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L-91-97
10 CFR 50.59

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

Gentlemen:

Re: St. Lucie Unit 2
Docket No. 50-389
Report of 10 CFR 50.59 Plant Changes

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 October 7, 1989 and October 6, 1990 and correlates with the information included in Revision 6 of the Updated Final Safety Analysis Report submitted under separate cover.

Very Truly Yours,

D A Sager

D. A. Sager
Site Vice President
St. Lucie Plant

DAS:JJB:jeo

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

Enclosure

DAS/PSL #395

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IE47

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

St. Lucie Plant Unit 2
Report of Changes Made
Under the Provisions of
10 CFR 50.59
for the period October 7, 1989
to October 6, 1990

Plant Change/Modifications reportable pursuant to 10 CFR 50.59 for St. Lucie Unit 2 FSAR Amendment 6

Number	Supplement	Title
222-284	0	2HVE-21 A & B Gravity Damper Removal
144-286	0	PASS Dissolved H2 Analyzer
100-287	0-1	CEDMCS/ACTM Upgrade
373-288	0	FHB L-Shaped Door
256-290	0-1	Refueling of PSL-2 For Cycle 6 Operation
399-988	0	Fuel Dispensing Facilities
315-989	0	Discharge Canal Nosing Area Drain Enhancement

PC/M 222-284 Rev. 0

ABSTRACT

The modifications associated with this change provide for the removal of gravity dampers associated with the Control Element Drive Mechanism (CEDM) cooling fans. The implementation of this modification did not create an unreviewed safety question per the requirements of 10CFR 50.59 or require prior NRC approval because the gravity dampers serve no safety function nor are they part of any Technical Specification. The removal of the dampers does not affect the ability of the CEDM cooling system to perform the normal plant operational requirement of cooling the CEDM assemblies.

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 the probability for an accident or malfunction of a different type than any evaluated may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The CEDM Cooling System is not safety related per Section 9.4.8.5 of the St Lucie Unit 2 FSAR. The CEDM Cooling System serves no safety function, is not required to achieve safe shutdown, or mitigate the consequences of a design basis event. The subject modification to the system does not affect the ability of the system to perform its intended function of cooling the CEDM magnetic jack coils under normal plant operation. Therefore, this PC/M neither increased the probability or consequences of a previously analyzed accident nor does it create any new type of accident. In addition, it does not reduce the margin of safety as defined in the bases for 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 and prior Commission approval for the implementation of this PC/M is not required.

ABSTRACT

This plant modification provided an alternate means for the Post Accident Sampling System (PASS) to measure and indicate the concentration of hydrogen in a sample of reactor coolant. The existing hydrogen analyzer will remain intact and serve as a backup. The implementation of this change did not create an unreviewed safety question per the requirements of 10CFR 50.59 or require prior NRC approval since there is no involvement with safety related equipment nor have any effect on plant Technical Specifications. This modification meets the existing PASS requirements to NUREG-0737, Quality Group D and ANSI B31.1.

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 the probability for an accident or malfunction of a different type than any evaluated may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The PASS Dissolved Hydrogen Analyzer installation included in this Engineering Package does not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification because it does not involve any safety related equipment and therefore has no effect on the availability, redundance, capability, 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 Safety Analysis Report will not be created since no new failure modes are introduced and there is no potential for interaction with any safety related equipment/ function. Furthermore, this modification meets the existing PASS requirements to NUREG-0737, Quality Group D and ANSI B31.1. 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 since this modification to the Post Accident Sampling System does not affect any technical specifications.

SAFETY EVALUATION (Continued)

The Post Accident Sampling System (PASS) is not required for safe shutdown of the plant and is not required to meet the requirements of Seismic Category I. However, this PCM is Quality Related because it enhances the existing system and is designed to meet the requirements of NUREG-0737 for monitoring RCS chemistry and activity resulting from a Design Basis Accident.

The implementation of this EP does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question.

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

PC/M 100-287 Rev. 0-1

ABSTRACT

This Engineering Package deals with modifications to electronic components in the Control Element Drive Mechanism Control System (CEDMCS). The modification results in an additional Control Room annunciation for continuous CEDM coil high voltage. The implementation of this change did not create an unreviewed safety question per the requirements of 10CFR 50.59 or require prior NRC approval since the CEDMCS is not a safety related system. The original design bases for the CEDMCS is not changed by this modification and no effect on any plant Technical Specification is created. This modification does not delete any alarm signals or interlock functions and does not alter the operational parameters of the CEDMs.

SAFETY EVALUATION

All of the modifications implemented by this Engineering Package are confined to the CEDMCS cabinets. The changes consists of: (1) replacing the existing CEA timer boards with Automatic CEDM Timer Module (ACTM) boards; (2) rewiring the LED driver and TFA latch boards to provide separate Time Failure Alarm (TFA), High Voltage Alarm (HVA), and Hold (HLD) signals; (3) modifying the HOLD bus power supplies so that momentary high voltage can be applied to the CEDM coils when transferring to the HOLD bus; (4) installing non-intrusive current sensors on the CEDM coil leads to provide input signals to the new ACTM boards; (5) replace the existing logic power supplies with similar supplies that have a greater output voltage adjustment range; (6) removal of the latch function from the HOLD (HLD) signals.

