FPL

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L-2007-167
10 CFR 50.59(d)

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

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

Pursuant to 10 CFR 50.59(d)(2), the attached report contains a brief description of any changes, tests, and experiments, including a summary of the 50.59 evaluation of each which were made on Unit 1 during the period of December 19, 2005 through May 27, 2007. This submittal correlates with the information included in Amendment 22 of the Updated Final Safety Analysis Report to be submitted under separate cover.

Please contact us should there by any questions regarding this information.

Very truly yours,

Gordon L. Johnston
Site Vice President
St. Lucie Plant

GLJ/tlt

Attachment

IE47
NRR

ST. LUCIE UNIT 1
DOCKET NUMBER 50-335
CHANGES, TESTS AND EXPERIMENTS
MADE AS ALLOWED BY 10 CFR 50.59
FOR THE PERIOD OF
DECEMBER 19, 2005 THROUGH MAY 27, 2007

INTRODUCTION

This report is submitted in accordance with 10 CFR 50.59 (d)(2), which requires that:

i) changes in the facility as described in the SAR;
ii) changes in procedures as described in the SAR; and
iii) tests and experiments not described in the SAR

that are conducted without prior Commission approval be reported to the Commission in accordance with 10 CFR 50.90 and 50.4. This report is intended to meet these requirements for the period of December 19, 2005 through May 27, 2007.

This report is divided into three (3) sections. First, changes to the facility as described in the Updated Final Safety Analysis Report (UFSAR) performed by a Plant Change/Modification (PC/M). Second, changes to the facility/procedures as described in the UFSAR, or tests/experiments not described in the UFSAR, which are not performed by a PC/M. And third, a summary of any fuel reload 50.59 evaluation.

Each of the documents summarized in Sections 1, 2 and 3 includes a 10 CFR 50.59 evaluation that evaluated the specific change(s). Each of these 50.59 evaluations concluded that the change does not require a change to the plant technical specifications, and prior NRC approval is not required.

St. Lucie Unit 1
Docket No. 50-335
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TABLE OF CONTENTS

SECTION 1

PLANT CHANGE / MODIFICATIONS

PLANT CHANGE/MODIFICATION 04059

REVISION 3

FEEDWATER CONTROL REPLACEMENT - PHASE 2

Summary:

This modification replaces the existing Feedwater Flow and Steam Generator Level Controls, the Reactor Coolant Pump (RCP) monitoring and display system, and the Atmospheric Dump Valve (ADV) controls with new Distributed Control System (DCS) equipment and software. Under PC/M 03068, a DCS, along with a Plant Data Network (PDN), were installed to replace the Digital Data Processing System (DDPS) and the Sequence of Events Recorder (SOER). Therefore, for this modification the existing DCS was expanded to include the FW, ADV and RCP systems. The new DCS equipment will also interface with the installed PDN.

In addition to the controller replacement, the positioner components of the Main Feedwater Regulating Valves (MFRV, FCV-9011 and FCV-9021), the Low Power Feedwater 15% Bypass Control Valves (LCV-9005 and LCV—9006), and the Atmospheric Dump Valves (ADV, HCV-08-2A and HCV-08-25) were replaced with digital positioners by this modification. The Manual/Auto (M/A) stations (FIC—9011/9021, LIC—9005/9006, and PIC—08-lA/1B) for these valves were also replaced. Ten (10) Reactor Turbine Generator Board (RTGB) 102 indicators and four (4) RTGB 102 recorders associated with the Feedwater and ADV System and 52 RCP indicators on RTGB 103 were replaced with Flat Panel Displays (FPDs). Additionally, inputs from the RCP vibration monitoring system on the back of RTGB 104 were brought into the DCS and will be displayed on the FPD on the front of RTGB 103. A Display Driver Cabinet was installed to drive the FPDs. Additionally, the four SG level bypass key lock selector switches located on the front of RTGB 102 were relocated to the inside of RTGB 102 to make room for the FPDs.

The existing power configuration for the FW and the ADV controls was modified slightly, moving the backup power off of the interruptible power supplies, PPl01/102A, onto 125VDC Buses 118 and 119 for Appendix R requirements. Additionally, isolation fuses are required for Appendix R isolation; however there is no additional space to mount a fuse block within the existing PP—119 Isolation Panel. Therefore, an additional fuse isolation panel, PP-119 Isolation Panel #2, was installed.

