UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

William M. Dean, Director

In the Matter of	)	Docket No. 50-389
	)	
Florida Power & Light Company	)	License No. NPF-16
	)	
St. Lucie Plant, Unit No. 2	)	
	)	

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

By petition dated March 10, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14071A431), as supplemented,[1] Southern Alliance for Clean Energy (SACE or the petitioner) requested a hearing on what the petitioner characterized as a *de facto* license amendment for the replacement of the steam generators (SGs) in 2007 at the St. Lucie Plant, Unit No. 2 (SL-2), under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, "Changes, tests and experiments." SACE requested that the U.S. Nuclear Regulatory Commission (NRC or the Commission) revoke the *de facto* license amendment and stay the restart of SL-2 from the March 3, 2014, refueling outage pending resolution of the hearing request. As the basis for this request, the petitioner stated that Florida Power & Light Company (the licensee) misapplied 10 CFR 50.59 and that the SG replacement should have

[1] Supplements (ADAMS Accession Nos. ML14071A431, ML14115A457, ML14115A458, ML14125A514, ML14128A557, ML14143A412, ML14147A523, ML14310A811, and ML14337A792)

required a license amendment.[2] The petitioner also expressed concerns (1) related to the inspection of the replacement SGs, and (2) regarding the effects of the extended power uprate (EPU) on SG tube inservice inspection and flow-induced effects on the SG internals.

The Commission, by a memorandum and order (CLI-14-04) dated April 1, 2014 (ADAMS Accession No. ML14091B118), denied SACE's request to stay the restart of SL-2 from the March 3, 2014, refueling outage. Subsequently, by a memorandum and order (CLI-14-11) dated December 19, 2014 (ADAMS Accession No. ML14353A114), the Commission denied SACE's hearing request, concluded that the NRC did not issue the licensee a *de facto* license amendment, and referred SACE's safety concerns regarding the replacement SGs at SL-2 to the NRC's Executive Director for Operations for disposition under 10 CFR 2.206, "Requests for action under this subpart." Therefore, the staff treated these concerns in SACE's hearing request as a petition for enforcement action pursuant to 10 CFR 2.206. On February 24, 2015 (ADAMS Accession No. ML15057A221), and August 5, 2015 (ADAMS Accession No. ML15217A443), SACE informed the NRC staff that it had decided not to request a meeting with the NRC's Petition Review Board with regard to its 10 CFR 2.206 petition.

By letter dated September 28, 2015 (ADAMS Accession No. ML15205A313), the NRC acknowledged receipt of SACE's 10 CFR 2.206 petition and notified SACE of the NRC's acceptance of a portion of the petition (i.e., one of SACE's safety concerns) for review in the 10 CFR 2.206 process. The portion of the petition that the NRC accepted for review under the 10 CFR 2.206 process addresses the licensee's application of 10 CFR 50.59 with respect to the change in a methodology for evaluating SGs, as described in the updated final safety analysis

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[2] Regulations in 10 CFR 50.59 set forth the circumstances under which a licensee can make changes to a facility as described in its UFSAR, make changes in the procedures described in the UFSAR, and conduct tests or experiments not otherwise described in the UFSAR without obtaining a license amendment.

report (UFSAR). The letter also stated that the NRC staff was evaluating whether the licensee properly applied 10 CFR 50.59 when it changed the structural analysis codes as described in the UFSAR.

The staff's September 28, 2015, letter explained why the NRC did not accept the remaining portion of the petition for review under the 10 CFR 2.206 process. This portion of the petition raised safety concerns related to (1) inspection of the replacement SGs, and (2) the effects of the EPU on SG tube inservice inspection and flow-induced effects on the SG internals. These concerns met the criteria for rejection in NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions," dated October 25, 2000 (ADAMS Accession No. ML041770328), because the concerns had already been reviewed, evaluated, and resolved by the NRC staff. The following paragraphs describe the NRC staff's prior resolutions of these concerns.