10CFR50.59 allows a licensee to modify plant hardware, operations, tests, or procedures without prior NRC approval as long as the modification does not involve (1) a revision to the Technical Specifications or (2) an unreviewed safety question. 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.

As stated in Section 2.1.2, the original design bases for the CEDMCS is not altered by this modification.

SAFETY EVALUATION (Continued)

Modifications are being made to only the CEDMCS with no change being made to the Reactor Trip Switch Gear or to the interface between the CEDMCS and the Reactor Trip Switch Gear. Also, the changes implemented by this Engineering Package do not delete any alarm signals or interlock functions. It does increase surveillance of the CEDMs by the addition of the HVA coil high voltage alarm. Since operational parameters of the CEDMs are not altered by the modifications in this Engineering Package, no revision to the Technical Specification is required.

As discussed in Section 2.1, the CEDMCS is a non-safety related system. Also, Section 2.1.2 states that the original design bases for the CEDMS is not altered by this modification. Therefore, the probability of occurrence or the consequences of previously analyzed accidents do not increase because of this modification and the present Final Safety Analysis Report (FSAR) conclusions remain valid. Since the CEDMS is a non-safety related system and the original design bases for the CEDMS is not altered by this modification, there exists no possibility of creating an accident of a different type than analyzed in the FSAR.

The CEDMCS is a non-safety related systems and this modification will not alter the original design bases for the CEDMCS, as established in the Unit 2 FSAR and Technical Specifications, the margins of safety as defined in the bases for any technical specifications are not decreased.

In conclusion, implementation of this Engineering Package will not involve a change in the Technical Specifications incorporated in the St Lucie Unit 2 license and does not involve an unreviewed safety question. Therefore, adequate protection of the public health and safety can be assured and this Engineering Package can be implemented without prior Commission approval.

PC/M 373-288 Rev. 0

ABSTRACT

This Engineering Package provided for modifications to the control circuits and associated alarm for the Fuel Handling Building Cask Area Shield Door (L-Shaped Door). Changes to the circuitry involve inhibiting door motion when the inflatable seal is inflated, preventing seal deflation upon restoration of power following loss of power, and correcting a loss of power nuisance alarm when power is available but the security interlock is disabled. The modifications implemented by this Engineering Package did not create an unreviewed safety question per the requirements of 10CFR 50.59 or require prior NRC approval since the function or capability of the L-Shaped Door to minimize leakage in the event of a Fuel Handling Accident has not been altered. The L-Shaped Door and the control circuits serve no safety related function and are not part of any plant Technical Specification.

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 the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This Engineering Package (EP) provides the engineering and design necessary to modify the FHB L-Shaped door control circuits and associated alarm to (1) inhibit door motion when the seal is inflated, (2) prevent seal deflation upon restoration of power following a loss of power, and (3) correct nuisance alarm indicating a loss of power when the security interlock is disabled. The RTGB annunciator window engraving will also be modified.

The modifications included in this Engineering Package do not involve an unreviewed safety question 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 Fuel Handling Building L-Shaped Door neither directly not indirectly serves any function required to bring the plant to safe shutdown. The ability of the FHB L-Shaped door to minimize leakage in the event of a Fuel Handling Accident as discussed in the Final Safety Analysis Report (FSAR) has not been reduced.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no changes have been made to the operational design of any control circuits or associated systems which are important to safety.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification since the Fuel Handling Building L-Shaped door control circuits do not perform a nuclear safety related function. The door and its associated control circuits do not form the basis for any Technical Specification.

This package is classified as Quality Related because the FHB L-shaped door serves to minimize the leakage of possible radiological releases during of a Fuel Handling Accident.

Due to the fact that this EP does not involve any cables essential to safe reactor shutdown or systems associated with achieving and maintaining safe shutdown conditions, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore, the proposed design of this package is in compliance with the applicable codes and St Lucie Unit 2 FSAR requirements for fire protection equipment.

Implementation of this Quality Related PCM does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question nor a change to the Plant Technical Specifications; thus, prior NRC approval for the implementation of this PCM is not required.

ABSTRACT

This Engineering Package provides for the safety evaluation of the St. Lucie Unit 2 Cycle 6 fuel reload design. The safety evaluation included a re-evaluation, by the NSSS vendor, of the fuel reload design analysis. The reanalysis shows that the Cycle 6 operation is bounded by previous analyses in that the consequences of an accident or malfunction have not been increased since all the transients meet current criteria. An evaluation and review of the St. Lucie Unit 2 Chapter 15 events was performed and it was determined that the inputs to most Cycle 6 events were bounding. The other events were reanalyzed and the results show that in every case, the respective reference analysis bound the Cycle 6 specific results. Fuel reload designs are classified as safety related since the designs provide the analysis for shutdown of the reactor and demonstrate the capability of maintaining the reactor in a safe shutdown condition. The NSSS vendor analysis determined that these changes can be implemented with no changes required to the plant Technical Specifications and that no unreviewed safety question per 10 CFR 50.59 is created. Therefore, prior NRC approval was not required. Results of reanalysis of transients and accidents provide results that meet acceptance criteria.

