guidelines for Control Room Design Review" "Dark Annunciator" concept. Under normal operating conditions, annunciators will not be illuminated.

The BAM Tank low level alarm circuits do not perform a Safety Related function, however, Table 7.5-2 of the St. Lucie Unit 1 FSAR incorrectly lists the BAM Tank level indicators as Safety Related. The alarm circuits receive their signals from these level indicators and provide main control room annunciation to Operations, informing them that the associated BAM Tank level is approaching Technical Specification limits. The selector switch added by this modification will be seismically mounted to prevent possible interaction with Safety Related equipment. This Engineering Package has evaluated the safety classification of the affected equipment, therefore this package is classified Safety Related.

A safety evaluation of this modification has been performed in accordance with 10 CFR 50.59. This evaluation concludes that implementation of this Engineering Package 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 for implementation of this modification is not required.

PC/M 060-191, Supplement 0

SAFETY EVALUATION

This modification provides the details to install a BAM Tank low level alarm selector switch. The BAM Tank's low level alarm provide annunciation in the Control Room (Windows N15 and N16) when the associated tank level (1A or 1B) approaches the Technical Specification 3.1.2.8 limit. This Technical Specification is satisfied by meeting at least two of four possible conditions for borated water sources. The setpoints are presently set for two BAM Tank operation, however, the Operations Department normally uses one BAM Tank (at a higher minimum water level and higher boric acid concentration), and the Refueling Water Tank (RWT) to meet Technical Specification 3.1.2.8. Since only one BAM Tank is required to be maintained above the low level alarm point, and operations uses the other BAM Tank at varying levels during normal plant operations, the low level alarm is often energized. This modification adds a selector switch to select the BAM Tank to be maintained per Technical Specification, and defeat the low level alarm of the non-selected BAM Tank. The low level alarm of the non-selected tank will no longer be annunciated, eliminating the nuisance alarm.

This evaluation addresses the acceptability of the installation of a BAM Tank low level alarm selector switch.

Table 7.5-2 of the St. Lucie Unit 1 FSAR lists the BAM Tank level indicators as Safety Related. Safety Related items are those necessary to assure one or more of the following:

1) integrity of the reactor coolant pressure boundary,

2) the capability to shut down and maintain the reactor in a safe shutdown condition, or

3) the capability to prevent or mitigate the consequences of accidents with potential exposures approaching 10 CFR Part 100 levels.

The BAM Tank level circuitry does not perform a Safety Related function since it is not necessary to assure these three criteria. The mounting of the selector switch on RTGB 105 is designed for seismic loading to prevent possible interaction with safety related equipment around it. This engineering package is classified as Safety Related, since this Safety Classification evaluation has defined the associated equipment to be of a lower classification as defined in the FSAR.

By implementing these changes, low level annunciation for the BAM Tanks will conform with the NUREG 0700 "Guidelines for Control Room Design Review" "Dark Annunciator" concept. Under normal operating conditions, annunciators are not illuminated.

The addition of the BAM Tank low level alarm selector switch does not alter the operation, nor the function of the borated water sources. This method of operation is bounded by existing Technical Specification. The licensing basis has not been altered by these modifications.

A review of the "Single Failure Analysis" for CVCS (PSL-1 FSAR, Table 9.3-25) was performed and determined that the analysis is not affected by this modification.

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. The modification included in this engineering package does not involve an unreviewed safety question because of the following reasons:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR)?

   The proposed activity does not increase the probability of occurrence of an accident because the Control Room annunciators are not accident initiating devices. The annunciators function to alert operators of abnormal plant conditions and serve no controlling function.

2) 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 Control Room annunciators are not used by the operators to mitigate an accident. Each BAM Tank has level indication available on RTGB 0105 and is used by the operators to determine the status of the BAM Tank's. The alarms do not increase the radiological doses of an accident.

3) 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 switch installed by this modification or the annunciators perform no safety related function. The switch mounting was also analyzed and found to be adequate and not affect the seismic response of RTGB 105.

4) 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 selector switch and the BAM Tank's low level alarm performs no nuclear safety related or equipment protection function. The BAM Tank's (1A and 1B) each have a low-low alarm and a high alarm. These alarms warn operators of potential abnormal conditions. The low-low alarm warns of emptying a tank which could lead to possible BAM pump damage. The high alarm warns of possible tank overflow.

5) 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 Control Room annunciators are not accident initiating devices. These annunciators warn operators of abnormal plant conditions and serve no controlling function.

6) 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 method of operation for a Control Room annunciator has not changed. The addition of the selector switch has also been analyzed to maintain the seismic integrity of the RTGB panel.

7) Does the proposed activity reduce the margin of safety as defined in the bases 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 alarm circuitry is not included in the bases for any Technical Specification. The existing Technical Specification ensures that adequate Shutdown Margin exists.

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. Prior NRC approval for the implementation of this modification is not required.

PC/M 403-190, Supplement 0

## ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to delete the Acoustic Flow Recorder (FR-1200). The inputs to the Recorder are also wired to the SAS System and both monitor and provide trend information for the Power Operated Relief Valves/Safety Relief Valves (PORVs/SRVs). The Acoustic Recorder is presently inoperable and spare parts are unavailable due to obsolescence. By implementing this (EP), this recorder will be deleted from the Post Accident Panel-1B, in the Unit 1 Control Room. A cover plate will be placed over the hole where the recorder existed. During an Emergency the Operators primary means for assessing leakage in the PORVs/SRVs is by use of the Flow Indicators (FI-1200, 1201, 1202, 1402, 1404) instead of the Acoustic Recorder (Ref. 6.10). In addition, the SAS trends may be used.

This Engineering Package is classified as Safety Related, since the Recorder is referenced in the FSAR, Page 7.5-36 and is listed on Table 7.5-2 as a safety related recorder. Regulatory Guide 1.97 response takes credit for the SAS System and the Flow Indicators. The deletion of the Acoustic Recorder will not have any impact or detrimental effects to safety equipment located in PAP-1B, nor will it adversely affect the seismic qualification of PAP-1B.

A safety evaluation of this modification has been performed in accordance with 10CFR50.59. 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.

## SAFETY EVALUATION

The Acoustic Flow Recorder (FR-1200) provides monitoring of flow through the Power Operated Relief Valves (PORVs) V1402, V1404 and Safety Relief Valves (SRVs) V1200, V1201, and V1202. The recorder receives its signals from the Acoustic monitors which sense the flow through the valves (PORVs/SRVs). Each Acoustic sensor sends a signal to its respective indicator on the PAP-1B which indicates valve position. The Safety Assessment System also has inputs from each flow transmitter, in which the SAS system records and trends each valve in the similar fashion to the Acoustic Recorder. Recorder (FR-1200) serves no Safety Related Function in achieving Safe Reactor Shutdown in the event of a Design Basis Event (DBE) and does not serve to mitigate the consequences thereof. However, since the recorder obtains 115VAC from a safety channel inside the PAP panel, it is considered Safety Related and is included in the list of Safety Related Recorders on Page 7.5-36, Table 7.5-2 in the FSAR. Therefore, this Engineering Package is classified as Safety Related.

SAFETY EVALUATION (Continued)

This Acoustic Flow Recorder provides the Operators with trend information for the Power Operated Relief Valves and Safety Relief Valves (PORVs/SRVs). The Recorder (FR-1200) will be deleted from the Control Room Post Accident Panel #1B. The SAS system provides the same information as the Acoustic Recorder. Therefore, the deletion of Acoustic Recorder has no adverse effects on the monitoring of the (PORVs/SRVs). This EP will provide the circuit and panel modifications required to permanently delete the Acoustic Flow Recorder.

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 created; 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. The modification included in this engineering package does not involve an unreviewed safety question because of the following reasons:

1) 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 while the Acoustic Flow Recorder (FR-1200) is Safety Related, it serves no controlling function. The deletion of the recorder does not affect the seismic integrity of Post Accident Panel-1B.

2) 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 due to the deletion of the Acoustic Flow Recorder (FR-1200) from Post Accident Panel-1B. The recorder is Safety Related, but serves no Safety Related function. The Emergency Operating Procedures use the Flow Indicators to assess leakage in the PORVs/SRVs.

3) 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 Acoustic Recorder performs no controlling functions and the removal of this recorder does not affect the seismic integrity of the panel (PAP-1B) nor does it affect the reliability of the Flow Indicating loops (F-1200, 1201, 1202, 1402, 1404).

SAFETY EVALUATION (Continued)

4) 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 recorder performs no Safety Related function and is not used to mitigate the effects of an accident and does not affect the seismic integrity of the panel (PAP-1B).

5) 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 recorder performs no Safety Related function and the Indicators are used during an emergency. A review of the failure mode analysis for the PORV and SRV Valve Position Indication (FSAR, Page 7.5-22 and Page 7.5-36) has been performed and it has not been impacted by this modification. In addition, the overall seismic integrity of the panel (PAP-1B) will not be degraded by this modification.

6) 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 Acoustic Flow Recorder performs no Safety Related function. The deletion of the Safety Related Recorder will not affect the reliability of the Flow Indicator loops (F-1200, 1201, 1202, 1402, 1404) or change the function of the loops.

7) 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 deleted recorder is not included in the basis of the Technical Specification for the Reactor Coolant System or any Technical Specification.

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 nor a change to the Plant Technical Specifications and prior NRC approval for the implementation of this modification is not required.

## ABSTRACT

This Engineering Package provides for the replacement of several obsolete Fischer and Porter Model 51-1401 indicating controllers installed on St Lucie Unit 1 for charging pump suction pressure and component cooling water temperature indication and control. In the current plant configuration, pressure indicating controllers PIC-2224X, PIC-2224Y and PIC-2224Z provide Charging Pump 1A, 1B and 1C local indication and pump trip on low suction pressure with no safety injection actuation signal (SIAS) present; in the event of a loss of coolant accident (LOCA), SIAS will override the low pressure trip to assure concentrated boric acid is injected into the reactor coolant system.

In the modifications provided with this EP, each charging pump indicating controller is replaced with a local indicator (pressure gauge) and pressure switch to retain the existing system design features. Since the pressure switches are interposed in Nuclear Safety Related Class 1E pump breaker control circuits, and since they are connected to ASME Class III piping, they will be qualified for Nuclear Safety Related Class 1E, Seismic Category I service. Therefore, this EP is classified as Nuclear Safety Related.

Additionally, this EP provides for the replacement of temperature indicating controllers TIC-14-4A and TIC-14-4B, which throttle temperature control valves TCV-14-4A and TCV-14-4B to regulate Intake Cooling Water flow depending upon Component Cooling Water outlet temperature from the Component Cooling Water Heat Exchanger (CCWHX) 1A and 1B, thus moderating CCW temperature. This EP replaces these obsolete controllers with new, currently available, pneumatic controllers. The replacement pneumatic controllers are qualified Seismic Category I.

The safety evaluation has shown that this EP does not constitute an unreviewed safety question nor require a change to the Technical Specification and therefore prior NRC approval is not required for implementation.

PC/M 005-190, Supplement 0

SAFETY EVALUATION

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 for modifications to the Chemical Volume and Control System, CVCS (Charging Pump suction pressure instrumentation), the Component Cooling Water (CCW) System and Intake Cooling Water (ICW) System (CCW Heat Exchanger temperature control).

The modifications have been evaluated under 10CFR50.59 and it has been determined that 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 the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased. The replacement.of the existing Fischer and Porter pressure indicating controllers with new ASCO pressure switches and Ashcroft pressure gauges and the Fischer and Porter temperature indicating controllers with Foxboro indicating controllers has no affect on the ability of the CVCS, ICW and CCW systems to perform their Nuclear Safety Related design basis function per the St Lucie - Unit 1 FSAR. Existing pressure and temperature measurement points (pressure taps and thermowells) are retained and utilized. All new equipment, including tubing, fittings and connectors, is qualified for Nuclear Safety Related service.

(ii) The possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis report is not created. The new pressure switches for Charging Pump suction pressure are Nuclear Safety Related Class 1E qualified and are sufficiently rated for their 125V dc control application. Further, the new configuration no longer requires 120V ac control power as it relies on pneumatic sensing only. Existing setpoints are retained and all new and existing equipment are compatible at all interface points. Existing channel independence and redundancy is retained after the implementation of the PC/M. The integrity and functions of systems, components and structures are either maintained or improved by the replacement equipment.

SAFETY EVALUATION (Continued)

(iii) The margin of safety as defined in the bases for any Technical Specification is not reduced since this modification maintains existing levels of protection for Nuclear Safety Related equipment (e.g., CVCS, CCW and ICW systems and components). Existing charging pump trip setpoints (10psia) and interlocks (time delays and SIAS permissives) are retained, as are CCW temperature control valves travel limiters (including the high limit relays and valve stops). CCW temperature indication and control are maintained or improved use at the Moore 58H high limit relays and Norgren Model 11-024-042 pressure regulators. These modifications maintain or enhance the existing design.

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 nor a change to the technical specifications and therefore prior Nuclear Regulatory Commission approval for the implementation of this PC/M is not required.

## ABSTRACT

This Engineering Package (EP) will provide the engineering and design details required to implement the modifications to the New Fuel Crane Hoist.

The purpose of the New Fuel Crane Hoist is to remove the new fuel assemblies from their shipping containers, place them in the new fuel storage racks and eventually to transport them to the Containment Fuel Handling Machine. This process requires the hoist operator to repeatedly cycle the motor to achieve the necessary load spotting or positioning. This "jogging" technique eventually results in excessive heat buildup at the "Magnetorque" motor to an extent that rotor warpage develops. Also, as each start cycle is initiated, inrush current delivered to the motor produces stress on the windings, resulting in potential premature failure.

The replacement of the existing wound motor and stepless controller with a squirrel cage motor and solid state "Smartorque" adjustable frequency controller will ensure the efficiency of the system during new fuel handling operations. In addition, a mechanical load brake will replace the electric load brake. The mechanical load brake is an additional means of redundancy, in case the dc magnetic load brake does not operate, to prevent the load from dropping due to failure of the electric motor. The bridge and trolley controls will not be affected by the above modifications.

The equipment being modified by this EP performs a non-nuclear safety related function. However, since the new fuel crane is seismically designed and is used in handling new fuel assemblies (and since mishandling could result in fuel damage) this EP has been classified as Quality Related.

Results of the safety evaluation conclude that modifications presented by this Engineering Package do not constitute an unreviewed safety question, do not require any changes to the Plant Technical Specifications and therefore, no prior NRC approval for the implementation of this PC/M is required.

The implementation of this PC/M will not have any impact on plant safety or operations.

## SAFETY EVALUATION

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.

