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

July 11, 2012

Mr. Richard L. Anderson
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER – EVALUATION OF RELIEF REQUEST VR-02 REGARDING INSERVICE TESTING OF SAFETY/RELIEF VALVES (TAC NO. ME7336)

Dear Mr. Anderson:

By letter dated September 29, 2011, NextEra Energy Duane Arnold, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for relief from the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), inservice testing requirements for all of the main steam safety/relief valves at the Duane Arnold Energy Center.

Specifically, pursuant to Title 10 of the *Code of Federal* Regulations (10 CFR), Paragraph 50.55a(a)(3)(i), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and finds, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and remains in compliance with ASME OM Code requirements.

This closes the NRC staff's action on the above submittal. If you have any questions, please contact Mr. Karl Feintuch at (301) 415-3079 or via e-mail at Karl.Feintuch@nrc.gov.

Sincerely,

Istvan Frankl, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure: Safety Evaluation

cc w/encl: Distribution via ListServ

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST VR-02 REGARDING THE

INSERVICE INSPECTION PROGRAM FOR MAIN STEAM SAFETY/RELIEF VALVES

NEXTERA ENERGY DUANE ARNOLD, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NUMBER 50-331

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 29, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112720486), NextEra Energy Duane Arnold, LLC (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) inservice testing (IST) requirements for the Duane Arnold Energy Center (DAEC). The licensee requested to use an alternative to testing conditions, locations, and frequencies, pertaining to the testing of the main steam safety/relief valves (SRVs) PSV4400, PSV4401, PSV4402, PSV4405, PSV4406, and PSV4407.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Paragraph 55a(a)(3)(i), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

Relief request VR-02 is applicable to the fourth 10-year IST program interval for DAEC.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(f), "Inservice Testing Requirements," require, in part, that IST of certain ASME Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda. Any proposed alternative must be submitted and authorized prior to implementation pursuant to paragraphs (a)(3)(i) or (a)(3)(ii).

In proposing alternatives, a licensee must demonstrate that the proposed alternative provides an acceptable level of quality and safety (10 CFR 50.55a(a)(3)(i)) or that compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety (10 CFR 50.55a(a)(3)(ii)). Section 50.55a allows the NRC to authorize alternatives to ASME OM Code requirements upon making necessary findings.

The DAEC fourth 10-year IST program interval commenced on February 6, 2006, and ends on February 5, 2016.

3.0 TECHNICAL EVALUATION

ASME OM Code Sections

- The applicable ASME OM Code edition and addenda for DAEC is the 2001 Edition through 2003 Addenda.
- ISTC-3510 [ASME OM Code, Section ISTC-3510], "Exercising Test Frequency," states, in part, that, "Power-operated relief valves shall be exercise tested once per fuel cycle."
- ISTC-5113, "Valve Stroke Testing," requires that active valves have their stroke times measured, and stroke testing shall be performed during normal operating conditions.
- ISTC-5113(e) states that, "Stroke testing shall be performed during normal operating conditions for temperature and pressure if practical."
- ISTC-5114, "Stroke Test Acceptance Criteria," requires that test results be compared to established reference values.
- ISTC-5114(c) states that, "Valves that stroke in less than 2 sec [seconds] may be exempted from ISTC-5115(b). In such cases the maximum limiting stroke time shall be 2 sec."
- Mandatory Appendix I, Section I-1320, "Test Frequencies, Class 1 Pressure Relief Valves," requires that Class 1 pressure relief valves be tested at least once every 5 years.
- Mandatory Appendix I, Section I-3310, "Class 1 Main Steam Pressure Relief Valves with Auxiliary Actuating Devices," specifies tests that are required to be performed on main steam SRVs before and after maintenance and/or set pressure adjustment.
- Mandatory Appendix I, Section I-3410, "Class 1 Main Steam Pressure Relief Valves with Auxiliary Actuating Devices," requires that each valve that has been maintained or refurbished in place, removed from maintenance and testing, or both, and reinstalled shall be remotely actuated at reduced or normal system pressure to verify open and close capability of the valve before resumption of electric power generation.
- ASME OM Code Case OMN-17, "Alternative Rules for Testing ASME Class 1 Pressure Relief/Safety Valves," provides an alternative test frequency for Class 1 pressure relief valves provided that the licensee disassembles and inspects each valve after as-found set-pressure testing to verify that parts are free of defects resulting from timed-related degradation or service-induced wear.

System Description

There are six Target Rock Model 7467F three-stage SRVs installed at DAEC. The SRVs are located between the reactor vessel and the main steam isolation valves. The SRVs have three required functional modes of operation, i.e., the overpressure protection mode, the relief mode actuated by the Automatic Depressurization System (ADS) Logic, and the relief mode actuated by Low-Low Set (LLS) Logic. The various functions of the SRVs are described respectively in Sections 5.2.2, 6.3.2.2.2, and 5.4.1.3, of the DAEC Updated Final Safety Analysis Report.