Regarding the first concern related to the inspection of replacement SGs, NRC inspectors reviewed several aspects of the replacement SGs at SL-2 under Inspection Procedure 50001, "Steam Generator Replacement Inspection," dated September 6, 2000 (ADAMS Accession No. ML003754462). The inspection scope included the following related to the replacement SGs:

- design and planning;
- removal and replacement;
- preservice and baseline inspections;
- welding and nondestructive examination;
- quality assurance program and corrective actions; and
- post-installation verification and testing.

This inspection also covered a review of the plant change modification packages and licensee procedures to design and replace the SGs. The NRC inspectors did not identify any findings of significance during their inspection. The inspection is documented in Section 4OA5.3 of the

NRC's Integrated Inspection Report No. 05000389(335)/2007005, dated February 1, 2008 (ADAMS Accession No. ML080350408).

Regarding the second concern related to the effects of the EPU on SG inservice inspections and flow-induced effects on the SGs, NRC staff reviewed and approved the SL-2 EPU amendment on September 24, 2012 (ADAMS Accession No. ML12268A167). The licensee's application for the EPU included evaluations of the replacement SGs, including inservice inspections and flow-induced effects. In its review, the NRC staff determined that the effects of the proposed EPU at SL-2 did not adversely affect the structural integrity of the replacement SGs and that the licensee had identified appropriate degradation management inspections.

The Advisory Committee on Reactor Safeguards (ACRS) also reviewed the SL-2-EPU application with respect to SG performance. By letter dated July 23, 2012 (ADAMS Accession No. ML12198A202), ACRS evaluated the licensee's root cause of SG tube wear indications and the licensee's action plan to address SG tube integrity. ACRS determined that the licensee's action plan adequately addressed the concerns about SG tube integrity.

By letters to the petitioner and licensee dated May 24, 2016 (ADAMS Accession Nos. ML16055A311 and ML16055A330, respectively), the NRC issued the proposed director's decision (ADAMS Accession No. ML16055A284) for comment. The petitioner and the licensee were asked to provide comments within 15 days on any part of the proposed director's decision considered to be erroneous or any issues in the petition that were not addressed. The NRC staff did not receive any comments on the proposed director's decision.

II. Discussion

Under 10 CFR 2.206(b), the director of the NRC office with responsibility for the subject matter shall either institute the requested proceeding or advise the person who made the request in writing that no proceeding will be instituted, in whole or in part, with respect to the request, and the reason for the decision. Accordingly, the decision of the Director of the Office of Nuclear Reactor Regulation is provided below.

Regulatory Background

Regulations in 10 CFR 50.59 require licensees to determine if any changes to their facilities or procedures described in the UFSAR, or tests or experiments not described in the UFSAR, will need prior NRC approval through a license amendment. An NRC-approved license amendment is required if the changes, tests, or experiments involve a change to the technical specifications or if they meet any one of the eight criteria in 10 CFR 50.59(c)(2). A 10 CFR 50.59 evaluation typically refers to a licensee's documented evaluation against the eight criteria in 10 CFR 50.59(c)(2) to determine if a proposed change, test, or experiment requires prior NRC approval through a license amendment under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit."

Disposition of Previously Unresolved Safety Concerns

As documented in Section 4OA2.4 of the NRC's Integrated Inspection Report No. 05000389(335)/2014005, dated January 30, 2015 (ADAMS Accession No. ML15030A323), the NRC staff opened an "unresolved item" that discussed the staff's plans to review the specific design and qualification approach for the SL-2 replacement SG tube-to-tubesheet joint.
In December 2015, the NRC staff finished its inspection activities for the unresolved item. The inspection results are documented in Section 4OA2 of the NRC's Integrated Inspection Report No. 05000389(335)/2015004, dated February 5, 2016 (ADAMS Accession No. ML16036A156). In resolving the unresolved item, the NRC inspectors identified the following two issues:

(1) a failure to verify the adequacy of the SL-2 replacement SGs tube-to-tubesheet weld design, and

(2) an inadequate 10 CFR 50.59 evaluation for the SL-2 SG tube-to-tubesheet welds.