SAFETY EVALUATION (Continued)

This Engineering Package covers the modifications to the St. Lucie Unit 1 New Fuel Crane Hoist. The replacement of the existing wound motor and stepless controller with a squirrel cage motor and solid state "Smartorque adjustable frequency controller will ensure the efficiency of the system during new fuel handling operations. The addition of a mechanical load brake adds extra assurance that a load cannot be dropped due to loss of power to the motor. The implementation of this EP increases the reliability of the New Fuel Crane Hoist and enhances the system operation.

The equipment being added by this EP performs no safety related function and will not interact with any safety related equipment or function. The New Fuel Crane is seismically designed and is used in handling new fuel assemblies, (and since mishandling could result in fuel damage), this EP has been classified as Quality 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 these modifications. The replacement of the existing equipment with a new motor controller and mechanical load brake will prevent: the heat buildup that now occurs at the "Magnetorque" motor, which results in motor warpage; the stress on the motor windings that will result in premature failure of the motor; and the possibility of a load being dropped due to loss of power to the motor. 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. There is no change in operation, capacity or function of the equipment due to these modifications. There is no adverse interaction with any safety related equipment or system, therefore, a failure of any safety related component which could cause, contribute to, or become a factor in a new type of accident cannot result from this modification.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The New Fuel Crane being modified by this PC/M is not addressed in the Technical Specifications.

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.

PC/M 009-189, Supplement 0

## ABSTRACT

This Engineering Package provides the engineering and design details required to replace the existing Main Fire Alarm Panels. The panels are part of the fire detection system. The existing duct mounted detectors are not compatible with the new panels and will also be replaced.

The existing panels are obsolete and spare parts are no longer available. The replacement panels represent the latest evolution in Honeywell's Fire Detection System's hardware and software. The new panels are fully compatible with the existing plant fire detection system.

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 is classified as Quality Related.

The installation of the equipment described above does not involve an unreviewed safety question, has no effect on plant safety or operation and does not require a change to the plant Technical Specifications. Therefore, prior NRC approval is not required for the implementation of this Engineering Package.

## SAFETY EVALUATION

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 Engineering Package (EP) provides the engineering and design details required to replace the existing obsolete Main Fire Alarm Panels with new updated panels. The existing duct mounted detectors will also be replaced since they are not compatible with the new panels. The new panels and detectors are fully compatible with the existing plant fire detection system.

The implementation of this EP will improve the reliability of the fire detection system, by replacing obsolete equipment. This ensures 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.

SAFETY EVALUATION (Continued)

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 by these modifications. The replacement of the obsolete fire alarm panels with new panels and the replacement of the duct mounted detectors will enhance the operation of the fire detection system. The new panels and detectors 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. Therefore, the implementation of these modifications cannot increase the probability of occurrence or the consequences of an accident or malfunction of equipment.

(ii) The possibility of an accident or malfunction of equipment of a different type than any evaluated previously is not created. The fire alarm panels and detectors are not required during an accident condition nor will they prevent safety related equipment from performing their functions. This modification does not affect any safety related equipment.

(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, 3/4.3.3.7, 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 or a change in the Technical Specifications and prior NRC approval for the implementation of this PC/M is not required.

PC/M 399-988 Supplement 1

## ABSTRACT

This Engineering Package provides the necessary details for the installation of a Fuel Dispensing Facility for the St. Lucie site. The facility will consist of the foundation and spill retainer for two tanks (one 8,000 gallon and one 10,000 gallon) as well as the fuel dispensers and necessary appurtenances.

The Fuel Dispensing Facility does not perform any Nuclear Safety-Related functions. It will be constructed at the south end of the site just west of the east basin and is not in the vicinity of any safety-related equipment or systems, nor will it impact any safety-related functions. Accordingly, this Engineering Package has been classified as Non-Nuclear Safety-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. Furthermore, the implementation of this modification does not require a change to the Plant Technical Specifications and has no detrimental effect on plant safety and operation. Therefore, prior NRC approval for implementation of this modification is not required.

Supplement 1

This supplement includes the design details for adding an electric sump pump to empty rainwater from the fuel dispensing facility spill retainer. In addition, the original design drawings will be updated to reflect as-built conditions and the PC/M Expiration Date has been extended to December 31, 1990. These changes require the original engineering design bases and design analysis to be amended, but have no effect on the original safety evaluation or the technical specifications.

## SAFETY EVALUATION

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. The modifications included in this engineering package do not involve an unreviewed safety question because of the following reasons:

SAFETY EVALUATION (Continued)

i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Updated Safety Analysis Report are not increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident. It has been shown that there will be no adverse impact on structures, systems, or components greater than 240 feet from the Fuel Dispensing Facility should an explosion occur there. Since the tanks will be located at least 700 feet from the nearest safety-related structure, system, or component and does not perform any function either directly or indirectly related to Power Plant operations, there will be no adverse impact on Nuclear Safety.

ii) The possibility of an accident or malfunction of a different type than any evaluated previously in the Final Updated Safety Analysis Report will not be created by this modification because the modification involves non-nuclear safety-related structures and failure of any items added by this modification will not impact any nuclear safety-related functions. In addition, any mishap at the Fuel Dispensing Facility including fire, explosion, and construction activities will not cause an accident or malfunction of any structure, system, or component important to Nuclear Safety.

iii) 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.

The Fuel Dispensing Facility does not perform any safety-related functions. A failure mode evaluation has been performed for this modification and it has been determined that no new failure modes have been introduced to the plant. An explosion analysis and fire analysis have been performed in accordance with the St. Lucie Unit 1 FSAR, the St. Lucie Unit 2 FSAR, and 10 CFR 50 Appendix R, and it has been concluded that there will be no adverse effect on the plant as the design of the facility meets or exceeds the requirements of these documents. Accordingly, this engineering package has been classified as non-nuclear safety-related.

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.

## ABSTRACT

This Engineering Package (EP) covers the replacement of the now obsolete Fischer and Porter transmitters with the currently manufactured equivalent Rosemount transmitters. The transmitters are providing tank level and process flow monitoring signals in the Makeup Water System and in the Steam Generator Blowdown Treatment Facility.

The existing Fischer and Porter transmitters do not provide any interface with the safety related systems, therefore, this EP is classified Non-Nuclear Safety Related. Since this modification is a one-for-one replacement of the existing Fischer and Porter transmitters with the equivalent Rosemount transmitters, the same classification applies.

The safety evaluation of this EP does not involve an unreviewed safety question, and does not require a change in the Plant Technical Specifications. This EP does not impact plant safety and operation, therefore, NRC approval for these modifications, prior to their implementation, is not required.

SUPPLEMENT 1

Supplement 1 of this EP has been issued to reflect current Rosemount model numbers for transmitters LT-36-1 and LT-36-3. The model number for these devices (as purchased) is 1151LT4EB0B22D and is included in this package with this revision.

This supplement does not effect the safety evaluation; the implementation of this PCM does not affect the Plant Technical Specifications and does not constitute an unreviewed safety question.

This EP has no impact on plant safety or operation.

## SAFETY EVALUATION

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.

The modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

SAFETY EVALUATION (Continued)

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Updated Safety Analysis Report, Chapters 8 & 9, are not increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Updated Safety Analysis Report, Chapters 8 & 9, will not be created by this modification because the function of the transmitters has not been altered.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since the new transmitters are all classified non-nuclear safety related and do not affect any technical specification.

The existing Fischer & Porter transmitters do not provide any interface with the safety related systems, therefore, this EP is classified Non-Nuclear Safety Related.

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 nor require a change to the technical specifications and therefore prior Commission approval for the implementation of this PCM is not required.

PC/M 020-187, Supplement 0-2

## ABSTRACT

This Engineering Package provides for the replacement of the Containment Level Monitoring System on St. Lucie - Unit 1 with instrumentation possessing impaired performance characteristics. The current system consists of one channel of narrow range level monitoring (from -6'-7 to 0'-0) and two redundant channels of wide range level monitoring (from -1'-0 to 26'-0). The replacement IMO Delaval level sensors utilize magnetic reed switches in a variable resistance network, providing improved accuracy and repeatability while minimizing drift. The new IMO Delaval sensors provide one channel of narrow range monitoring (-6"8 to 0'-0) and two redundant channels of wide range monitoring (from -1'-0 to 16'-1). Additionally, the new equipment (including Control Room mounted modular transmitters/receivers) will provide ease of calibration, reducing stay-time and minimizing man-rem exposure in the containment sump area.

The new instrumentation is qualified for Nuclear Safety Related Class 1E service and is environmentally qualified per 10CFR50.49. Existing plant commitments to Regulatory Guide 1.97 (Post Accident Monitoring) have been reviewed and are addressed in this Engineering Package. Since this PC/M modifies equipment required for post-accident monitoring which is qualified for Nuclear Safety Related use, this PC/M is designated Nuclear Safety Related.

The safety evaluation of this package has shown that the implementation of this PC/M does not constitute an Unreviewed Safety Question and requires no revision to the Unit 1 Technical Specifications, therefore, prior NRC approval is not required for implementation. This PC/M has no impact on plant safety and operation, or the Plant Technical Specifications.

<u>SUPPLEMENT 1</u>

Supplement 1 of this EP was issued to remove holdpoints on implementation and to issue conduit support details and associated calculations.

<u>SUPPLEMENT 2</u>

Supplement 2 of this EP is issued to remove holdpoints on system start-up/operation. All pertinent qualification documentation for the new Containment Level Monitoring System has been received and is issued with this EP supplement.

The safety evaluation of this package shows that the implementation of this PC/M does not constitute an Unreviewed Safety Question and requires no revision to the Unit 1 Technical Specifications, therefore, prior NRC approval is not required for implementation. This PC/M has no impact on plant safety and operation, or the Plant Technical Specifications.

PC/M 020-187, Supplement 0-2

SAFETY EVALUATION

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 consequence 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.

The modifications have been evaluated under 10CFR50.59 and it has been determined that the modifications included in this Engineering Package do not involve an unreviewed safety question as demonstrated by the answer to the questions below:

(i) Does the Proposed Activity Increase the Probability of Occurrence of an Accident Previously Evaluated in the Safety Analysis Report?

This modification does not increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report. The new containment level monitoring instrumentation installed with this PC/M has been qualified for Nuclear Safety Related Class 1E service in order to satisfy Regulatory Guide 1.97 and existing plant design requirements. This design utilizes physical and electrical separation for redundant channels and provides new equipment (GEMS level sensors) which has been qualified for harsh environment in accordance with 10CFR50.49.

The new instrumentation (including level sensors and Control Room mounted modular transmitter/receivers) have been specified to assure that system electrical load is within the limits of the existing 120 VAC power supplies utilized. Fuses are provided on the existing 120 VAC power feeds to preclude the propagation of a fault of this equipment from affecting other safety circuits. Power conditioning, including voltage step and rectifying, is provided internally at the modular transmitter/receivers in the Control Room. No other operating equipment (pumps, motors, etc) is affected by this modification, either through direct control or interlock. The Containment Level Monitoring System instrumentation is used for monitoring only and does not use any setpoints for alarm or control functions. No other system interfaces are involved with this PC/M.

(ii) Does the Proposed Activity Increase the Consequences of an Accident Previously Evaluated in the Safety Analysis Report?

SAFETY EVALUATION (Continued)

The consequences of an accident previously evaluated in the Safety Analysis Report are not increased with the implementation of this PC/M. The replacement of the existing Containment Level Monitoring instrumentation with the new GEMS sensors does not change, degrade or prevent actions described or assumed in accidents described in the Safety Analysis Report. In accordance with St. Lucie - Unit 1 Emergency Operating Procedures, subsequent to a Loss of Coolant Accident (LOCA), plant operations is required to continually monitor containment level to verify proper transfer of water from the Refueling Water Tank (RWT) to the containment. Variance of rate of change of RWT level against containment level outside prescribed limits may indicate leakage (release of fission product) outside containment. The new GEMS instrumentation installed with this PC/M satisfies the criterion that the relationship between RWT and containment levels be monitored post-LOCA. As stated above, improved system accuracy will provide additional confidence that indicated level is true; extrapolation of data will be more precise.

The new instrumentation has been demonstrated by test to be accurate to ± 3% of indicated range post-LOCA, which is an improvement over the ± 5% accuracy (optimum conditions) provided by the existing Barton transmitters.

New mounting hardware in the containment building, as well as the new sensors themselves, are stainless steel composition and do not introduce any new halogens or hydrogen generating material into the containment. Existing electrical penetrations are utilized in this design such that the implementation of this PC/M will not require an additional penetration be installed in containment.

No plant structure, system or component used in mitigating the radiological consequences of an accident described in the Safety Analysis Report is affected by this modification. The replacement of the existing ITT Barton instrumentation with the new GEMS sensors has no effect on the ability of any other safety related structure, system or component to operate within its specified design limits.

(iii) Does the Proposed Activity Increase the Probability of Occurrence of a Malfunction of Equipment Important to Safety Previously Evaluated in the Safety Analysis Report?

SAFETY EVALUATION (Continued)

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased with the implementation of this PC/M. This modification has been designed to satisfy all applicable Nuclear Safety-Related Class 1E, Seismic Category I criteria. Additionally, the new sensors (installed in the Containment Building) are qualified for harsh environment in accordance with 10CFR50.49. These features provide assurance that the Containment Level Monitoring System will operate as required in providing Regulatory Guide 1.97 (post-accident monitoring instrumentation) information to plant operations during and after LOCA.

The new IMO Delaval (GEMS) level sensors have improved reliability over the existing ITT Barton sealed sensor level transmitters. The existing instrumentation is more subject to drift and less accurate than the replacement instrumentation. Therefore, the probability of malfunction of the Containment Level Monitoring System will be reduced by this modification.

This modification has no adverse effect on any safety related system, structure or component. Existing embedded plates on the biological shield wall and in the reactor sump are utilized for the mounting of the new instrument rack assemblies, and have been analyzed for all applicable loads. The new rack assemblies (including mounting of GEMS instruments, protective cages and wireway) have been seismically designed. New conduit supports have been provided in these areas and designed for all applicable loads. At the electrical penetration assemblies (EPAs), existing feed-throughs are utilized with neither adverse effect on the EPAs themselves nor on the containment vessel or integrity thereof. Existing electrical raceway between the Control Room and the Penetration Room in the Reactor Auxiliary Building is retained and utilized without affect on the raceway or supports. In the Control Room, the new modular transmitter/receivers are installed in the Control Room Auxiliary Console (CRAC). The component mountings have been seismically designed and analyzed; the seismic qualification of the console has been reviewed and is not affected by this modification. Existing panel mounted meters and power supplies are retained and utilized and are compatible with the new system (120 VAC input power, 4-20 mA DC signal current).