Safety/Relief Valve Operation

The SRV consists of three main sections – the pilot valve, the remote actuator, and the main valve. Reactor vessel pressure is felt on the top and bottom of the main valve piston, and on the pilot valve bellows via the pilot sensing port. Since reactor vessel pressure is equalized across the main valve piston due to the drilled orifice, it is the main valve spring and differential pressure across the main disc, which keeps the main disc seated. When reactor vessel pressure reaches the self-actuated pressure relief function opening set-point, the pilot valve is forced open against spring pressure, allowing steam to flow into the chamber above the second stage piston. Steam pressure exerted on the second stage piston causes the second stage disc to unseat, providing a relief path for steam above the main disc. Since the steam can escape through the passage in the valve body faster than it can be admitted through the small main piston orifice, pressure above the main piston will decrease. Pressure will continue to decrease until reactor vessel pressure acting on the lower side of the main piston overcomes the spring pressure and forces the main disc off its seat. When reactor system pressure decreases to approximately 50 pounds per square inch (psi) below the self-actuated pressure relief function opening setpoint, the pilot valve will reseat and the main valve spring pressure will reseat the main disc.

Remote actuation of an SRV may be accomplished by its hand-switch located in the Main Control Room, selected switches on the Remote Shutdown panel or by automatic initiation signals (ADS or LLS). Upon receipt of a remote initiation signal from any of these sources, the SRV's solenoid operating valve (SOV) is energized and 90-psi nitrogen pressure is directed to a diaphragm near the top of the SRV. This causes the diaphragm and the attached rod to be forced downward. The rod then makes physical contact with the second stage piston causing the second stage disc to unseat. The remainder of the valve operation is the same as for the self-actuation function previously described. Removal of the remote initiation signal allows nitrogen pressure to vent off of the diaphragm via the exhaust port of the SOV, thus permitting the spring to reseat the main disc.

3.1 Licensee's Proposed Alternatives and Basis

3.1.1 ASME OM Code, Sections ISTC-5113 and ISTC-5114

In its September 29, 2011 submittal, the licensee provided the following alternative and basis for relief from the requirements of ISTC-5113 and ISTC-5114:

The proposed alternatives provide adequate assurance that valve stroke time in the power-actuated mode will be acceptable. Stroke timing of the SRVs will be performed at an offsite test facility. Currently, as-found stroke time testing is performed prior to and after performing maintenance at the test facility. After completion of maintenance, plant surveillance tests with steam at reduced pressure are performed in order to detect gross failures of the SRVs to change position. The surveillance tests performed at DAEC are not as refined as the valve response time test performed at the offsite test facility. The design requirement for the valve stroke time is 0.45 seconds, from signal initiation to valve full open in the power-actuated mode (0.40 seconds for signal initiation from start of valve motion and 0.050 seconds (50 milliseconds) for valve stroke to full open). Measuring valve stroke times to this level of accuracy in-situ at the power plant is not practical and only possible under the controlled conditions of the offsite facility. Per ISTC-5114(c), the maximum permissible valve stroke time can be up to 2 seconds. Consequently, the in-situ test acceptance criterion becomes essentially a "failure to open" criterion. Therefore, the tests performed at DAEC can only detect gross failures to change position and cannot monitor for valve performance degradation between tests.

In-situ stroke timing is not useful for identifying valve degradation over several operating cycles. Rather, an in-situ exercise test will be used to ensure that the valve will function in the power-actuated mode. This test will be performed at the frequency prescribed in ISTC-3510 for power-operated relief valves. Stroke time at the test facility will demonstrate that the valve performs acceptably compared to the stroke times of known good performing valves. Since the test facility can not duplicate the electrical control system at the plant, actuation of the valve at the test facility is accomplished through a simplified electrical actuation. Observation of the end of the operating stroke at the test facility is indirect, based on evidence of steam flow and pressure, as it is at the nuclear facility, since the relief valves have no positive open indication. Although these observations may result in minor differences in measured stroke time compared to those measured when installed in the plant, the stroke times measured at the test facility will be comparable to each other and thus can be used to detect any abnormality in valve performance.

As an alternate to the testing required by ISTC-5113(e), stroke times will be measured at the offsite test facility. Stroke times will also be measured following valve rebuild. The timing will begin with the actuating electrical signal and end with the indirect indication of the end of the operating stroke. Stroke time acceptance criteria will be a pre-established reference value that represents good performance for the valve type. An in-situ exercise test of the valve in the power-actuated mode will be performed at the frequency prescribed in ISTC-3510. The in-situ exercise test will be performed prior to the resumption of electric power generation. Main disc movement and set-pressure verification are not required.

