UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 19, 2014

Mr C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
Post Office Box 1295, Bin - 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (VEGP), SAFETY EVALUATION OF RELIEF REQUEST VEGP-ISI-ALT-10, VERSION 1, FOR THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NOS. MF3377 AND MF3378)

Dear Mr. Pierce:

By letter dated January 16, 2014, as supplemented by letter dated February 28, 2014, Southern Nuclear Operating Company, Inc. (SNC) submitted proposed alternative "ISI Program Alternative VEGP-ISI-ALT-10, Version 1" for U. S. Nuclear Regulatory Commission (NRC) review and authorization. SNC proposes to perform the system leakage test of the reactor pressure vessel (RPV) flange O-ring leakoff lines at the Vogtle Electric Generating Plant (VEGP) using the pressure developed when the refueling cavity is filled to the normal refueling water level in lieu of the pressure required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, paragraph IWC-5221. SNC requested authorization to use the proposed alternative pursuant to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50) Paragraph 55a(a)(3)(ii) on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff determined that the proposed alternative, "ISI Program Alternative VEGP-ISI-ALT-10, Version 1," provides reasonable assurance of structural integrity and leak tightness, and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and therefore authorizes use of the proposed alternative at the VEGP, during the Third Inservice Inspection Interval which began May 31, 2007 and is

scheduled to end on May 30, 2017. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

Robert Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure:
Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE VEGP-ISI-ALT-10, VERSION 1, REGARDING SYSTEM LEAKAGE TEST OF REACTOR PRESSURE VESSEL FLANGE LEAK-OFF PIPING

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated January 16, 2014, as supplemented by letter dated February 28, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14016A488 and ML14059A352, respectively), Southern Nuclear Operating Company, Inc. (the licensee) submitted proposed alternative "ISI Program Alternative VEGP-ISI-ALT-10, Version 1" for U. S. Nuclear Regulatory Commission (NRC) review and authorization. The licensee proposes to perform the system leakage test of the reactor pressure vessel (RPV) flange O-ring leakoff lines at Vogtle Electric Generating Plant (VEGP), Units 1 and 2, using the pressure developed when the refueling cavity is filled to the normal refueling water level in lieu of the pressure required by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, paragraph IWC-5221. The licensee requested authorization to use the proposed alternative pursuant to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50) Paragraph 55a(a)(3)(ii) on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), *Inservice Inspection Requirements*, ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components*," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year inspection interval and subsequent 10-year inspection intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein.

Enclosure

Paragraph 55a(a)(3) of 10 CFR 50 states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on analysis of the regulatory requirements, the NRC staff finds that the regulatory authority exists to authorize the licensee's proposed alternative on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(a)(3)(ii).

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request for Alternative

Components for which Relief is Being Requested

VEGP, Units 1 and 2, ASME Code Class 2, 3/8", 3/4", and 1" Nominal Pipe Size (NPS) Reactor Pressure Vessel (RPV) flange leakoff piping originating at the RPV flange and terminating at the last boundary isolation valve (AOV HV8032).

ASME Code Requirements

The code of record for VEGP, Units 1 and 2, Third Inservice Inspection Interval, which began May 31, 2007 and is scheduled to end on May 30, 2017 is the ASME Code, Section XI, 2001 Edition through the 2003 Addenda.

Paragraph IWC-2500 of the ASME Code, Section XI, Table IWC-2500-1, Category C-H, Item Number C7.10, requires that a system leakage test with a VT-2 visual examination of Class 2 pressure retaining components be performed each inspection period. Paragraph IWC-5221 requires that the system leakage test be conducted at the pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability.

Licensee's Proposed Alternative

The licensee proposes to examine the Class 2 portion of the leak detection system consisting of the accessible portions of the RPV head flange O-ring leak-off piping once an inspection period. The leak-off piping shall be examined using the VT-2 visual examination method and will be performed by certified VT-2 examiners. The test shall be conducted at ambient conditions after the refueling cavity has been flooded to its minimum water level for refueling operations of 23 feet above the top of the RPV flange for at least four (4) hours. A static pressure of approximately 10 pounds per square inch, gauge (psig) is expected at a 23 foot depth in borated water.