As the new design addresses and satisfies applicable Nuclear Safety Related Class 1E criteria as stated above, and does not increase challenges to any nuclear safety related structure, system or component, the probability of occurrence of a malfunction of Nuclear Safety Related equipment is not increased with the implementation of this PC/M.

SAFETY EVALUATION (Continued)

(iv) Does the Proposed Activity Increase the Consequences of a Malfunction of Equipment Important to Safety Previously Evaluated in the Safety Analysis Report?

This modification does not increase the consequences of a malfunction of Nuclear Safety Related equipment previously evaluated in the Safety Analysis Report. As discussed above, the Containment Level Monitoring System provides system status during normal and post-LOCA conditions. This system (including the new instrumentation installed with this PC/M) is utilized to indicate proper transfer of emergency core cooling water from the Refueling Water Tank (RWT) to the containment vessel by monitoring the change in containment level as it varies inversely with decreasing RWT level.

The new IMO Delaval (GEMS) sensors provide essentially the same continuous range of indication of containment level as do the existing Barton instrumentation, via one narrow range and two redundant wide range channels of indication. The only differences are that the GEMS narrow range (LE-07-14A) measures from minus 6'-8" instead of minus 6'-7", and the upper limit of the wide range (LE-07-13A4 and LE-07-13B4) is 16'-1" rather than 26'-0". An overall increase in range of indication with improved accuracy across that range (as compared with the existing Barton instrumentation) is provided with this modification. This PC/M does not have any effect on any emergency operating procedures and does not modify any required operator actions per plant procedures. By maintaining existing levels of post-accident monitoring instrumentation (albeit with increased indicating range and improved accuracy), the consequences of a malfunction of Nuclear Safety Related equipment previously evaluated in the Safety Analysis Report are not increased with the implementation of this PC/M.

(v) Does the Proposed Activity Create the Possibility of an Accident of a Different Type than any Previously Evaluated in the Safety Analysis Report?

This modification does not create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report. This PC/M modifies equipment required to monitor and respond to a Class 2 accident, which may lead to a breach of barriers and fission product releases. The loss of all containment level monitoring instrumentation could prevent mitigative action in the event of leakage outside the Containment. The new instrumentation serves the same function as the ITT Barton instrumentation which it replaces. Plant criteria for redundancy and separation are maintained with the new design, which is more accurate than the existing equipment, and plant commitments for post-accident monitoring per Regulatory Guide 1.97 are satisfied. Additionally, existing plant systems utilized to verify and/or isolate the accident scenario discussed above (e.g. Containment Isolation Actuation System, CIAS, Area Radiation Monitoring System, ARMS, and RAB sump level alarms) are not affected by this modification. No other accident scenarios evaluated in the Safety Analysis Report are affected by this modification and no other types of credible accidents can be created by the implementation of this PC/M.

(vi) 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 create the possibility of a malfunction of nuclear safety related equipment of a different type than any previously evaluated in the Safety Analysis Report. This PC/M replaces existing pneumatic level sensors with new electro-magnetic sensors which will be subjected to submergence in the post-LOCA environment. Accordingly, the new GEMS sensors have been environmentally tested and qualified for submergence. Although the new system utilizes electronics instead of pneumatics in a submerged state, no new failure modes are postulated due to this new design feature. Additionally, the new sensors monitor level over the full flood range (to 26'-1"). Steel cages have been constructed to protect the float stem assembly against damage from impact (personnel or equipment/debris).

The new GEMS sensors utilize different materials than those contained in the existing Barton Sensors. As part of the environmental qualification of the new equipment, these materials have been tested to assure their proper operation over the required design life of the plant. This provides assurance that the equipment will not malfunction due to material breakdown resulting from the post-LOCA environment. Outside containment, there are no harsh environment concerns to cause equipment failure due to age related mechanisms. There are no malfunctions of a different type than previously evaluated in the SAR which could be created by the implementation of this PC/M.

(vii) 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 as defined in the basis for any technical specification at St. Lucie - Unit 1. This modification maintains the existing design makeup of one narrow range and two redundant and electrically and physically independent channels of containment level. Technical Specification requirements to maintain all channels operable, with required action in the event of any channel failure are maintained and satisfied with this PC/M. Technical Specifications requirements for monthly channel check and channel calibration during each refueling outage are not affected by this PC/M. There are no controls, interfaces or interlocks associated with the Containment Level Monitoring System. The margin of safety for the Containment Level Monitoring System is not affected by this PC/M.

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 nor require a change to Technical Specifications and therefore prior Nuclear Regulatory Commission approval for the implementation of this PC/M is not required.

PC/M 142-186, Supplement 0-2


ABSTRACT

This Engineering Package (EP) details the requirements for modifications related to the replacement of the spent fuel storage racks, accomplished under PC/M 018-186. The tasks to be covered by this EP are as follows:

1. Modification of the existing spent fuel cooling system sparger pipe on the east side of the spent fuel pool to eliminate an interference with the new spent fuel storage racks.

2. Modifications to the hook limits of the existing spent fuel handling machine to permit the machine to access the outermost storage cells of the new racks.

3. Modifications to the existing work platform on the north side of the spent fuel storage pool to permit this platform to be moveable to eliminate interferences with the installation of the new racks and the future movement of spent fuel into the storage cells in this area.

4. Replacement of the existing underwater lighting fixtures with new low-profile portable fixtures.

5. Provision of suitable areas for receipt inspection and leveling of the new racks and laydown of the existing racks after removal from the Fuel Handling Building.

Modifications 1 through 4 are classified Safety Related because they will involve interfaces with the spent fuel pool and the spent fuel storage racks, which are Safety Related, and a modification to Safety-Related piping. Modification 5 is classified Non-Nuclear Safety Related because it involves no interfaces with any safety-related systems or equipment.

These modifications have been evaluated in accordance with 10CFR50.59. The safety evaluation has shown that the implementation of this Engineering Package does not constitute an unreviewed safety question nor are Technical Specifications affected; and, therefore, prior Commission approval for its implementation is not required. This modification will have no effect on plant safety and will facilitate plant operation.

<u>Supplement 1</u>

This supplement adds the following modifications to facilitate the rack replacement:

1. Additional modifications to the spent fuel handling machine, to physically extend the rails for the trolley and for the bridge and thus further expand the travel limits of the machine. Some of the handrail on the bridge will be made removable to avoid interference with items on the north wall of the room.

2. Further modification of the work platform, to make its supports removable so that it can, alternatively, be stored in an elevated position.

ABSTRACT (Continued)

3. Modification of the spent fuel pool purification suction piping, skimmer suction connection, and ion exchange discharge piping to eliminate interference with the new racks.

4. Modification of the service air piping on the north wall of the room to eliminate interference with the fuel handling machine.

This Supplement retains the Safety Related classification originally designated. Modification of the service air piping has been classified as Non-Safety Related since the piping is Non-Nuclear Safety Related. Modifications to the purification and ion exchanger piping and the service air piping have been classified as Quality Related due to their location in the vicinity of the spent fuel racks, although the systems themselves are Non-Nuclear Safety Related. These additional modifications do not introduce an unreviewed safety question as defined in 10 CFR 50.59, nor do they alter any Technical Specifications. Consequently, prior Commission approval is not required for the implementation of this package.

Supplement 2

This supplement replaces the originally designed work platform with one of a different design, which can be swung horizontally out of the way when required during refueling operations. The configuration of the new platform necessitates the relocation of pool temperature and level instrumentation. This modification is classified Quality Related because of potential interaction with the safety related spent fuel racks. This supplement also removes a holdpoint related to vendor information on the trolley bumper relocation which was unavailable at the time of the previous supplement. These additional modifications do not introduce an unreviewed safety question as defined in 10CFR 50.59, nor do they alter any Technical Specification. Consequently, prior NRC approval is not required for the implementation of this package.

SAFETY EVALUATION

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 basis for any technical specification is reduced.

This Engineering Package provides for the installation of modifications related to the replacement of the spent fuel storage racks. It has been classified Safety Related because it involves modifications to and interfaces with Safety Related Systems. It does not involve an unreviewed safety question. The following are the bases for this conclusion:

SAFETY EVALUATION (Continued)

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased for the following reasons:

a. The sparger pipe has no bearing on the probability of occurrence of previously evaluated accidents. The sparger pipe in its modified configuration still conforms to its original design criteria. The thermal-hydraulic effects of the modification upon the spent fuel cooling system and the spent fuel racks will be evaluated as part of PCM 018-186. The modification of the sparger pipe will, therefore, not increase the consequences of previously evaluated accidents. The potential for dropping parts of the pipe during construction is discussed in (ii).

b. The modification of the spent fuel handling machine hook limits will enable the machine to access areas at the periphery of the pool which were formerly inaccessible. The hook limits are not a factor in any previously evaluated accidents. This Engineering Package does not require modifications to the load weight system and the system will be demonstrated operable within seven (7) days prior to the crane use and at least once per seven (7) days thereafter during crane operation, in accordance with the Technical Specification, Section 4.9.7. The machine will not handle fuel or any loads in excess of 2000 lbs during the implementation of the modification and therefore will not increase the probability or consequences of previously evaluated accidents. The potential for dropping removable sections of the handrail is discussed in (ii).

c. The work platform which is being modified by this PCM is not a factor in any previously evaluated accidents. The potential for dropping the platform during construction is discussed in (ii).

d. The underwater lighting fixtures which are being modified by this PCM are not a factor in any previously evaluated accidents. The potential for dropping the fixtures during construction is discussed in (ii).

e. The new storage and inspection areas which are being provided by this PCM have no interface with any safety-related equipment or systems and hence are not a factor in any previously evaluated accidents.

SAFETY EVALUATION (Continued)

f. The functions of the spent fuel pool level switch and temperature element have no bearing on the probability of occurrence of previously evaluated accidents. The operability of these components will be unaffected by their relocation, hence the modification will not increase the consequences of previously evaluated accidents.

g. The spent fuel pool purification piping, ion exchanger piping, and the service air piping modified in this PCM are not factors in any previously evaluated accidents. The potential for dropping parts of the pipe is discussed in (ii).

h. The new work platform being installed by this PCM is not a factor in any previously evaluated accidents. The potential for dropping portions of the platform during construction is discussed in (ii)h.

(ii) There is no possibility for an accident or malfunction of a different type than any evaluated previously for the following reasons:

a. The heaviest portion of the sparger pipe that will be moved out of the pool during construction weighs less than a fuel assembly. Any possible load drop will, therefore, be enveloped by the previously evaluated fuel assembly drop accident.

b. The modification to the spent fuel handling machine hook limits requires that the crane and trolley limit switches be temporarily disabled during implementation. Plant Administrative Procedure AP 10438 will preclude the transfer of loads in excess of 2000 lbs over spent fuel. No fuel will be handled by the machine during this time. Any possible load drop will, therefore, be enveloped by the previously evaluated fuel assembly drop accident. The possibility of dropping the handrails, which weigh no more than 20 pounds, is similarly enveloped.

c. The modification of the work platform makes its supports pinned and removable. Although the design provides for seismic mounting of the platform in its storage and elevated positions, there is the possibility of dropping the platform while it is being moved. Also, the modification to the work platform may require the platform to be removed from the Fuel Handling Building. Any possible load drop as a result of these operations will be enveloped by the previously evaluated fuel assembly drop accident, since the weight of the platform before or after the modification does not exceed 2000 pounds. This modification does not violate Technical Specification 3.9.7.

SAFETY EVALUATION (Continued)

d. The modification to the underwater lighting fixtures will require the existing fixtures to be removed from the spent fuel pool. Neither the new nor the existing fixtures weighs more than a fuel assembly; therefore, any possible load drop will be enveloped by the previously evaluated fuel assembly drop accident.

e. The new storage and inspection areas have no safety related function, nor is there any possibility for their interaction with any safety related equipment or systems, therefore no possibility for a new type of accident is created.

(ii) f. No possibility for a new type of accident is created by the relocation of the spent fuel pool level switch and temperature element since their operability is unaffected. A drop of either of these components in the pool would be enveloped by the previously evaluated fuel assembly drop accident.

g. The possibility of dropping a portion of the spent fuel pool purification piping, ion exchanger piping, or the service air piping is enveloped by the previously evaluated fuel assembly drop accident as considered for several items above.

h. There is a possibility of dropping portions of the new work platform into the spent fuel pool during its installation. Any such drop will be enveloped by the previously evaluated fuel assembly drop accident, since the weight of any component which could be dropped is less than 2000 pounds.

The design of the modifications implemented by this Engineering Package ensures that the modified components will have no interaction with safety-related equipment, systems, or structures, except those interactions which are specifically addressed in the design; all potential construction accidents have been determined to be enveloped by previously evaluated accidents. Therefore, no possibility for a new type of accident exists.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification. The limitations set forth therein will be strictly maintained with the implementation of this Engineering Package, since the design intent, limitations, and constraints of the spent fuel pool, the spent fuel pool cooling system, and the spent fuel handling system remain unchanged.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question nor require a change to technical specifications and therefore prior Commission approval for the implementation of this Engineering Package is not required.

PC/M 045-986 Supplement 5

## ABSTRACT

This Plant Change/Modification includes the installation of a perimeter security barrier (intake canal crossing), intrusion detection system (perimeter and underwater), surveillance system (closed circuit television), security lighting and communications (paging).

This PCM is not classified as Safety Related since the canal crossing structures and components of the intrusion detection, surveillance, lighting and communication systems do not perform any safety function, and are located away from, and have no effect on, any safety related components. However, this PCM shall be considered quality related and quality related design requirements shall apply because of the following reasons: a) the perimeter security barrier closes the existing gap in the security perimeter fence as required by 10CFR73.55, b) the nature of the bridge and adjacent walls construction requires QC surveillance to assure proper installation of the concrete piles and correct use of concrete materials and mixes, and c) QC inspection/testing of the security system installation is required to assure proper operation and integration with the existing security system.

This PCM does not constitute an unreviewed safety question and enhances the existing plant security system. The installation of the items described above have no impact on plant operation and do not affect any safety related equipment.

SUPPLEMENT 4

This supplement incorporates the details required to install the second sonar system, monostatic microwave and their associated cables and conduits at the Intake Canal south crossing. The original safety evaluation is not affected by the modifications detailed in this supplement.

SUPPLEMENT 5

This supplement incorporates the details required to install a new intrusion detection system (perimeter and underwater) and to partially remove the existing system (perimeter and underwater) at the Intake Canal south crossing. The original safety evaluation is not affected by the modifications detailed in this supplement.