The reason for this modification was to reduce the instability (oscillations) of the Feedwater Regulating System (FWRS) and to upgrade this control system, along with the Atmospheric Dump Valve controls and the RCP monitoring and display system, to a DCS. The existing MFRV, Low Power Feedwater 15% Bypass Control Valve, and ADV pneumatic positioners were replaced with digital positioners to enhance tuning and diagnostic capability. Additionally, the existing DCS was migrated to new software Version 8.0 to improve system performance. Copper blades on zone switches 1Z01 and 1Z04 were changed to fiber blades to support testing and future expansion of the PDN.

SECTION 2

50.59 EVALUATIONS

St. Lucie Unit 1
Docket No. 50-335
L-2007-167 Attachment

EVALUATION SENS-07-007
REVISION 0

ROOF VENTILATOR RV-4 FOR ELECTRICAL EQUIPMENT ROOM 1A EXHAUST OUT OF SERVICE FOR MORE THAN 90 DAYS.

Summary:

In November 2006, the Mechanical Maintenance department was assigned to troubleshoot the rough running fan associated with roof ventilator RV-4. The equipment clearance order (ECO) was executed on November 17, 2006, at 0819 hours. When the cover was removed from the fan, extensive damage was found to the fan blades. Two of the four blades completely fractured apart and fell into the cable tray below. The damaged fan assembly was removed and taken to the maintenance shop for disassembly. The two remaining fan blades had linear indications around the blade rivets. In accordance with administrative procedure ADM-09.08, "Operations In-Plant Equipment Clearance Orders," if a 50.59 Screening was not previously performed, the ECO is in place under the expectation that the system alteration would not be in effect for greater than 90 days during at-power operations. Repair of RV-4 was scheduled to begin on February 26, 2007. Section 6.15.1.B.2 of ADM-09.08 requires a 50.59 Screening for those ECOs that have been in place for longer than 60 days. The 50.59 Screening determined that the temporary alteration of the electrical equipment room 1A ventilation exhaust in support of maintenance on RV-4 adversely affected its UFSAR design function. Therefore, this 10 CFR 50.59 Evaluation was written for the temporary alteration.

EVALUATION SENS-07-018
REVISION 1

OPERATION WITH THE POTENTIAL FOR AN ICI THIMBLE-TO-TSP FLANGE CAP SCREW BECOMING A LOOSE PART

Summary:

During the cutting operation associated with the In-Core Instrumentation (ICI) Thimble Replacement Project (PC/M 04101), inspection found that twelve (12) thimble tube stubs have a loose ICI Thimble-to-Thimble Support Plate (TSP) joint; i.e., able to be rotated about their centerline or able to be displaced laterally or vertically relative to their installed position. This condition was determined to be the result of improper seating of the ICI hold-down assemblies to the TSP during the installation of modified hold-down assemblies in the 1978-1990 timeframe. The improper seating resulted in gaps between the hold-down flanges and the TSP, which resulted in the loose thimble-to-TSP joints.

An evaluation was completed by Westinghouse which evaluated the safety significance of operating Unit 1 with the loose ICI thimbles. The evaluation assessed potential impacts resulting from the loose ICI thimbles and concluded that the only potential impacts of concern were that vibration of the thimble-to-TSP joint could cause the flange cap screws to back out and become a loose part. Vibration could cause thimble or thimble flange failure that could result in the ICI thimble dropping down. Failure of the ICI thimble could then result in loss of ICI detector assembly operation.

PSL Engineering review of the Westinghouse evaluation concluded that some of the potential consequences of the above issue constituted adverse affects. As a result, PSL Engineering concluded that operation with the potential for a cap screw becoming a loose part involved a change to an SSC which could adversely affect an UFSAR-described design function. Therefore, a 10 CFR 50.59 Evaluation was written to document that prior NRC approval was not required to operate Unit 1 with the potential for an ICI thimble-to-TSP flange cap screw becoming a loose part.