SAFETY EVALUATION

Based on the technical evaluation and the results of the reanalysis included in the Reload Safety Evaluation report, it is concluded that the Cycle 6 reload design meets all design criteria, it bounded by the results of the referenced analyses, and can be implemented with no changes required to the existing St Lucie Unit 2 Technical Specifications. Therefore, it can be stated that:

(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.

The St Lucie Unit 2 Cycle 6 reload design does not change the overall configuration of the plant. The mode of operation of the plant remains unchanged. Therefore, the probability of occurrence of an accident or malfunction, previously evaluated in the safety analysis report, is not increased. The RSE report (Reference 2) demonstrates that the consequences of an accident or malfunction have not been increased beyond these evaluated in the previous analyses since all the transients meet current criteria.

(ii) A possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created.

The St Lucie Unit 2 Cycle 6 reload design does not change the overall configuration of the plant. The mode of operation of the plant remains unchanged. Therefore, a possibility for a new accident or equipment malfunction has not been created.

(iii) The margin of safety as defined in the basis for any Technical Specification is not reduced.

SAFETY EVALUATION (Continued)

FPL performed an evaluation and review of the St Lucie Unit 2 Chapter 15 events to verify that the inputs to the safety analyses and the results are bounding for Cycle 6 applications. Based on this evaluation it was determined that inputs to all Cycle 6 events were bounded except the following; the Increased Main Steam Flow, the Pre-Trip Steam Line Break pin census, the Uncontrolled Control Element Assembly Withdrawal from Subcritical or Low Power Condition, and the Small Break LOCA event. The results of these reanalyses given in the RSE report (Reference 2) show that in each case, the respective reference analysis bound the Cycle 6 specific results. Therefore, there is no reduction in the margin of safety relative to the Technical Specification basis.

As per Federal Regulation 10CFR50.59 this activity does not involve an unreviewed safety question and does not require a change to Technical Specifications, therefore, implementation of this reload is permissible without prior NRC approval.

PC/M 399-988 Rev. 0

ABSTRACT

This Engineering Package provided for the installation of two underground tanks to be used as a Fuel Dispensing Facility for the St. Lucie site. These tanks are located at a sufficient distance from any safety related equipment or systems such that any failure at the facility will have no effect on any plant system. This facility serves no safety related function and the addition of this facility did not create an unreviewed safety question per the requirements of 10CFR 50.59 or require prior NRC approval. The plant Technical Specifications were not affected by the installation or operation of this facility.

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 the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced. The modifications included in this engineering package do not involve an unreviewed safety question because of the following reasons:

(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 (see Attachment 7.3, prepared by Ebasco Services, Inc.). 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 can 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.

SAFETY EVALUATION (Continued)

(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 (Attachment 7.3) in accordance with the St Lucie Unit 1 FSAR, the St Lucie Unit 2 FSAR, and 10CFR50 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 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 directed modifications to the discharge canal nosing area to improve the drainage characteristics. The purpose is to provide for the better direction of surface runoff water to catch basins and/or the plant storm drain system. The systems and structures affected by this modification are not safety related and have no impact upon any safety related systems or functions, therefore per the requirements of 10CFR 50.59 an unreviewed safety question is not created and prior NRC approval was not required. The components modified by this Engineering Package do not form or are not included in the basis of any plant Technical Specification.

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 the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced. The modifications included in this engineering package do not involve an unreviewed safety question because of the following reasons:

(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. The enhancements to the discharge canal nosing area drainage system do not involve any equipment or functions related to nuclear safety, nor will its failure increase the probability of an accident since they are not located in the vicinity of any safety-related equipment. Furthermore, the affected components and systems are not a factor in mitigating the consequences of any accidents, therefore the implementation of this modification cannot increase the consequences of any design basis event.

(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 systems and failure of any items added by this modification will not impact any nuclear safety-related functions. Since there are no safety-related systems in the vicinity of the modifications, no conceivable construction mishap can affect nuclear safety.

SAFETY EVALUATION (Continued)

(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 Specification.

This engineering package enhances the drainage system to more effectively collect surface runoff water as was intended in the original design. No additional volume of water beyond what was originally designed for will be collected by the system. The systems affected by this modification do not perform any safety-related functions and are not in the vicinity of any safety-related equipment. Therefore there can be no impact to nuclear safety during construction or operation. Accordingly, this engineering package is classified as non-nuclear safety-related.

The implementation of this engineering package does not require a change to the Plant Technical Specifications nor does it create an unreviewed safety question.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question nor a change to the Plant Technical Specifications and prior NRC approval for the implementation of this modification is not required.