## SAFETY EVALUATION

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 report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

SAFETY EVALUATION (Continued)

This modification is not classified as Safety Related since the underwater intrusion system (canal crossing and associated systems, such as additional lights, paging stations, fencing and security system hardware) does not perform any safety function, and is located away from, and has no effect on any safety related components. The Ultimate Heat Sink analysis described in FSAR Section 9.2.7 is not affected by this modification since the failure of this crossing during a seismic event will not impede the flow of water, nor limit the intake canal and intake structure bay area from providing the plant with the primary source of shutdown cooling water capacity to dissipate reactor decay heat during normal and emergency shutdown conditions. This modification is on the intake canal area only, therefore the secondary source of cooling water (Big Mud Creek) is not affected.

The modifications included in this PCM do not involve any unreviewed safety questions because:

i The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the underwater intrusion detection system shall be installed in accordance with the quality related requirements, and this modification will have no effect on equipment performing a safety function.

ii There is no possibility for an accident or malfunction of a different type than any previously evaluated since the underwater intrusion detection system has no safety function, no changes have been made to any operational design, and the addition of security system hardware (TV cameras for surveillance, microwave 300B, fence protection FPS II, and a barrier net (Safenet) system) enhances the plant security system by implementing permanent security in the area of the intake canal crossing and integrating these modifications into the existing system. A canal crossing failure during a seismic event will not provide blockage of the primary water source to the intake cooling water system.

This canal crossing is not seismically designed and is a multi-span type bridge. Should a seismic event occur the canal crossing spans may individually collapse since there are construction joints separating each span. Based on April 1986 soil samples, the canal bottom contains firm soil at approximately elevation -26 ft. A collapse of this structure during a seismic event could hypothetically tip the 16 ft wide bridge walkway and drive it on edge into the canal, thereby leaving the top edge of the walkway no higher than elevation -10 ft at the center of the canal.

SAFETY EVALUATION (Continued)

This hypothetical scenario would still provide constant flow of water to the intake cooling water system since the lowest water elevation in the canal (with two units operating) is elevation -9 ft and the cooling water would still continue to flow over the collapsed canal crossing. In addition to water flowing over the collapsed sections, water can flow in between each collapsed section of the bridge, thereby not blocking the primary water source to the intake cooling water system.

Since the intake structure is provided with a method to prevent floating debris from entering the intake cooling water system, any possible floating debris from the canal crossing would not clog the intake cooling water system. In addition, based on engineering judgement the components associated with this modification would sink to the bottom if a seismic event were to hypothetically collapse this structure, therefore no floating debris could float downstream to clog the intake cooling water system.

iii This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PCM does not require a change to the plant technical specification.

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, therefore prior Commission approval is not required for implementation of this PCM.

FPL

**Inter-Office Correspondence**

JPN-PSLP-92-0623

To: S. A. Valdes
St. Lucie Plant

Date: JUL 7 1992

From: T. E. Roberts
Project Engineering Manager

Department: JPN/JB

Subject: **ST. LUCIE UNIT 1**
**10CFR50.59 SUMMARY OF PLANT CHANGES**
**FILE: SPEG 91-021-10**

Federal Regulations require that FPL issue a report containing a summary of each 10CFR50.59 plant change implemented for St. Lucie Unit 1.

Attached is the submittal containing summaries of safety evaluations for those PC/Ms requiring a 10CFR50.59 review based on changes made to the facility from January 23, 1991 to January 22, 1992.

If there are any questions, please contact Sergio Verduci at (407)225-9410.

RWW/SAV/bjs

Attachments

Copies: G. J. Boissy (MGM/PSL)
J. J. Breen (LIC/PSL)
[redacted]
H. R. Gavankar (PEG/SR)
M. R. Gordon (PEG/SR)
D. H. West (TS/PSL)
D. M. Wolf (ENG/PSL)

RE: St. Lucie Plant
Docket No. 50-335
10CFR50.59 Report

St. Lucie Plant Unit 1
Report of Changes Made To The Facility
Under the Provisions of
10CFR50.59
for the period January 23, 1991
to January 22, 1992

NOTE: The safety evaluations in this report are chronologically arranged starting with those created 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
10CFR50.59 for St. Lucie Unit 1

| Number | Supplement | Title |
|---|---|---|
| 439-191 | 0 | Intake Cooling Water System Flow Enhancement. |
| 252-191 | 0 | Containment Spray Vent Valve Installation. |
| 216-191 | 0-1 | Steam Generator Tube Stabilization with Flexible Stakes. |
| 160-191 | 0 | Removal of Turbine Runback. |
| 090-191 | 0 | Motor Operated Valve Arc Suppression Varistors. |
| 060-191 | 0 | BAM Tank Low Level Alarm Selector Switch Installation. |
| 403-190 | 0 | Removal of Acoustic Flow Monitor Recorder. |
| 005-190 | 0 | Fisher & Porter Indicating Controllers Replacement. |
| 312-189 | 0 | New Fuel Crane Hoist. |
| 009-189 | 0 | Replacement of Honeywell Fire Detection Panels. |
| 399-988 | 1 | Fuel Dispensing Facility. |
| 073-987 | 0-1 | Fisher & Porter Transmitter Replacement. |
| 020-187 | 0-2 | Replacement of Containment Level Monitoring System. |
| 142-186 | 0-2 | Spent Fuel Pool Rerack Platform Modification. |
| 045-986 | 5 | Installation of a Perimeter Security Barrier, Intrusion Detection and Surveillance System. |

## ABSTRACT

This Engineering Package (EP) provides the engineering to remove the 45° open mechanical stop on Temperature Control Valves I-TCV-14-4A and I-TCV-14-4B and replace orifices I-SO-21-1A and I-SO-21-1B. The purpose of these modifications is to increase the flow capacity of the Intake Cooling Water (ICW) system during accident system alignment, and to enhance the capability of the system to allow increased pressure drop through the strainer and heat exchanger.

This modification will allow the ICW system to pass 14,500 gpm of seawater with an inlet temperature of 95°F considering increased heat exchanger and strainer differential pressure drops. There will be no impact on normal operation since the temperature control valves downstream of the Turbine Cooling Water (TCW), Open Blowdown Cooling Water (OBCW) and Component Cooling Water (CCW) heat exchanger will still modulate flow as required.

The modifications considered in this EP affect the ICW system which is a safety related system. The ICW system is classified as quality Group C and Seismic Category I. Therefore, this modification is classified as Safety Related. The safety evaluation provided in Section 3.0 has shown that this EP does not constitute an unreviewed safety question. Implementation of this EP will have no adverse impact on plant safety or operation, does not require a change to the Plant Technical Specifications and does not reduce the margin of safety for any Technical Specification. Therefore, prior NRC approval is not required for implementation of this EP.

## SAFETY EVALUATION

The proposed change involves the removal of the 45° mechanical stop from valves I-TCV-14-4A and -4B, the recalibration of the valve controller to match the 90° stroke and the replacement of restriction orifices I-SO-21-1A and -1B with orifices having a larger flow area. The proposed change will enhance the flow passing capacity of the ICW system. This change is being made to accommodate for higher heat exchanger pressure drops and will enhance the heat removal capacity of the system in the accident configuration.

The change affects both the ICW and CCW systems. Both these systems are classified as Safety Related by FSAR Subsection 9.2. The effect on the ICW system will be to allow more flow to be passed in the safety related portion of the system. The safety related function of the ICW system is to remove the post accident heat load from the CCW system. During shutdown modes, post accident heat loads are lower than during power operation. The change will not affect the ability of the ICW system to remove post accident heat loads since the change enhances the heat removal capacity of ICW by increasing its flow. The ICW component most directly affected by this change is the ICW pump. Calculations (Attachment 7.4) show the ICW pump will still be operating within its tested performance. Therefore there will be no effect on the operability of the ICW pump.

The safety related function of the CCW system is to remove the post accident heat load from containment and reactor via the containment spray system and reactor containment fan coolers. The effect of this change will be to enhance heat removal from the CCW system. Thus the CCW system will meet its design basis requirements.

The increased velocity in the tubes of the CCW heat exchanger is the only other effect. This is not expected to be a concern since design basis calculation (Attachment 7.4) shows the flow with the TCV wide open would not exceed design limits.

The proposed change has no effect on the Technical Specifications. Technical Specification 3.7.4.1 requires the ICW system to be operable. This modification in no way changes the operability of the ICW system. The bases for 3.7.4.1 states that the ICW system is required to provide sufficient cooling. This modification supports this bases and enhances it by providing increased cooling capacity due to higher flow rates.

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.

The modifications have been evaluated under 10CFR50.59 and it has been determined that the modifications included in this Engineering Package do not involve an unreviewed safety question as demonstrated by the answer to the questions below:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report?

   This modification does not increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report. The Temperature Control Valves and Orifice modified by this EP are not considered in the initiation of any accident and do not affect any equipment considered as accident initiating components.

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

   The consequences of an accident previously evaluated in the Safety Analysis Report are not increased with the implementation of this PCM. After implementation of this PCM, the flow capacity of the ICW system in the accident alignment will increase. The system remains capable of delivering the minimum flow requirements for accident conditions. All components retain their functions and capabilities with the increased flow.

SAFETY EVALUATION (Continued)

3) 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 system flow capacity has been enhanced and all components remain capable of operating with the increased flow. The ICW pump flow remains within the tested range. The proposed modification does not increase the severity or possibility of pressure surges on startup of the ICW pumps. A review of the surge pressure study shows that the transient pressure surge is influenced primarily by the minimum position of the temperature control valves and the position of vacuum breakers within the system. This modification does not impact either of these design features.

4) Does the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report?

This modification does not increase the consequences of a malfunction of Nuclear Safety Related equipment previously evaluated in the Safety Analysis Report. The ICW system remains capable of performing its safety functions with the increased flow. No new failure modes are introduced by this modification. As stated above the consequences of a transient pressure surge have not been changed.

5) Does the proposed activity create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report?

This modification does not create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report. The ICW system remains within its capacity to function for safety related functions and no new failures are created. The ICW system, before and after the modification, is not considered as initiating any accident.

6) 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 create the possibility of a malfunction of nuclear safety related equipment of a different type than any previously evaluated in the Safety Analysis Report. As stated previously no new failure modes are introduced via this modification and all failures analyzed in the FSAR for the ICW system remain unchanged.

SAFETY EVALUATION (Continued)

7) 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 as defined in the basis for any technical specification at St Lucie - Unit 1. The only margin of safety that could be impacted is the bases for Technical Specification 3.7.4.1. which requires sufficient capacity from the ICW system to cool vital equipment. This modification, by increasing the flow capacity of the system, enhances this margin of safety by providing greater heat sink capabilities during accident conditions.

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 nor impact the Technical Specifications and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required.

PC/M 252-191, Supplement 0

## ABSTRACT

Engineering Evaluation (EE) JPN-PSL-SEMP-91-029, Rev. 0," Engineering Evaluation of Shutdown Cooling System Transient Response", states air in the Containment Spray (CS) header is causing pressure transients in the Shutdown Cooling (SDC) piping when the Low Pressure Safety Injection (LPSI) pumps are operated. As shown in various design documents, the CS header has no means of being vented. The EE recommends that vent valves be installed on the CS header upstream of the containment isolation valves. As valves 1-FCV-07-1A and 1-FCV-07-1B are normally closed and isolate the containment, the above mentioned vents need to be located at high points of the headers upstream of these isolation valves.

This Engineering Package (EP) provides the specific design information necessary to install one 3/4" vent in each CS header immediately up stream of valves 1-FCV-07-1A and 1-FCV-07-1B.

The CS system performs a safety related function, as described in FSAR, Section 6.2.2. As such, this EP has been classified as Safety Related. This EP does not have any adverse impact on plant safety and/or operation. Based on a Failure Mode and Effects Analysis and a review of the changes to be implemented by this EP against the requirements of 10CFR50.59, 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 modification is not required.

## SAFETY EVALUATION

With respect to Title 10 of the Codes of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety questions: (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 Final Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced. The modification here in does not involve an unreviewed safety question because of the following reasons:

1) 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 previously evaluated because the operation and functionality of the CS System has not been changed by this modification. The vents are not used during power operation or involved in any way with any safety function of this system. Based on this, the probability of occurrence of an analyzed accident remains unchanged.

SAFETY EVALUATION (Continued)

2) 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 these vents meet all regulatory requirements specified in the FSAR. Their operating and pressure retaining characteristics were shown to be acceptable. In all operational modes they are a passive pressure boundary component. The vents do not increase the radiological doses of an accident.

3) 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 this system's function and performance remain unchanged by the addition of these vents. Furthermore these vents do not directly or indirectly affect equipment important to safety. These vents do not degrade the reliability or increase challanges, directly or indirectly, for equipment important to safety.

4) 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 vents perform no active nuclear safety related or equipment protection function. The vent valves are administratively controlled to prevent their being left open and are designed to be physically strong enough to preclude being broken off. In the extreme unlikely event that the vent fails, the worse case scenario is the loss of one of the two redundant headers, which has been analyzed in the FSAR. Therefore, their operability will have no affect on the consequences of a malfunction of equipment important to safety.

5) 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 vents are not accident initiating devices. The vent valves are only operated when filling the SDC system and serve no active or controlling function.

SAFETY EVALUATION (Continued)

6) 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 method of operation of the CS system has not changed. The addition of the vents has been analyzed to maintain the seismic and pressure boundary integrity of the CS headers.

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

   The proposed activity does not reduce the margin of safety as defined in the bases for any Technical Specification because T.S. Section 3/4.6.2 requires that both loops of the CS be operable, all valves in the CS System be properly aligned and that the pumps and initiation system be tested. The Technical Specification bases ensures that adequate containment cooling and depressurization capability exists during a LOCA and is not affected by this EP.

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 Specification. Prior NRC approval for the implementation of this modification is not required.

PC/M 216-191, Supplement 0-1


## ABSTRACT

The purpose of this Engineering Package is to provide documentation to perform the maintenance practice of installing steam generator flexible tube stakes ("stakes"). Stakes will be installed in those Steam Generator (S/G) tubes exhibiting circumferential cracking near the tube expansion transition region (i.e. at or near the top of the tubesheet). In-Service Inspection (via eddy current testing) will determine when circumferential cracking exists and when installation of stakes are necessary. The stakes are designed to prevent tube to tube contact, should the crack progress and the tube sever. All staked tubes are to be plugged in accordance with PC/M 108-191M.

This PC/M is classified as Safety Related, since the stakes may be relied upon to prevent a severed tube from contacting other S/G tubes, which are part of the RCS pressure boundary.

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 for implementation is not required.