As described below, the inspectors determined that the first issue was a violation of quality assurance requirements for design control in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criterion III, “Design Control,” while the second issue was a minor violation of 10 CFR 50.59.

(1) Failure to Verify Adequacy of Replacement SG Tube-to-Tubesheet Weld Design

In 2007, the licensee replaced the SGs in SL-2. As described in the January 30, 2015, inspection report, the inspectors identified a potential difference in design approaches between the original and replacement SGs for SL-2. In response, the licensee entered the issue into its corrective action program and determined that the original SL-2 SGs were designed with a tubesheet joint with tube-to-tubesheet welds, considered as structural welds, to function as the tube-to-tubesheet joint pressure boundary. The replacement SGs were designed with a tubesheet joint that relies on the tube radial expansion against the tubesheet to function as the tube-to-tubesheet joint pressure boundary. However, the tubesheet joint still has a tube-to-tubesheet weld that is classified as a seal weld, not a structural weld, and was not relied on to create the tube-to-tubesheet joint pressure boundary.

Based on the design information made available by the licensee, the inspectors determined that the licensee did not perform the necessary American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) analyses to support the pressure-retaining (or structural) function of the welds, as in the design basis of the original SGs. Specifically, the licensee did not perform the primary stress analyses of the tube-to-tubesheet welds in the SL-2 replacement SGs to verify that the design-basis loads would not result in stresses beyond the limits established in the ASME Code. The failure to perform the stress analyses for the SL-2 replacement SGs, in accordance with the ASME Code, was attributed to the failure of the licensee's design review process to verify that the replacement SG tube-to-tubesheet welds were designed as pressure-retaining welds, with the corresponding analyses, and consistent with the design basis of the original SGs.

As part of its corrective actions, the licensee performed primary stress analyses for the SL-2 replacement SG tube-to-tubesheet welds, consistent with what was done for the original SGs. The analyses demonstrated that the applicable ASME Code stress limits were satisfied under design-basis conditions. Therefore, structural integrity of the welds was demonstrated, consistent with the design basis for the original SGs, and the tube-to-tubesheet welds could be considered as structural welds.

The licensee determined that the 10 CFR 50.59 evaluation performed in support of the replacement SGs did not specifically identify and address the change in the design basis of the tube-to-tubesheet joints because the change to the joint design basis was not included in the design technical report reviewed by the licensee. Because the licensee subsequently demonstrated that the tube-to-tubesheet welds in the replacement SGs met the applicable ASME Code requirements and were consistent with the design basis of the original SGs, the conclusions of the original 10 CFR 50.59 evaluations for the replacement SGs were not affected.

The inspectors determined that the licensee's failure to perform the primary stress analyses for the SL-2 replacement SG tube-to-tubesheet welds was a violation of quality assurance requirements for design control in 10 CFR Part 50, Appendix B, Criterion III.
In accordance with the NRC's Enforcement Policy (ADAMS Accession No. ML15029A148), the violation was treated as a non-cited violation because of its very low safety significance.

Because of the very low safety significance of the violation, the NRC does not have a basis for expanding its current level of regulatory oversight in accordance with the agency's Reactor Oversight Process and the Enforcement Policy, or otherwise taking the petitioner's requested enforcement actions against the licensee. The NRC published Regulatory Issue Summary 2016-02, "Design Basis Issues Related to Tube-to-Tubesheet Joints in Pressurized-Water Reactor Steam Generators," dated March 23, 2016 (ADAMS Accession No. ML15169A543), to inform licensees of existing requirements for tube-to-tubesheet welds.