Licensee's Basis for Requesting Relief

Any significant leakage due to a through-wall leak of the piping or failure of the outer O-ring would be expected to clearly exhibit boric acid residue accumulation that would be discernible during the proposed alternate VT-2 visual examination that will be performed. Additionally, the static head developed with the leak detection line filled with water and the time the line is filled with water will allow for the detection of any gross indications in the line. Performing the pressure test with only static head pressure in the piping does not reduce the margin of safety in operating the plant or detecting leaks. Regardless of how the piping is examined, leakage that could occur during plant operation would still be detected. This conclusion is arrived at by one scenario in which it is assumed the leak-off piping develops a 100 percent through-wall leak.

3.2 NRC Staff Evaluation

The subject lines conduct potential leakage of the inner and the outer RPV head flange O-rings to a thermal detector where the elevated temperature of any leakage is sensed and an alarm in the control room is tripped when a temperature of 170 °F is reached. During normal operation, one line collects potential leakage past the inner O-ring when the valve on the second line, which collects leakage past the outer O-ring, is closed. If there is significant leakage past the inner O-ring, the valve on the line collecting inner O-ring leakage would be closed and the valve on the outer O-ring leakage line would be opened, allowing any potential leakage past the outer O-ring to travel to the thermal detector. As long as there is no indication of leakage past the outer O-ring, the plant is not required to shut down for O-ring leakage. It is only under the condition when the inner O-ring is leaking and the valve on the inner O-ring leakage line is closed will there be significant pressure in the inner O-ring leakoff line.

The subject leakoff lines are heavy walled schedule 160 stainless steel (SA-312, Grade 304L piping and SA-213, Type 304L or 316L tubing) with a plant design pressure of 2485 psig at 680 °F. The reactor operating pressure is 2235 psig. The common header for the leakoff lines is routed to the reactor coolant drain tank. In response to the NRC staff's request for additional information (RAI, ADAMS Accession No. ML14059A352), the licensee states that the plant has not experienced any incidents of significant RPV flange O-ring leakage in its previous years of operation.

The licensee has identified several methods of pressurizing the subject lines to the system pressure required by ASME Code, Section XI, paragraph IWC-5221, prior to performing the required VT-2 visual examination. These methods include: modification of the RPV flange to install mechanical threads and installation of a threaded plug into each leakoff line to establish a boundary for leakage testing; pressurizing the lines after installation of the RPV head at the end of the refueling outage; and pressurizing the lines prior to removing the RPV head at the beginning of the refueling outage.

Modification of the RPV flange to install a threaded plug into each leakoff line would require a design modification to install mechanical threads into each leakoff line at the RPV flange. Threaded plugs would then have to be installed prior to the pressure test and removed after the test was complete. The NRC staff finds that performing the modification, as well as installation and removal of the plugs for each leakage test, would result in significant radiological dose, which would be contrary to As Low As Reasonably Achievable (ALARA) considerations.

Furthermore, installation and removal of the plugs could present foreign material exclusion issues.

Applying system pressure to the leakoff lines for the purpose of system leakage testing with the RPV head installed after refueling would require pressurizing the lines with a hydrostatic test pump in the direction opposite to the intended design function of the O-rings. The NRC staff finds that such pressurization could unseat the installed O-rings, likely resulting in the need to replace the O-rings which would require depressurizing and removal of the reactor vessel head. The licensee states that removal and reinstallation of the head to replace the O-rings would be accompanied by an additional 3 roentgen equivalent man (REM) radiological dose.

In order to pressurize the subject lines without harming new O-rings that would be put into service after refueling, the leakoff lines could be pressurized prior to removing the head for refueling activities. Performing a leakage test prior to the removal of the head would require installation of a vent near the RPV flange to purge the lines of air. The vent installation would require work in an area where radiological dose rates are elevated, approximately 100 milli REM per hour. The NRC staff finds that this evolution would be contrary to ALARA considerations.

The NRC staff has reviewed these options and finds that there is a hardship associated with each. The NRC staff concludes that performing the VT-2 visual examination while the subject lines are at ASME Code-required system pressure would present a hardship without a compensating increase in the level of quality and safety.