**It is noted that implementation of this PC/M is contingent upon Facility Review Group approval of Revision 1 to PC/M 108-191M (S/G Tube Plugging). PC/M 108-191M Revision 1 will provide the documentation to permit the installation of the current ABB/Combustion Engineering welded S/G tube plugs.**

This PC/M provides the documentation to install stakes as a maintenance practice; the S/G instruction manual will be revised to provide the drawings necessary to perform this maintenance on an "as-needed" basis.

<u>SUPPLEMENT NO. 1</u>

Supplement No. 1 to this Engineering Package provides revised drawings of the S/G flexible tube stakes ("stakes"). Neither the design, the installation, nor the performance of the stakes is adversely affected by these drawing revisions. Supplement No. 1 does not effect, amend nor change the original Safety Evaluation, the Technical Specification or the Technical Specification bases.

PC/M 216-191, Supplement 0-1

SAFETY EVALUATION

Since the primary function is to prevent the staked tube from contacting other tubes, it has the potential to affect the RCS pressure boundary, this Engineering Package is classified as Safety Related. An FSAR Change Package (Attachment 7.6) for Section 5.5.1.3a is included to delineate installation of stakes in plugged S/G tubes which have circumferential cracks near the expanded region just above the S/G tubesheet.

The addition of stakes does not alter the operation, performance nor function of the S/G. Since it will be installed in tubes which are being plugged, it does not reduce the available S/G heat transfer area insofar as S/G plugging limits and asymmetry between S/G's is addressed by PC/M 108-191M. The stakes are being installed in a tube with circumferential cracks to prevent tube-to-tube contact should the staked tube sever. S/G operation with the tube stakes is bounded by the Technical Specifications (Reference 6.2). The licensing basis has not been altered by this modification.

A review of the potential failure modes for the S/G, as discussed in FSAR Chapter 5, was performed and is documented in Table 3-1. It was determined that no new failure modes are introduced, nor the probability of existing failure modes are increased by this modification.

The modifications associated with this Engineering Package do not affect any Limiting Condition for Operation (LCO) nor any Technical Specification Basis. The LCO's (and associated basis) reviewed were 3/4.4.5 and 3/4.7.2. Therefore this modification is bounded by the Technical Specifications (Reference 6.2).

With respect to Title 10 of the Code of Federal Regulations, Part 50.59(a)(2), 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 may be created; or (iii) if the margin of safety as defined in the basis of any technical specification is reduced. The modification provided by this Engineering Package does not involve an unreviewed safety question because each of the following seven (7) questions are answered with a negative response.

SAFETY EVALUATION (Continued)

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR)?

The function of the stake is to prevent the staked and plugged tube from contacting another tube should the former sever. Only those tubes with a circumferential crack near the expansion transition region will be staked. The installation of a stake does not increase the probability of a plugged tube severing, nor does it increase the probability of a S/G tube plug failure. Therefore this modification does not increase the probability of occurrence of an accident previously analyzed in the SAR.

2) Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

Accidents previously evaluated include S/G tube rupture (FSAR Section 15.4.4). S/G tube stakes will be installed in plugged tubes (isolated from primary systems) with the design criteria of preventing tube to tube contact in the event the plugged tube severs, under design conditions. Since the stakes will be installed in non-active tubes the consequences of a S/G tube rupture are not affected by this modification.

Other accidents previously analyzed include Main Steam Line Break (FSAR 15.4.6); Main Feedwater System (FSAR 15.2.10), and Control Malfunction.

The stake is designed to perform its function under design conditions, i.e. stabilize the staked tube under all conditions the S/G is subjected to. Therefore, this modification does not increase the consequences of an accident previously evaluated in the SAR.

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

Since the S/G tube stakes will be installed in plugged tubes (inactive) it can not affect active tubes or other components in the S/G unless its S/G tube degrades and severs. Should this event occur, the S/G tube stake will be retained within both ends of the S/G tube and not contact other S/G tubes or S/G internals, under design conditions.

SAFETY EVALUATION (Continued)

4) 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 plugged tubes, isolated from active tubes and S/G internals. It is not considered credible to postulate the failure of a S/G support component such as a stake. Therefore, the stake will remain in the plugged tube and will not interact with other S/G internals. Since the purpose of this modification is to minimize the probability of a plugged tube from contacting other tubes, the S/G tube stakes do not increase the consequences of a malfunction of equipment important to safety.

5) 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 crack near the expansion transition region above the tubesheet and the tube is subsequently plugged. Since the stakes are isolated from the RCS and the failure of a support component is not considered credible, the stake by itself cannot 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.

6) 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. Should the tube in which it is installed continue to degrade and sever, it is designed to prevent tube-to-tube contact. It is designed to remain in place in a severed tube and to have minimal effect on the wear of the staked tube. It does 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.

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

The Technical Specification (Reference 6.2) basis reviewed were 3/4.4.5, 3/4.4.6.2 and 3/4.7.2. 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, nor affect the capability to detect imperfections, nor does it inhibit or promote RCS leakage.

SAFETY EVALUATION (Continued)

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

PC/M 160-191, Supplement 0

## ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to delete the Main Turbine Runback feature. The modification will leave the Turbine Runback logic unchanged. It will no longer be initiated. Turbine Runback occurs whenever there is a loss of a Steam Generator Feed Pump (SGFP) above 60% power or the loss of both Heater Drain Pumps (HDP) above 92% power. Deleting the Turbine Runback feature will be accomplished by lifting the Turbine Runback leads in the Turbine Digital Electronic Hydraulic Control (DEH) Cabinet, Sequence of Events Cabinet, and at Reactor Turbine Generator Boards (RTGB) 101 & 102. Cables 10712F, 10712J and 10717Z will be spared and all drawings will be revised accordingly. In addition the Control Room Turbine Runback Annunciator Window D27 will be rendered inoperative and will be spared.

The function of the Turbine Runback is to run the Turbine/Generator back at a predetermined rate upon loss of a SGFP or both HDP's until Turbine/Generator output decreases to 60% & 92% respectively as measured by turbine first stage (impulse) pressure. During a Turbine Runback the Main Governor valves throttle the steam flow until the load matches the setpoint of 60% or 92% load depending on the initiating event. During this event the Turbine/Generator RPM remains constant.

St. Lucie Unit 1 experienced a Reactor trip from 100% power on June 14, 1987 & June 30, 1988, due to a Turbine Runback which was caused by the loss of the 1B SGFP (Ref. 6.7 and 6.8). During both events a turbine runback was automatically initiated to approximately 60% power. In less than 30 seconds into the transient the Reactor Protection System initiated a Reactor trip on a high pressurizer pressure signal.

The purpose of removing the Turbine Runback feature is to minimize the effects of a partial loss of feedwater transient on the Plant and to provide the plant operators with additional time to restore 100% feedwater flow before a Reactor trip occurs.

A Thermal Hydraulics Analysis (Ref. 6.9) of a loss of SGFP transient was performed with and without the Turbine Runback feature. This analysis demonstrates that by removing the Turbine Runback feature a Reactor trip could be avoided, provided the plant operators restore 100% feedwater flow within 110 seconds into the event. If full feedwater flow is not restored within the 110 seconds, a reactor trip will occur on low steam generator levels. This transient will not challenge the Pressurizer Power Operated Relief Valves (PORV's) or the Main Steam Safety Valves (MSSV's) which lifted in the two SGFP loss events mentioned earlier.

Although the Main Turbine, Turbine Controls and the Turbine Runback feature do not perform a safety function per FSAR Section 7.7, this EP is classified as Quality Related because it requires work to be performed in the Control Room.

ABSTRACT (Continued)

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.

## SAFETY EVALUATION

This Engineering Package (EP) provides the engineering and design necessary to delete the Main Turbine Runback feature. Turbine Runback occurs whenever there is a loss of a Steam Generator Feed Pump (SGFP) above 60% power or the loss of both Heater Drain Pumps (HDP) above 92% power.

The purpose of removing the Turbine Runback feature is to minimize the effects of a partial loss of feedwater transient on the plant and to provide the plant operator with additional time to restore 100% feedwater flow before a Reactor trip occurs.

There are no licensing requirements impacted by this modification.

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. The modification included in this engineering package does not involve an unreviewed safety question because of the following reasons:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in SAR?

   The proposed activity does not increase the probability of occurrence of an accident previously evaluated because the functionality of the Main Turbine, Turbine Controls or Reactor Protection trip signals have not been changed by this modification. Based on this, the probability of occurrence of an analyzed accident remains unchanged.

SAFETY EVALUATION (Continued)

2) 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 due to the deletion of the Main Turbine Runback feature because the Main Turbine, Turbine Controls, or Runback feature do not serve a Safety Related function.

3) 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 a malfunction of equipment important to safety because deletion of the Turbine Runback feature does not adversely affect Safety Related equipment. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety.

4) 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 deletion of the Turbine Runback feature does not adversely affect Safety Related equipment. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety.

5) 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 during the loss of a SGFP transient without Turbine Runback the heat removal from the Reactor Coolant System (RCS) by the secondary side of the Plant is maintained and does not challenge the Reactor Protection System. The Turbine Runback feature is not Safety Related and the deletion of the Turbine Runback does not affect any Safety Related signals required to initiate a Reactor trip. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety. For these reasons, the proposed activity does not create the possibility of an accident of a different type than previously described in the FSAR.

SAFETY EVALUATION (Continued)

6) 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 deletion of the Turbine Runback does not adversely affect Safety Related equipment or signals required to initiate a Reactor trip. This modification does not degrade the reliability or increase challenges, directly or indirectly for equipment important to safety.

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

The proposed activity does not reduce the margin of safety as defined in the basis for any Technical Specification, because deletion of the Turbine Runback feature has been shown by analysis not to effect the basis of the Technical Specification.

The foregoing discussions constitute, per 10 CFR 50.59(b), the written safety evaluation which provides the basis that this modification does not involve an unreviewed safety question, nor a change to the Plant Technical Specifications. As such, prior NRC approval for the implementation of this PC/M is not required.

PC/M 090-191, Supplement 0

## ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to add discharge resistors across the shunt field of 125 VDC motor operated valves (MOVs) at St. Lucie Unit 1. The addition of discharge resistors is based on recommendations which state that upon MOV de-energization, high voltage transients are created which can cause insulation degradation and reduced motor life. This concern is addressed in the NRC Information Notice 88-72.

The safety evaluation of this EP has determined that this PCM does not constitute an unreviewed safety question as defined in 10 CFR 50.59 and does not require a change in the plant Technical Specifications. This PCM has no adverse impact on plant safety or operation. Therefore, this PCM can be implemented without prior NRC approval.

This EP involves modification of Nuclear Safety Related MOVs, and is therefore classified as Nuclear Safety Related.

## SAFETY EVALUATION

This Engineering Package provides the documentation necessary to install discharge resistors on the DC motor operated valves (MOVs) located at St. Lucie Unit 1. The following MOVs are affected by this EP:

| Tag Number | MOV Description |
|---|---|
| MV-08-3 | AFWP 1C Turbine Steam Valve |
| MV-08-13 | Steam Generator 1A to AFWP1C Turbine |
| MV-08-14 | Steam Generator 1B to AFWP 1C Turbine |
| MV-09-11 | AFWP 1C Discharge to Steam Generator 1A |
| MV-09-12 | AFWP 1C Discharge to Steam Generator 1B |

Presently, high voltage transients are induced during 125 VDC MOV de-energization that can potentially cause motor insulation degradation and reduced motor life, as addressed in NRC Notice 88-72. The use of discharge resistors will protect the DC MOVs against the potential effects of voltage transients and will not adversely affect operation of the MOV nor the AFW system.

This modification installs a discharge resistor in the local DC starter boxes across the shunt field winding of each 125 VDC MOV. This circuit configuration will allow induced voltage transients to dissipate across the fixed resistance load of the discharge resistor. The addition of discharge resistors has been evaluated to have no adverse affect to the safety-related MOVs. This EP does not affect (1) designed reactor coolant pressure boundary; (2) capability to shut down and maintain the reactor in a safety shutdown condition; (3) the capability to prevent or mitigate the consequences of accidents with potential exposures comparable to 10 CFR 100 levels.

SAFETY EVALUATION (Continued)

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. The modification included in this Engineering Package does not involve an unreviewed safety question for the following reasons:

1) *Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR)?*

   This modification does not adversely affect any equipment whose malfunction is postulated in the SAR. This modification adds a discharge resistor to the AFW MOVs and does not change the function, nor operation of the MOVs. This modification does not circumvent the valves' safety functions. This modification does not affect MOV pressure boundaries. Therefore, the probability of occurrence of an accident previously described in the SAR is not increased by this modification.

2) Does the proposed activity increase the consequences of an accident previously evaluated in the SAR?

   This modification does not affect the function or operation of the AFW System MOVs, nor does it affect other systems and components that are relied upon to mitigate accidents. Therefore, the consequences of an accident previously evaluated in the SAR in not increased by this modification.

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

   This modification adds discharge resistors across the shunt field winding of each 125 VDC MOV to protect against potential high voltage transients that may occur during de-energization of a MOV. However, a malfunction of the discharge resistor will result in an open circuit across the MOV shunt field winding, which in turn will be electrically equivalent to the existing MOVs. The reliability of the MOV motors will be increased by this change. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR has not increased by this modification.

SAFETY EVALUATION (Continued)

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

   The addition of the discharge resistors will not alter the original safety function of the MOVs. The discharge resistors have an open circuit failure mode and do not affect the existing performance of the MOV and therefore, the implementation of this EP would not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

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

   This modification adds protection against high voltage transient that may cause motor insulation degradation and reduced motor life. Protection is added by installing a discharge resistor across each DC MOV shunt field winding. This change does not alter the function or operation of a MOV, nor create any new failure mode or conditions which could 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 EP.

6) 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 modification does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated. The credible failure mode of the resistor is an open circuit. Therefore, a modified MOV circuit with a failed discharge resistor is electrically identical to the existing MOV circuit. The addition of the resistor does not create a new failure mode of the modified MOVs.

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

   The modification does not reduce the margin of safety as defined in the bases for any Technical Specification. Adding a discharge resistor to an MOV as per this modification enhances the reliability of the MOV. The installation of each discharge resistor will increase the battery loading approximately 0.083 amperes at 125 VDC. This increase loading occurs only during the time of MOV operation (several seconds). The 125 VDC system calculations were reviewed and determined that the small increased loading has a negligible affect on the 125 VDC System. The addition of a discharge resistor does not adversely affect the MOVs performance. The existing margin of safety as defined in the basis for any Technical Specification remains unchanged after the implementation of this modification.

SAFETY EVALUATION (Continued)

This modification does not reduce the margin of safety as defined in the basis for any Technical Specification. The implementation of this EP does not require a change to the plant Technical Specifications. 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.