(2) Inadequate 10 CFR 50.59 Evaluation for Tube-to-Tubesheet Welds

The vendor for the SG replacement, AREVA, performed and documented for the licensee a 10 CFR 50.59 evaluation of the SG replacement. In support of the SG replacement, the vendor used computer programs for the structural design of the SGs, including for the design of the tube-to-tubesheet joint welds that were different from the programs described in the UFSAR for the design of the original SGs. The inspectors determined that the computer programs described in the UFSAR were methods of evaluation subject to the provisions of 10 CFR 50.59(c)(2)(viii) and, thus, any changes to these methods would require a written evaluation. However, the inspectors identified that such changes in methods of evaluation were not specifically addressed in the licensee's 10 CFR 50.59 evaluation. The inspectors did not identify any concerns with the application of the remaining seven criteria in 10 CFR 50.59(c)(2)(i-vii) within the scope of their review (i.e., the tube-to-tubesheet joint welds).

The licensee entered the issue regarding the 10 CFR 50.59 evaluation into its corrective action program. As part of its corrective actions, the licensee revised its original 10 CFR 50.59 evaluation to include the evaluation of changes in computer programs used for the structural design of the replacement SGs. The licensee concluded that no departure from a method of evaluation occurred (i.e., the criterion in 10 CFR 50.59(c)(2)(viii) was not met) because the

UFSAR only provided a general functional description of the computer programs used to design the original SGs, and the UFSAR did not explicitly define the calculational framework behind the structural analysis performed by the computer programs. Additionally, the vendor for the replacement SGs stated that its computer programs met the applicable quality assurance program requirements and were benchmarked against classical solutions or other industry-acceptable codes.

In accordance with the NRC's Enforcement Policy, the inspectors determined that the failure to maintain a written evaluation (providing the basis for the determination that a license amendment was not required for changes in computer programs described in the UFSAR) was a minor violation of 10 CFR 50.59. The violation was minor because it involved a change to the UFSAR where there was not a reasonable likelihood that the change would ever require NRC approval per 10 CFR 50.59. Prior NRC approval was not required since the criterion of 10 CFR 50.59(c)(2)(viii) was not met, as the licensee's 10 CFR 50.59 revised evaluation showed that no departure from a method of evaluation occurred when the licensee changed the computer codes in the UFSAR. The inspectors found the licensee's technical justification reasonable and the licensee's revised 10 CFR 50.59 evaluation generally consistent with the guidelines of Nuclear Energy Institute 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," dated November 2000 (ADAMS Accession No. ML003771157), as endorsed by Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59 Changes, Tests, and Experiments," dated November 2000 (ADAMS Accession No. ML003759710).

In summary, the NRC inspectors identified a minor violation of 10 CFR 50.59. In accordance with the NRC Enforcement Policy, minor violations must be corrected; however, given their low safety significance, they are not subject to enforcement action. The licensee corrected the minor violation. In addition, the NRC inspectors determined that no license amendment was required because none of the eight criteria in 10 CFR 50.59(c)(2) was met, and the SG replacement, as related to the tube-to-tubesheet joint welds, did not involve changes to the technical specifications. Therefore, the NRC does not have a basis for expanding its current level of regulatory oversight in accordance with the agency's Reactor Oversight Process and the Enforcement Policy, or otherwise taking the petitioner's requested enforcement actions against the licensee.

III. Conclusion

Based on the NRC's inspection results, as described above, the NRC does not have a basis for taking the petitioner's requested enforcement actions against the licensee. The NRC did not find that the continued operation of the plant would adversely affect the health and safety of the public. Therefore, the NRC denies the petitioner's requested enforcement actions against the licensee.

As provided in 10 CFR 2.206(c), the NRC will file a copy of this director's decision with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision, unless the Commission, on its own motion, institutes a review of the decision within that time.

Dated at Rockville, Maryland, this 8th day of July 2016.

For the Nuclear Regulatory Commission

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William M. Dean, Director,
Office of Nuclear Reactor Regulation