EVALUATION SECJ-04-064
REVISION 1

CASK CRANE MAIN HOIST WEIGHT LIMIT INCREASE TO 150 TONS

Summary:

The cask handling cranes at St. Lucie Units 1 and 2 have been replaced with new single-failure-proof cask cranes of greater capacity. The new cranes have a 150 ton single-failure-proof main hoist and a conventional 25 ton auxiliary hoist. The purposes of this evaluation are to: (a) increase the main hoist (cask crane main hoist, main hook, supporting structures and systems) load limit to the maximum critical load capacity of the new main hoist (150 tons), (b) expand the main hoist safe load path to include all the area that is accessible to the main hoist, and (c) allow the main hoist to place the cask lid on the cask (including lowering the lid onto the cask and/or lifting the lid off the cask) when the cask contains spent fuel.

This is not a change in the capacity of the current (new) cask handling cranes. It replaces the previous (old) crane load limits with the current crane maximum critical load capacity. The load limits for the previous cranes, which were not single-failure-proof, were 25 tons for Unit 1 and 100 tons for Unit 2. The maximum critical load capacity for the new cranes, which are single-failure-proof, is 160 tons.

SECTION 3

RELOAD EVALUATION

PLANT CHANGE/MODIFICATION 05060

REVISIONS 0 & 1

ST. LUCIE UNIT 1 CYCLE 21 RELOAD

Summary:

This engineering package (EP) provides the reload core design of St. Lucie Unit 1 Cycle 21 developed by FPL Company and AREVA-NP, Inc. The Cycle 21 core is designed for a nominal cycle length of 12,130 EFPH, based on a nominal Cycle 20 length of 11,100 EFPH. The Cycle 21 reload design supports an end-of-cycle Tavg coastdown at full power with a maximum reduction in primary coolant temperature of 26°F.

The primary design change to the core for Cycle 21 is the replacement of 73 irradiated fuel assemblies with 72 fresh Region CC fuel assemblies and 1 irradiated fuel assembly (Region 5) currently residing in the spent fuel pool. All assemblies in the Cycle 21 reload core are of the debris-resistant design. The fuel assembly design of Region CC fuel is the same as that of the previous cycle, Region BB fuel design. This fuel design includes the use of high thermal performance (HTP) spacer grids and the use of FuelGuard lower tie plate. The fuel assembly design for Region CC fuel utilizes radial enrichment zoning similar to that used in Region BB fuel. A new 20 Gad pattern is used in Sub-region CC5 (4 assemblies).

The implementation instructions provided in this EP, for core reconfiguration from Cycle 20 to Cycle 21, support a full core offload only. This EP supports offload initiation as early as 145 hours after Cycle 20 shutdown if the temperature of the component cooling water (CCW) inlet flow to the spent fuel pool heat exchanger is less than or equal to 95°F. If CCW inlet flow temperature exceeds 95°F, initiation of core offload is restricted to equal to or greater than 168 hours after shutdown.

The safety analysis for Cycle 21 reload design was performed by AREVA-NP and by FPL using NRC approved methodology. The analyses for Cycle 21 reload support a Departure from Nucleate Boiling Ratio (DNBR) limit at the 95/95 probability/confidence level, consistent with the applicable DNB correlation previously approved by the NRC. The linear heat rate (LHR) corresponding to the fuel centerline melt limit for Cycle 21 is 23.67 kW/ft. All analyses in support of this EP were performed with the assumption of steam generator tube plugging level not to exceed 15% average, and with a maximum asymmetry of ±7% about the average.

During the Unit 1 Cycle 20 refueling outage, it was discovered that excore detector #7 was not in its normal operating position. The as-found position of detector #7 was approximately 10 degree azimuthal and 3 inch radial shift away from its normal operating position. This EP supports the continued operation with linear range detector # 7 in the as-found position. Additionally, this EP supports the replacement of linear range detector #8 with a new shorter design of the same design used in Unit 2 in Cycle 15; and the implementation of new design shorter ICIs and ICI thimbles.

Revision 1 of this EP documents the implementation of the Startup Test Activity Reduction (STAR) program for St. Lucie Unit 1, starting in Cycle 21, to allow an option to eliminate rod worth measurements during zero power physics test, following refueling. The supporting evaluation for the EP is based on the STAR program approved by the NRC for the participating Combustion Engineering (CE) designed pressurized water reactors. The objective of the STAR program is to reduce startup testing operations while ensuring that the core can be operated as designed, and startup evolution following STAR would use normal operating procedures. Although STAR allows the elimination of rod worth measurements and moderator temperature coefficient (MTC), this revision is only for the elimination of rod worth measurements.