## ABSTRACT

This Engineering Package provides for modification to the borated water level annunciation circuitry for the 1A and 1B Boric Acid Makeup (BAM) Tanks. The modification involves the addition of a low level alarm selector switch which provides three options. The new switch allows enabling of the low level alarm circuity for the 1A Tank (1B not enabled), or enabling of the low level alarm circuitry for the 1B Tank (1A not enabled), or enabling of the low level alarm circuitry for both the 1A and the 1B Tanks.

The purpose of this modification is to eliminate undesired annunciation (i.e., nuisance alarms and continuously lit windows) in the main control room from low borated water levels in the BAM Tanks. Current plant practice during normal operation involves use of one BAM Tank in combination with the Refueling Water Tank to satisfy the borated water source requirements of Technical Specification 3.1.2.8. Operating in this mode, the other BAM Tank is not required to be operable and as a result its water level frequently drops low enough to cause the undesired annunciation. The elimination of this undesired annunciation is consistent with the NUREG 0700 "Guidelines for Control Room Design Review" "Dark Annunciator" concept. Under normal operating conditions, annunciators will not be illuminated.

The BAM Tank low level alarm circuits do not perform a Safety Related function, however, Table 7.5-2 of the St. Lucie Unit 1 FSAR incorrectly lists the BAM Tank level indicators as Safety Related. The alarm circuits receive their signals from these level indicators and provide main control room annunciation to Operations, informing them that the associated BAM Tank level is approaching Technical Specification limits. The selector switch added by this modification will be seismically mounted to prevent possible interaction with Safety Related equipment. This Engineering Package has evaluated the safety classification of the affected equipment, therefore this package is classified Safety Related.

A safety evaluation of this modification has been performed in accordance with 10 CFR 50.59. This evaluation concludes that implementation of this Engineering Package 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 for implementation of this modification is not required.

# SAFETY EVALUATION

This modification provides the details to install a BAM Tank low level alarm selector switch. The BAM Tank's low level alarm provide annunciation in the Control Room (Windows N15 and N16) when the associated tank level (1A or 1B) approaches the Technical Specification 3.1.2.8 limit. This Technical Specification is satisfied by meeting at least two of four possible conditions for borated water sources. The setpoints are presently set for two BAM Tank operation, however, the Operations Department normally uses one BAM Tank (at a higher minimum water level and higher boric acid concentration), and the Refueling Water Tank (RWT) to meet Technical Specification 3.1.2.8. Since only one BAM Tank is required to be maintained above the low level alarm point, and operations uses the other BAM Tank at varying levels during normal plant operations, the low level alarm is often energized. This modification adds a selector switch to select the BAM Tank to be maintained per Technical Specification, and defeat the low level alarm of the non-selected BAM Tank. The low level alarm of the non-selected tank will no longer be annunciated, eliminating the nuisance alarm.

This evaluation addresses the acceptability of the installation of a BAM Tank low level alarm selector switch.

Table 7.5-2 of the St. Lucie Unit 1 FSAR lists the BAM Tank level indicators as Safety Related. Safety Related items are those necessary to assure one or more of the following:

1) integrity of the reactor coolant pressure boundary,

2) the capability to shut down and maintain the reactor in a safe shutdown condition, or

3) the capability to prevent or mitigate the consequences of accidents with potential exposures approaching 10 CFR Part 100 levels.

The BAM Tank level circuitry does not perform a Safety Related function since it is not necessary to assure these three criteria. The mounting of the selector switch on RTGB 105 is designed for seismic loading to prevent possible interaction with safety related equipment around it. This engineering package is classified as Safety Related, since this Safety Classification evaluation has defined the associated equipment to be of a lower classification as defined in the FSAR.

By implementing these changes, low level annunciation for the BAM Tanks will conform with the NUREG 0700 "Guidelines for Control Room Design Review" "Dark Annunciator" concept. Under normal operating conditions, annunciators are not illuminated.

SAFETY EVALUATION (Continued)

The addition of the BAM Tank low level alarm selector switch does not alter the operation, nor the function of the borated water sources. This method of operation is bounded by existing Technical Specification. The licensing basis has not been altered by these modifications.

A review of the "Single Failure Analysis" for CVCS (PSL-1 FSAR, Table 9.3-25) was performed and determined that the analysis is not affected by this modification.

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. The modification included in this engineering package does not involve an unreviewed safety question because of the following reasons:

1) Does the proposed activity increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR)?

   The proposed activity does not increase the probability of occurrence of an accident because the Control Room annunciators are not accident initiating devices. The annunciators function to alert operators of abnormal plant conditions and serve no controlling function.

2) 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 Control Room annunciators are not used by the operators to mitigate an accident. Each BAM Tank has level indication available on RTGB 0105 and is used by the operators to determine the status of the BAM Tank's. The alarms do not increase the radiological doses of an accident.

3) 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 switch installed by this modification or the annunciators perform no safety related function. The switch mounting was also analyzed and found to be adequate and not affect the seismic response of RTGB 105.

4) 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 selector switch and the BAM Tank's low level alarm performs no nuclear safety related or equipment protection function. The BAM Tank's (1A and 1B) each have a low-low alarm and a high alarm. These alarms warn operators of potential abnormal conditions. The low-low alarm warns of emptying a tank which could lead to possible BAM pump damage. The high alarm warns of possible tank overflow.

5) 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 Control Room annunciators are not accident initiating devices. These annunciators warn operators of abnormal plant conditions and serve no controlling function.

6) 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 method of operation for a Control Room annunciator has not changed. The addition of the selector switch has also been analyzed to maintain the seismic integrity of the RTGB panel.

7) Does the proposed activity reduce the margin of safety as defined in the bases 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 alarm circuitry is not included in the bases for any Technical Specification. The existing Technical Specification ensures that adequate Shutdown Margin exists.

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. Prior NRC approval for the implementation of this modification is not required.

## ABSTRACT

This Engineering Package (EP) includes the engineering and design necessary to delete the Acoustic Flow Recorder (FR-1200). The inputs to the Recorder are also wired to the SAS System and both monitor and provide trend information for the Power Operated Relief Valves/Safety Relief Valves (PORVs/SRVs). The Acoustic Recorder is presently inoperable and spare parts are unavailable due to obsolescence. By implementing this (EP), this recorder will be deleted from the Post Accident Panel-1B, in the Unit 1 Control Room. A cover plate will be placed over the hole where the recorder existed. During an Emergency the Operators primary means for assessing leakage in the PORVs/SRVs is by use of the Flow Indicators (FI-1200, 1201, 1202, 1402, 1404) instead of the Acoustic Recorder (Ref. 6.10). In addition, the SAS trends may be used.

This Engineering Package is classified as Safety Related, since the Recorder is referenced in the FSAR, Page 7.5-36 and is listed on Table 7.5-2 as a safety related recorder. Regulatory Guide 1.97 response takes credit for the SAS System and the Flow Indicators. The deletion of the Acoustic Recorder will not have any impact or detrimental effects to safety equipment located in PAP-1B, nor will it adversely affect the seismic qualification of PAP-1B.

A safety evaluation of this modification has been performed in accordance with 10CFR50.59. 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.

## SAFETY EVALUATION

The Acoustic Flow Recorder (FR-1200) provides monitoring of flow through the Power Operated Relief Valves (PORVs) V1402, V1404 and Safety Relief Valves (SRVs) V1200, V1201, and V1202. The recorder receives its signals from the Acoustic monitors which sense the flow through the valves (PORVs/SRVs). Each Acoustic sensor sends a signal to its respective indicator on the PAP-1B which indicates valve position. The Safety Assessment System also has inputs from each flow transmitter, in which the SAS system records and trends each valve in the similar fashion to the Acoustic Recorder. Recorder (FR-1200) serves no Safety Related Function in achieving Safe Reactor Shutdown in the event of a Design Basis Event (DBE) and does not serve to mitigate the consequences thereof.. However, since the recorder obtains 115VAC from a safety channel inside the PAP panel, it is considered Safety Related and is included in the list of Safety Related Recorders on Page 7.5-36, Table 7.5-2 in the FSAR. Therefore, this Engineering Package is classified as Safety Related.

SAFETY EVALUATION (Continued)

This Acoustic Flow Recorder provides the Operators with trend information for the Power Operated Relief Valves and Safety Relief Valves (PORVs/SRVs). The Recorder (FR-1200) will be deleted from the Control Room Post Accident Panel #1B. The SAS system provides the same information as the Acoustic Recorder. Therefore, the deletion of Acoustic Recorder has no adverse effects on the monitoring of the (PORVs/SRVs). This EP will provide the circuit and panel modifications required to permanently delete the Acoustic Flow Recorder.

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 created; 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. The modification included in this engineering package does not involve an unreviewed safety question because of the following reasons:

1) 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 while the Acoustic Flow Recorder (FR-1200) is Safety Related, it serves no controlling function. The deletion of the recorder does not affect the seismic integrity of Post Accident Panel-1B.

2) 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 due to the deletion of the Acoustic Flow Recorder (FR-1200) from Post Accident Panel-1B. The recorder is Safety Related, but serves no Safety Related function. The Emergency Operating Procedures use the Flow Indicators to assess leakage in the PORVs/SRVs.

3) 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 Acoustic Recorder performs no controlling functions and the removal of this recorder does not affect the seismic integrity of the panel (PAP-1B) nor does it affect the reliability of the Flow Indicating loops (F-1200, 1201, 1202, 1402, 1404).

SAFETY EVALUATION (Continued)

4) 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 recorder performs no Safety Related function and is not used to mitigate the effects of an accident and does not affect the seismic integrity of the panel (PAP-1B).

5) 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 recorder performs no Safety Related function and the Indicators are used during an emergency. A review of the failure mode analysis for the PORV and SRV Valve Position Indication (FSAR, Page 7.5-22 and Page 7.5-36) has been performed and it has not been impacted by this modification. In addition, the overall seismic integrity of the panel (PAP-1B) will not be degraded by this modification.

6) 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 Acoustic Flow Recorder performs no Safety Related function. The deletion of the Safety Related Recorder will not affect the reliability of the Flow Indicator loops (F-1200, 1201, 1202, 1402, 1404) or change the function of the loops.

7) 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 deleted recorder is not included in the basis of the Technical Specification for the Reactor Coolant System or any Technical Specification.

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 nor a change to the Plant Technical Specifications and prior NRC approval for the implementation of this modification is not required.

ABSTRACT

This Engineering Package provides for the replacement of several obsolete Fischer and Porter Model 51-1401 indicating controllers installed on St Lucie Unit 1 for charging pump suction pressure and component cooling water temperature indication and control. In the current plant configuration, pressure indicating controllers PIC-2224X, PIC-2224Y and PIC-2224Z provide Charging Pump 1A, 1B and 1C local indication and pump trip on low suction pressure with no safety injection actuation signal (SIAS) present; in the event of a loss of coolant accident (LOCA), SIAS will override the low pressure trip to assure concentrated boric acid is injected into the reactor coolant system.

In the modifications provided with this EP, each charging pump indicating controller is replaced with a local indicator (pressure gauge) and pressure switch to retain the existing system design features. Since the pressure switches are interposed in Nuclear Safety Related Class 1E pump breaker control circuits, and since they are connected to ASME Class III piping, they will be qualified for Nuclear Safety Related Class 1E, Seismic Category I service. Therefore, this EP is classified as Nuclear Safety Related.

Additionally, this EP provides for the replacement of temperature indicating controllers TIC-14-4A and TIC-14-4B, which throttle temperature control valves TCV-14-4A and TCV-14-4B to regulate Intake Cooling Water flow depending upon Component Cooling Water outlet temperature from the Component Cooling Water Heat Exchanger (CCWHX) 1A and 1B, thus moderating CCW temperature. This EP replaces these obsolete controllers with new, currently available, pneumatic controllers. The replacement pneumatic controllers are qualified Seismic Category I.

The safety evaluation has shown that this EP does not constitute an unreviewed safety question nor require a change to the Technical Specification and therefore prior NRC approval is not required for implementation.

## SAFETY EVALUATION

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 for modifications to the Chemical Volume and Control System, CVCS (Charging Pump suction pressure instrumentation), the Component Cooling Water (CCW) System and Intake Cooling Water (ICW) System (CCW Heat Exchanger temperature control).

The modifications have been evaluated under 10CFR50.59 and it has been determined that 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 the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased. The replacement of the existing Fischer and Porter pressure indicating controllers with new ASCO pressure switches and Ashcroft pressure gauges and the Fischer and Porter temperature indicating controllers with Foxboro indicating controllers has no affect on the ability of the CVCS, ICW and CCW systems to perform their Nuclear Safety Related design basis function per the St Lucie - Unit 1 FSAR. Existing pressure and temperature measurement points (pressure taps and thermowells) are retained and utilized. All new equipment, including tubing, fittings and connectors, is qualified for Nuclear Safety Related service.

(ii) The possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis report is not created. The new pressure switches for Charging Pump suction pressure are Nuclear Safety Related Class 1E qualified and are sufficiently rated for their 125V dc control application. Further, the new configuration no longer requires 120V ac control power as it relies on pneumatic sensing only. Existing setpoints are retained and all new and existing equipment are compatible at all interface points. Existing channel independence and redundancy is retained after the implementation of the PC/M. The integrity and functions of systems, components and structures are either maintained or improved by the replacement equipment.

SAFETY EVALUATION (Continued)

(iii) The margin of safety as defined in the bases for any Technical Specification is not reduced since this modification maintains existing levels of protection for Nuclear Safety Related equipment (e.g., CVCS, CCW and ICW systems and components). Existing charging pump trip setpoints (10psia) and interlocks (time delays and SIAS permissives) are retained, as are CCW temperature control valves travel limiters (including the high limit relays and valve stops). CCW temperature indication and control are maintained or improved use at the Moore 58H high limit relays and Norgren Model 11-024-042 pressure regulators. These modifications maintain or enhance the existing design.

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 nor a change to the technical specifications and therefore prior Nuclear Regulatory Commission approval for the implementation of this PC/M is not required.

PC/M 312-189, Supplement 0

## ABSTRACT

This Engineering Package (EP) will provide the engineering and design details required to implement the modifications to the New Fuel Crane Hoist.

The purpose of the New Fuel Crane Hoist is to remove the new fuel assemblies from their shipping containers, place them in the new fuel storage racks and eventually to transport them to the Containment Fuel Handling Machine. This process requires the hoist operator to repeatedly cycle the motor to achieve the necessary load spotting or positioning. This "jogging" technique eventually results in excessive heat buildup at the "Magnetorque" motor to an extent that rotor warpage develops. Also, as each start cycle is initiated, inrush current delivered to the motor produces stress on the windings, resulting in potential premature failure.