The licensee is proposing to conduct a VT-2 visual examination of the leakoff lines after the refueling cavity has been filled to its normal refueling water level for at least 4 hours. The static pressure at the RPV flange due to the refueling water level is approximately 10 psig. In response to the NRC staff's RAI concerning purging the lines of air prior to performing the VT-2 visual examination, the licensee states that the blind flange downstream of the 089 valve will be removed and the 089 valve will be opened, allowing water to flow from the piping for approximately five minutes, to ensure the piping is water solid prior to the beginning of the four-hour pressurization hold time. The NRC staff finds that the procedure is adequate to produce a water-solid line where any leakage would be detected during a VT-2 visual examination. The NRC staff also notes that the flushing procedure will clear the lines of contaminants that could promote stress corrosion cracking. Therefore, the NRC staff finds the flushing procedure acceptable.

The NRC staff notes that the system leakage test requirements of the ASME Code, IWC-5220 are focused on demonstrating leak tightness rather than structural integrity. The NRC staff requested additional information regarding whether the proposed low test pressure would be sufficient to demonstrate the ability of the leakoff lines to perform their required function. If a leakoff line has a large through-wall flaw, leakage would be evident under either a high or low pressure test condition. However, a leak from a small, tight crack may not be evident in the 4 hour time when the piping subjected to the low pressure of the refueling water head. The NRC staff notes that the subject piping is pressurized for several days during each refueling outage when the refueling cavity is filled. Any coolant leakage during either the present or a previous refueling outage would result in boric acid accumulation that would be evident during a VT-2 visual examination. The NRC staff finds that the VT-2 visual examination after the leakoff lines have been subjected to the refueling water head at approximately 10 psig pressure provides evidence of leak tightness, and provides evidence that the lines can transport potential O-ring

leakage to the thermal detector. Therefore, the NRC staff finds that the proposed alternative is acceptable.

The NRC staff recognizes that an opportunity exists to examine the inner O-ring leakoff line while at reactor operating pressure, if there is leakage past the inner O-ring and the valve on the inner O-ring leakoff line is closed. In response to the NRC staff's RAI, the licensee states that there is approximately 16 feet of piping on both units that is visually accessible outside the bioshield and this piping can be visually examined with relative ease while at power operations. The licensee further states that Annunciator Response Procedures 17012-1 and 17012-2 are being revised to have Operations personnel perform a visual examination (non-VT-2) for leakage of the accessible portions of the line ten minutes after line isolation. Following examination by Operations personnel, qualified VT-2 examiners will perform a visual examination of the accessible piping outside the bioshield and will perform a visual examination through the cage doors looking for evidence of leakage utilizing methods described in IWA-5241(b). VT-2 examiners will perform examinations after a minimum hold time of four hours. The NRC staff finds that this procedure to examine the inner O-ring leakoff line while it is pressurized by inner O-ring leakage provides reasonable assurance of leak tightness while maintaining ALARA practices and thus finds it acceptable.

The NRC staff finds, based on evaluation of past performance, as well as the service conditions, and materials of construction, that service induced degradation is unlikely. The NRC staff further finds that if any significant leakage were to occur in the leakoff line during the time of pressurization during each refueling outage, boric acid accumulation would be discernible during a subsequent visual examination. The NRC staff therefore finds that the proposed low test pressure will provide reasonable assurance of the leak tightness of the subject leakoff lines, and will demonstrate that the leakoff lines can perform their intended function. The NRC staff also finds that requiring compliance with the system leakage test pressure requirements would result in a hardship without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative, "ISI Program Alternative VEGP-ISI-ALT-10, Version 1," provides reasonable assurance of structural integrity and leak tightness, and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and therefore authorizes use of the proposed alternative at Vogtle Electric Generating Plant, Units 1 and 2, during the Third Inservice Inspection Interval which began May 31, 2007 and is scheduled to end on May 30, 2017.

All other ASME Code, Section XI requirements for which relief was not specifically requested and authorized in the subject proposed alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Jay Wallace, NRR

Date: March 19, 2014

scheduled to end on May 30, 2017. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Sincerely,

/ra/

Robert Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure:
Safety Evaluation

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