The replacement of the existing wound motor and stepless controller with a squirrel cage motor and solid state "Smartorque" adjustable frequency controller will ensure the efficiency of the system during new fuel handling operations. In addition, a mechanical load brake will replace the electric load brake. The mechanical load brake is an additional means of redundancy, in case the dc magnetic load brake does not operate, to prevent the load from dropping due to failure of the electric motor. The bridge and trolley controls will not be affected by the above modifications.

The equipment being modified by this EP performs a non-nuclear safety related function. However, since the new fuel crane is seismically designed and is used in handling new fuel assemblies (and since mishandling could result in fuel damage) this EP has been classified as Quality Related.

Results of the safety evaluation conclude that modifications presented by this Engineering Package do not constitute an unreviewed safety question, do not require any changes to the Plant Technical Specifications and therefore, no prior NRC approval for the implementation of this PC/M is required.

The implementation of this PC/M will not have any impact on plant safety or operations.

## SAFETY EVALUATION

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.

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SAFETY EVALUATION (Continued)

This Engineering Package covers the modifications to the St. Lucie Unit 1 New Fuel Crane Hoist. The replacement of the existing wound motor and stepless controller with a squirrel cage motor and solid state "Smartorque adjustable frequency controller will ensure the efficiency of the system during new fuel handling operations. The addition of a mechanical load brake adds extra assurance that a load cannot be dropped due to loss of power to the motor. The implementation of this EP increases the reliability of the New Fuel Crane Hoist and enhances the system operation.

The equipment being added by this EP performs no safety related function and will not interact with any safety related equipment or function. The New Fuel Crane is seismically designed and is used in handling new fuel assemblies, (and since mishandling could result in fuel damage), this EP has been classified as Quality 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 these modifications. The replacement of the existing equipment with a new motor controller and mechanical load brake will prevent: the heat buildup that now occurs at the "Magnetorque" motor, which results in motor warpage; the stress on the motor windings that will result in premature failure of the motor; and the possibility of a load being dropped due to loss of power to the motor. 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. There is no change in operation, capacity or function of the equipment due to these modifications. There is no adverse interaction with any safety related equipment or system, therefore, a failure of any safety related component which could cause, contribute to, or become a factor in a new type of accident cannot result from this modification.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The New Fuel Crane being modified by this PC/M is not addressed in the Technical Specifications.

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.

## ABSTRACT

This Engineering Package provides the engineering and design details required to replace the existing Main Fire Alarm Panels. The panels are part of the fire detection system. The existing duct mounted detectors are not compatible with the new panels and will also be replaced.

The existing panels are obsolete and spare parts are no longer available. The replacement panels represent the latest evolution in Honeywell's Fire Detection System's hardware and software. The new panels are fully compatible with the existing plant fire detection system.

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 is classified as Quality Related.

The installation of the equipment described above does not involve an unreviewed safety question, has no effect on plant safety or operation and does not require a change to the plant Technical Specifications. Therefore, prior NRC approval is not required for the implementation of this Engineering Package.

## SAFETY EVALUATION

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 Engineering Package (EP) provides the engineering and design details required to replace the existing obsolete Main Fire Alarm Panels with new updated panels. The existing duct mounted detectors will also be replaced since they are not compatible with the new panels. The new panels and detectors are fully compatible with the existing plant fire detection system.

The implementation of this EP will improve the reliability of the fire detection system, by replacing obsolete equipment. This ensures 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.

SAFETY EVALUATION (Continued)

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 by these modifications. The replacement of the obsolete fire alarm panels with new panels and the replacement of the duct mounted detectors will enhance the operation of the fire detection system. The new panels and detectors 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. Therefore, the implementation of these modifications cannot increase the probability of occurrence or the consequences of an accident or malfunction of equipment.

(ii) The possibility of an accident or malfunction of equipment of a different type than any evaluated previously is not created. The fire alarm panels and detectors are not required during an accident condition nor will they prevent safety related equipment from performing their functions. This modification does not affect any safety related equipment.

(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, 3/4.3.3.7, 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 or a change in the Technical Specifications and prior NRC approval for the implementation of this PC/M is not required.

PC/M 399-988 Supplement 1

## ABSTRACT

This Engineering Package provides the necessary details for the installation of a Fuel Dispensing Facility for the St. Lucie site. The facility will consist of the foundation and spill retainer for two tanks (one 8,000 gallon and one 10,000 gallon) as well as the fuel dispensers and necessary appurtenances.

The Fuel Dispensing Facility does not perform any Nuclear Safety-Related functions. It will be constructed at the south end of the site just west of the east basin and is not in the vicinity of any safety-related equipment or systems, nor will it impact any safety-related functions. Accordingly, this Engineering Package has been classified as Non-Nuclear Safety-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. Furthermore, the implementation of this modification does not require a change to the Plant Technical Specifications and has no detrimental effect on plant safety and operation. Therefore, prior NRC approval for implementation of this modification is not required.

Supplement 1

This supplement includes the design details for adding an electric sump pump to empty rainwater from the fuel dispensing facility spill retainer. In addition, the original design drawings will be updated to reflect as-built conditions and the PC/M Expiration Date has been extended to December 31, 1990. These changes require the original engineering design bases and design analysis to be amended, but have no effect on the original safety evaluation or the technical specifications.

## SAFETY EVALUATION

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. The modifications included in this engineering package do not involve an unreviewed safety question because of the following reasons:

SAFETY EVALUATION (Continued)

i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Updated Safety Analysis Report are not increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident. It has been shown that there will be no adverse impact on structures, systems, or components greater than 240 feet from the Fuel Dispensing Facility should an explosion occur there. Since the tanks will be located at least 700 feet from the nearest safety-related structure, system, or component and does not perform any function either directly or indirectly related to Power Plant operations, there will be no adverse impact on Nuclear Safety.

ii) The possibility of an accident or malfunction of a different type than any evaluated previously in the Final Updated Safety Analysis Report will not be created by this modification because the modification involves non-nuclear safety-related structures and failure of any items added by this modification will not impact any nuclear safety-related functions. In addition, any mishap at the Fuel Dispensing Facility including fire, explosion, and construction activities will not cause an accident or malfunction of any structure, system, or component important to Nuclear Safety.

iii) 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.

The Fuel Dispensing Facility does not perform any safety-related functions. A failure mode evaluation has been performed for this modification and it has been determined that no new failure modes have been introduced to the plant. An explosion analysis and fire analysis have been performed in accordance with the St. Lucie Unit 1 FSAR, the St. Lucie Unit 2 FSAR, and 10 CFR 50 Appendix R, and it has been concluded that there will be no adverse effect on the plant as the design of the facility meets or exceeds the requirements of these documents. Accordingly, this engineering package has been classified as non-nuclear safety-related.

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.

PC/M 073-987 Supplement 1

## ABSTRACT

This Engineering Package (EP) covers the replacement of the now obsolete Fischer and Porter transmitters with the currently manufactured equivalent Rosemount transmitters. The transmitters are providing tank level and process flow monitoring signals in the Makeup Water System and in the Steam Generator Blowdown Treatment Facility.

The existing Fischer and Porter transmitters do not provide any interface with the safety related systems, therefore, this EP is classified Non-Nuclear Safety Related. Since this modification is a one-for-one replacement of the existing Fischer and Porter transmitters with the equivalent Rosemount transmitters, the same classification applies.

The safety evaluation of this EP does not involve an unreviewed safety question, and does not require a change in the Plant Technical Specifications. This EP does not impact plant safety and operation, therefore, NRC approval for these modifications, prior to their implementation, is not required.

SUPPLEMENT 1

Supplement 1 of this EP has been issued to reflect current Rosemount model numbers for transmitters LT-36-1 and LT-36-3. The model number for these devices (as purchased) is 1151LT4EBOB22D and is included in this package with this revision.

This supplement does not effect the safety evaluation; the implementation of this PCM does not affect the Plant Technical Specifications and does not constitute an unreviewed safety question.

This EP has no impact on plant safety or operation.

## SAFETY EVALUATION

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.

The modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

SAFETY EVALUATION (Continued)

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Final Updated Safety Analysis Report, Chapters 8 & 9, are not increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Final Updated Safety Analysis Report, Chapters 8 & 9, will not be created by this modification because the function of the transmitters has not been altered.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since the new transmitters are all classified non-nuclear safety related and do not affect any technical specification.

The existing Fischer & Porter transmitters do not provide any interface with the safety related systems, therefore, this EP is classified Non-Nuclear Safety Related.

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 nor require a change to the technical specifications and therefore prior Commission approval for the implementation of this PCM is not required.

PC/M 020-187, Supplement 0-2

## ABSTRACT

This Engineering Package provides for the replacement of the Containment Level Monitoring System on St. Lucie - Unit 1 with instrumentation possessing impaired performance characteristics. The current system consists of one channel of narrow range level monitoring (from -6'-7 to 0'-0) and two redundant channels of wide range level monitoring (from -1'-0 to 26'-0). The replacement IMO Delaval level sensors utilize magnetic reed switches in a variable resistance network, providing improved accuracy and repeatability while minimizing drift. The new IMO Delaval sensors provide one channel of narrow range monitoring (-6"8 to 0'-0) and two redundant channels of wide range monitoring (from -1'-0 to 16'-1). Additionally, the new equipment (including Control Room mounted modular transmitters/receivers) will provide ease of calibration, reducing stay-time and minimizing man-rem exposure in the containment sump area.

The new instrumentation is qualified for Nuclear Safety Related Class 1E service and is environmentally qualified per 10CFR50.49. Existing plant commitments to Regulatory Guide 1.97 (Post Accident Monitoring) have been reviewed and are addressed in this Engineering Package. Since this PC/M modifies equipment required for post-accident monitoring which is qualified for Nuclear Safety Related use, this PC/M is designated Nuclear Safety Related.

The safety evaluation of this package has shown that the implementation of this PC/M does not constitute an Unreviewed Safety Question and requires no revision to the Unit 1 Technical Specifications, therefore, prior NRC approval is not required for implementation. This PC/M has no impact on plant safety and operation, or the Plant Technical Specifications.

<u>SUPPLEMENT 1</u>

Supplement 1 of this EP was issued to remove holdpoints on implementation and to issue conduit support details and associated calculations.

<u>SUPPLEMENT 2</u>

Supplement 2 of this EP is issued to remove holdpoints on system start-up/operation. All pertinent qualification documentation for the new Containment Level Monitoring System has been received and is issued with this EP supplement.

The safety evaluation of this package shows that the implementation of this PC/M does not constitute an Unreviewed Safety Question and requires no revision to the Unit 1 Technical Specifications, therefore, prior NRC approval is not required for implementation. This PC/M has no impact on plant safety and operation, or the Plant Technical Specifications.

## SAFETY EVALUATION

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 consequence 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.

The modifications have been evaluated under 10CFR50.59 and it has been determined that the modifications included in this Engineering Package do not involve an unreviewed safety question as demonstrated by the answer to the questions below:

(i) Does the Proposed Activity Increase the Probability of Occurrence of an Accident Previously Evaluated in the Safety Analysis Report?

This modification does not increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report. The new containment level monitoring instrumentation installed with this PC/M has been qualified for Nuclear Safety Related Class 1E service in order to satisfy Regulatory Guide 1.97 and existing plant design requirements. This design utilizes physical and electrical separation for redundant channels and provides new equipment (GEMS level sensors) which has been qualified for harsh environment in accordance with 10CFR50.49.

The new instrumentation (including level sensors and Control Room mounted modular transmitter/receivers) have been specified to assure that system electrical load is within the limits of the existing 120 VAC power supplies utilized. Fuses are provided on the existing 120 VAC power feeds to preclude the propagation of a fault of this equipment from affecting other safety circuits. Power conditioning, including voltage step and rectifying, is provided internally at the modular transmitter/receivers in the Control Room. No other operating equipment (pumps, motors, etc) is affected by this modification, either through direct control or interlock. The Containment Level Monitoring System instrumentation is used for monitoring only and does not use any setpoints for alarm or control functions. No other system interfaces are involved with this PC/M.

(ii) Does the Proposed Activity Increase the Consequences of an Accident Previously Evaluated in the Safety Analysis Report?

SAFETY EVALUATION (Continued)

The consequences of an accident previously evaluated in the Safety Analysis Report are not increased with the implementation of this PC/M. The replacement of the existing Containment Level Monitoring instrumentation with the new GEMS sensors does not change, degrade or prevent actions described or assumed in accidents described in the Safety Analysis Report. In accordance with St. Lucie - Unit 1 Emergency Operating Procedures, subsequent to a Loss of Coolant Accident (LOCA), plant operations is required to continually monitor containment level to verify proper transfer of water from the Refueling Water Tank (RWT) to the containment. Variance of rate of change of RWT level against containment level outside prescribed limits may indicate leakage (release of fission product) outside containment. The new GEMS instrumentation installed with this PC/M satisfies the criterion that the relationship between RWT and containment levels be monitored post-LOCA. As stated above, improved system accuracy will provide additional confidence that indicated level is true; extrapolation of data will be more precise.

The new instrumentation has been demonstrated by test to be accurate to $\pm$ 3% of indicated range post-LOCA, which is an improvement over the $\pm$ 5% accuracy (optimum conditions) provided by the existing Barton transmitters.

New mounting hardware in the containment building, as well as the new sensors themselves, are stainless steel composition and do not introduce any new halogens or hydrogen generating material into the containment. Existing electrical penetrations are utilized in this design such that the implementation of this PC/M will not require an additional penetration be installed in containment.

No plant structure, system or component used in mitigating the radiological consequences of an accident described in the Safety Analysis Report is affected by this modification. The replacement of the existing ITT Barton instrumentation with the new GEMS sensors has no effect on the ability of any other safety related structure, system or component to operate within its specified design limits.

(iii) Does the Proposed Activity Increase the Probability of Occurrence of a Malfunction of Equipment Important to Safety Previously Evaluated in the Safety Analysis Report?

SAFETY EVALUATION (Continued)

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased with the implementation of this PC/M. This modification has been designed to satisfy all applicable Nuclear Safety Related Class 1E, Seismic Category I criteria. Additionally, the new sensors (installed in the Containment Building) are qualified for harsh environment in accordance with 10CFR50.49. These features provide assurance that the Containment Level Monitoring System will operate as required in providing Regulatory Guide 1.97 (post-accident monitoring instrumentation) information to plant operations during and after LOCA.

The new IMO Delaval (GEMS) level sensors have improved reliability over the existing ITT Barton sealed sensor level transmitters. The existing instrumentation is more subject to drift and less accurate than the replacement instrumentation. Therefore, the probability of malfunction of the Containment Level Monitoring System will be reduced by this modification.

This modification has no adverse effect on any safety related system, structure or component. Existing embedded plates on the biological shield wall and in the reactor sump are utilized for the mounting of the new instrument rack assemblies, and have been analyzed for all applicable loads. The new rack assemblies (including mounting of GEMS instruments, protective cages and wireway) have been seismically designed. New conduit supports have been provided in these areas and designed for all applicable loads. At the electrical penetration assemblies (EPAs), existing feed-throughs are utilized with neither adverse effect on the EPAs themselves nor on the containment vessel or integrity thereof. Existing electrical raceway between the Control Room and the Penetration Room in the Reactor Auxiliary Building is retained and utilized without affect on the raceway or supports. In the Control Room, the new modular transmitter/receivers are installed in the Control Room Auxiliary Console (CRAC). The component mountings have been seismically designed and analyzed; the seismic qualification of the console has been reviewed and is not affected by this modification. Existing panel mounted meters and power supplies are retained and utilized and are compatible with the new system (120 VAC input power, 4-20 mA DC signal current).

As the new design addresses and satisfies applicable Nuclear Safety Related Class 1E criteria as stated above, and does not increase challenges to any nuclear safety related structure, system or component, the probability of occurrence of a malfunction of Nuclear Safety Related equipment is not increased with the implementation of this PC/M.

(iv) Does the Proposed Activity Increase the Consequences of a Malfunction of Equipment Important to Safety Previously Evaluated in the Safety Analysis Report?

This modification does not increase the consequences of a malfunction of Nuclear Safety Related equipment previously evaluated in the Safety Analysis Report. As discussed above, the Containment Level Monitoring System provides system status during normal and post-LOCA conditions. This system (including the new instrumentation installed with this PC/M) is utilized to indicate proper transfer of emergency core cooling water from the Refueling Water Tank (RWT) to the containment vessel by monitoring the change in containment level as it varies inversely with decreasing RWT level.

The new IMO Delaval (GEMS) sensors provide essentially the same continuous range of indication of containment level as do the existing Barton instrumentation, via one narrow range and two redundant wide range channels of indication. The only differences are that the GEMS narrow range (LE-07-14A) measures from minus 6'-8" instead of minus 6'-7", and the upper limit of the wide range (LE-07-13A4 and LE-07-13B4) is 16'-1" rather than 26'-0". An overall increase in range of indication with improved accuracy across that range (as compared with the existing Barton instrumentation) is provided with this modification. This PC/M does not have any effect on any emergency operating procedures and does not modify any required operator actions per plant procedures. By maintaining existing levels of post-accident monitoring instrumentation (albeit with increased indicating range and improved accuracy), the consequences of a malfunction of Nuclear Safety Related equipment previously evaluated in the Safety Analysis Report are not increased with the implementation of this PC/M.

(v) Does the Proposed Activity Create the Possibility of an Accident of a Different Type than any Previously Evaluated in the Safety Analysis Report?

SAFETY EVALUATION (Continued)

This modification does not create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report. This PC/M modifies equipment required to monitor and respond to a Class 2 accident, which may lead to a breach of barriers and fission product releases. The loss of all containment level monitoring instrumentation could prevent mitigative action in the event of leakage outside the Containment. The new instrumentation serves the same function as the ITT Barton instrumentation which it replaces. Plant criteria for redundancy and separation are maintained with the new design, which is more accurate than the existing equipment, and plant commitments for post-accident monitoring per Regulatory Guide 1.97 are satisfied. Additionally, existing plant systems utilized to verify and/or isolate the accident scenario discussed above (e.g. Containment Isolation Actuation System, CIAS, Area Radiation Monitoring System, ARMS, and RAB sump level alarms) are not affected by this modification. No other accident scenarios evaluated in the Safety Analysis Report are affected by this modification and no other types of credible accidents can be created by the implementation of this PC/M.

(vi) 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 create the possibility of a malfunction of nuclear safety related equipment of a different type than any previously evaluated in the Safety Analysis Report. This PC/M replaces existing pneumatic level sensors with new electro-magnetic sensors which will be subjected to submergence in the post-LOCA environment. Accordingly, the new GEMS sensors have been environmentally tested and qualified for submergence. Although the new system utilizes electronics instead of pneumatics in a submerged state, no new failure modes are postulated due to this new design feature. Additionally, the new sensors monitor level over the full flood range (to 26'-1"). Steel cages have been constructed to protect the float stem assembly against damage from impact (personnel or equipment/debris).

SAFETY EVALUATION (Continued)

The new GEMS sensors utilize different materials than those contained in the existing Barton Sensors. As part of the environmental qualification of the new equipment, these materials have been tested to assure their proper operation over the required design life of the plant. This provides assurance that the equipment will not malfunction due to material breakdown resulting from the post-LOCA environment. Outside containment, there are no harsh environment concerns to cause equipment failure due to age related mechanisms. There are no malfunctions of a different type than previously evaluated in the SAR which could be created by the implementation of this PC/M.

(vii) 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 as defined in the basis for any technical specification at St. Lucie - Unit 1. This modification maintains the existing design makeup of one narrow range and two redundant and electrically and physically independent channels of containment level. Technical Specification requirements to maintain all channels operable, with required action in the event of any channel failure are maintained and satisfied with this PC/M. Technical Specifications requirements for monthly channel check and channel calibration during each refueling outage are not affected by this PC/M. There are no controls, interfaces or interlocks associated with the Containment Level Monitoring System. The margin of safety for the Containment Level Monitoring System is not affected by this PC/M.

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 nor require a change to Technical Specifications and therefore prior Nuclear Regulatory Commission approval for the implementation of this PC/M is not required.

ABSTRACT

This Engineering Package (EP) details the requirements for modifications related to the replacement of the spent fuel storage racks, accomplished under PC/M 018-186. The tasks to be covered by this EP are as follows:

1. Modification of the existing spent fuel cooling system sparger pipe on the east side of the spent fuel pool to eliminate an interference with the new spent fuel storage racks.

2. Modifications to the hook limits of the existing spent fuel handling machine to permit the machine to access the outermost storage cells of the new racks.

3. Modifications to the existing work platform on the north side of the spent fuel storage pool to permit this platform to be moveable to eliminate interferences with the installation of the new racks and the future movement of spent fuel into the storage cells in this area.

4. Replacement of the existing underwater lighting fixtures with new low-profile portable fixtures.

5. Provision of suitable areas for receipt inspection and leveling of the new racks and laydown of the existing racks after removal from the Fuel Handling Building.

Modifications 1 through 4 are classified Safety Related because they will involve interfaces with the spent fuel pool and the spent fuel storage racks, which are Safety Related, and a modification to Safety-Related piping. Modification 5 is classified Non-Nuclear Safety Related because it involves no interfaces with any safety-related systems or equipment.

These modifications have been evaluated in accordance with 10CFR50.59. The safety evaluation has shown that the implementation of this Engineering Package does not constitute an unreviewed safety question nor are Technical Specifications affected; and, therefore, prior Commission approval for its implementation is not required. This modification will have no effect on plant safety and will facilitate plant operation.

Supplement 1

This supplement adds the following modifications to facilitate the rack replacement:

1. Additional modifications to the spent fuel handling machine, to physically extend the rails for the trolley and for the bridge and thus further expand the travel limits of the machine. Some of the handrail on the bridge will be made removable to avoid interference with items on the north wall of the room.

2. Further modification of the work platform, to make its supports removable so that it can, alternatively, be stored in an elevated position.

ABSTRACT (Continued)

3. Modification of the spent fuel pool purification suction piping, skimmer suction connection, and ion exchange discharge piping to eliminate interference with the new racks.

4. Modification of the service air piping on the north wall of the room to eliminate interference with the fuel handling machine.

This Supplement retains the Safety Related classification originally designated. Modification of the service air piping has been classified as Non-Safety Related since the piping is Non-Nuclear Safety Related. Modifications to the purification and ion exchanger piping and the service air piping have been classified as Quality Related due to their location in the vicinity of the spent fuel racks, although the systems themselves are Non-Nuclear Safety Related. These additional modifications do not introduce an unreviewed safety question as defined in 10 CFR 50.59, nor do they alter any Technical Specifications. Consequently, prior Commission approval is not required for the implementation of this package.

Supplement 2

This supplement replaces the originally designed work platform with one of a different design, which can be swung horizontally out of the way when required during refueling operations. The configuration of the new platform necessitates the relocation of pool temperature and level instrumentation. This modification is classified Quality Related because of potential interaction with the safety related spent fuel racks. This supplement also removes a holdpoint related to vendor information on the trolley bumper relocation which was unavailable at the time of the previous supplement. These additional modifications do not introduce an unreviewed safety question as defined in 10CFR 50.59, nor do they alter any Technical Specification. Consequently, prior NRC approval is not required for the implementation of this package.

## SAFETY EVALUATION

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 basis for any technical specification is reduced.

This Engineering Package provides for the installation of modifications related to the replacement of the spent fuel storage racks. It has been classified Safety Related because it involves modifications to and interfaces with Safety Related Systems. It does not involve an unreviewed safety question. The following are the bases for this conclusion:

SAFETY EVALUATION (Continued)

d. The modification to the underwater lighting fixtures will require the existing fixtures to be removed from the spent fuel pool. Neither the new nor the existing fixtures weighs more than a fuel assembly; therefore, any possible load drop will be enveloped by the previously evaluated fuel assembly drop accident.

e. The new storage and inspection areas have no safety related function, nor is there any possibility for their interaction with any safety related equipment or systems, therefore no possibility for a new type of accident is created.

(ii) f. No possibility for a new type of accident is created by the relocation of the spent fuel pool level switch and temperature element since their operability is unaffected. A drop of either of these components in the pool would be enveloped by the previously evaluated fuel assembly drop accident.

g. The possibility of dropping a portion of the spent fuel pool purification piping, ion exchanger piping, or the service air piping is enveloped by the previously evaluated fuel assembly drop accident as considered for several items above.

h. There is a possibility of dropping portions of the new work platform into the spent fuel pool during its installation. Any such drop will be enveloped by the previously evaluated fuel assembly drop accident, since the weight of any component which could be dropped is less than 2000 pounds.

The design of the modifications implemented by this Engineering Package ensures that the modified components will have no interaction with safety-related equipment, systems, or structures, except those interactions which are specifically addressed in the design; all potential construction accidents have been determined to be enveloped by previously evaluated accidents. Therefore, no possibility for a new type of accident exists.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification. The limitations set forth therein will be strictly maintained with the implementation of this Engineering Package, since the design intent, limitations, and constraints of the spent fuel pool, the spent fuel pool cooling system, and the spent fuel handling system remain unchanged.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question nor require a change to technical specifications and therefore prior Commission approval for the implementation of this Engineering Package is not required.

## ABSTRACT

This Plant Change/Modification includes the installation of a perimeter security barrier (intake canal crossing), intrusion detection system (perimeter and underwater), surveillance system (closed circuit television), security lighting and communications (paging).

This PCM is not classified as Safety Related since the canal crossing structures and components of the intrusion detection, surveillance, lighting and communication systems do not perform any safety function, and are located away from, and have no effect on, any safety related components. However, this PCM shall be considered quality related and quality related design requirements shall apply because of the following reasons: a) the perimeter security barrier closes the existing gap in the security perimeter fence as required by 10CFR73.55, b) the nature of the bridge and adjacent walls construction requires QC surveillance to assure proper installation of the concrete piles and correct use of concrete materials and mixes, and c) QC inspection/testing of the security system installation is required to assure proper operation and integration with the existing security system.

This PCM does not constitute an unreviewed safety question and enhances the existing plant security system. The installation of the items described above have no impact on plant operation and do not affect any safety related equipment.

SUPPLEMENT 4

This supplement incorporates the details required to install the second sonar system, monostatic microwave and their associated cables and conduits at the Intake Canal south crossing. The original safety evaluation is not affected by the modifications detailed in this supplement.

SUPPLEMENT 5

This supplement incorporates the details required to install a new intrusion detection system (perimeter and underwater) and to partially remove the existing system (perimeter and underwater) at the Intake Canal south crossing. The original safety evaluation is not affected by the modifications detailed in this supplement.

## SAFETY EVALUATION

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 report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

SAFETY EVALUATION (Continued)

This modification is not classified as Safety Related since the underwater intrusion system (canal crossing and associated systems, such as additional lights, paging stations, fencing and security system hardware) does not perform any safety function, and is located away from, and has no effect on any safety related components. The Ultimate Heat Sink analysis described in FSAR Section 9.2.7 is not affected by this modification since the failure of this crossing during a seismic event will not impede the flow of water, nor limit the intake canal and intake structure bay area from providing the plant with the primary source of shutdown cooling water capacity to dissipate reactor decay heat during normal and emergency shutdown conditions. This modification is on the intake canal area only, therefore the secondary source of cooling water (Big Mud Creek) is not affected.

The modifications included in this PCM do not involve any unreviewed safety questions because:

i The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the underwater intrusion detection system shall be installed in accordance with the quality related requirements, and this modification will have no effect on equipment performing a safety function.

ii There is no possibility for an accident or malfunction of a different type than any previously evaluated since the underwater intrusion detection system has no safety function, no changes have been made to any operational design, and the addition of security system hardware (TV cameras for surveillance, microwave 300B, fence protection FPS II, and a barrier net (Safenet) system) enhances the plant security system by implementing permanent security in the area of the intake canal crossing and integrating these modifications into the existing system. A canal crossing failure during a seismic event will not provide blockage of the primary water source to the intake cooling water system.

This canal crossing is not seismically designed and is a multi-span type bridge. Should a seismic event occur the canal crossing spans may individually collapse since there are construction joints separating each span. Based on April 1986 soil samples, the canal bottom contains firm soil at approximately elevation -26 ft. A collapse of this structure during a seismic event could hypothetically tip the 16 ft wide bridge walkway and drive it on edge into the canal, thereby leaving the top edge of the walkway no higher than elevation -10 ft at the center of the canal.

SAFETY EVALUATION (Continued)

This hypothetical scenario would still provide constant flow of water to the intake cooling water system since the lowest water elevation in the canal (with two units operating) is elevation -9 ft and the cooling water would still continue to flow over the collapsed canal crossing. In addition to water flowing over the collapsed sections, water can flow in between each collapsed section of the bridge, thereby not blocking the primary water source to the intake cooling water system.

Since the intake structure is provided with a method to prevent floating debris from entering the intake cooling water system, any possible floating debris from the canal crossing would not clog the intake cooling water system. In addition, based on engineering judgement the components associated with this modification would sink to the bottom if a seismic event were to hypothetically collapse this structure, therefore no floating debris could float downstream to clog the intake cooling water system.

iii This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PCM does not require a change to the plant technical specification.

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, therefore prior Commission approval is not required for implementation of this PCM.