3.1.2 ASME OM Code, Mandatory Appendix I, Sections I-1320 and I-3410

In the same submittal dated September 29, 2011, the licensee also provided the following alternative and basis for relief from the requirements of Mandatory Appendix I, Paragraphs I-1320(a) and I-3410(d):

Exercising of the SRV after reinstallation can only be performed during reactor startup when there is sufficient steam pressure to actuate the main disc. Past history indicates that the main discs may not re-seat properly after being exercised during reactor startup resulting in steam leakage into the suppression pool. SRV leakage increases both pool temperature and level, requiring more frequent use of Residual Heat Removal System to maintain the corresponding temperature limits for the suppression pool in the Technical Specifications (TSs) for the plant. This leakage results in a decrease in plant performance and could force a plant shutdown to repair a leaking SRV. Past operating history indicates that the exercising performed during reactor startup is of no significant benefit in ensuring the proper operation of the individual SRV subassemblies.

This alternative request proposed to implement Code Case OMN-17. Section (a) of OMN-17 requires that safety valves shall be tested at least once every 72 months (six years) with a minimum of 20 percent (%) of the SRV group being tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 72-month interval, if they exist. The test interval for any individual valve that is in service shall not exceed 72 months except that a six-month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods.

The SRV pilot assemblies removed during a refueling outage are tested at an offsite facility. The as-found testing is performed prior to the resumption of power operation from that refueling outage. The valves are refurbished, as necessary, to meet the acceptance criteria of zero leakage, and are certified in writing as being leak free. The valves are then reinstalled in the plant in a subsequent refueling outage and proper pilot operation is confirmed through leak rate testing of the pilot air operators and associated accumulator piping followed by a so-called dry lift test at reactor power. Following reinstallation, the electrical and pneumatic connections will be verified by energizing the SOVs using the respective control switches and inspecting the pneumatic actuator for movement and leakage during a dry lift test. While this test will actuate the SRV second stage, operating experience at other plants indicates that it does not initiate second stage leakage or otherwise damage the valve when performed with no steam pressure; thus, making it a better alternative test to an in-situ steam test during reactor startup.

Several aspects of SRV design and operation can contribute to valve leakage. As mentioned earlier, these include test pressure, pilot valve disc and rod configuration, and overall system and valve cleanliness. Actuation of the SRVs after laboratory testing by any means allows these contributors to impact the ability of the valve to re-close completely. DAEC has made significant efforts to minimize the effects of these contributors. In 1999, the DAEC TSs are being changed to permit an as-found tolerance of ±3% and ±1% as-left tolerance on the SRV opening setpoints. Since that time, the DAEC has not had any SRV setpoint failures and had only one instance of seat leakage during testing at the offsite facility in 2009. There have been two instances of valve leakage during power operation; a pilot valve leak in 2004 and a second stage leak in 2010. This recent event occurred shortly after performing the in-situ test at reduced system pressure and is believed to be a contributing cause of the valve failure.

ASME OM Code, Mandatory Appendix I, Section I-1320 establishes the five year frequency for SRV testing. DAEC proposes to use Code Case OMN-17 to change the frequency of testing to six years, including a six month grace period, to coincide with the 24-month refueling cycle at DAEC. Code Case OMN-17 has been included in the 2009 Edition of the ASME Code.

Mandatory Appendix I, Paragraph I-3410(d) requires manual actuation testing at reduced or normal system pressure following reinstallation after maintenance and/or testing. The 2004 Edition of the ASME OM Code does not require Owners to stroke SRVs at reduced or normal system pressure following maintenance or testing. DAEC proposes to utilize this later edition of the Code to support future testing of the SRVs.

Additionally, reducing challenges to the SRVs is recommended in NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.K.3.16. This recommendation is based on a stuck-open SRV being a possible loss-of-coolant accident. This relief request is consistent with the NRC recommendation.

As an alternate to the testing required by Mandatory Appendix I, Paragraph I-3410(d), DAEC proposes to actuate the SRVs in the relief mode at the certified test facility. A test solenoid valve will be energized, the actuator will stroke, and the 2nd stage rod movement will be verified. This test will verify that, given a signal to energize the solenoid valve, the 2nd stage disc rod will travel to unseat the 2nd stage disc. The 2nd stage function will be recorded in the test documentation package for future reference, as needed. Alternate testing is justified since the remaining segments of the SRV relief mode of operation are verified by other tests. The ability of the pilot disc to open is demonstrated in the safety mode actuation bench test. The integrity of the pneumatic and solenoid system for the SRVs is verified by performance of post maintenance leak rate testing, continuity testing, and a functional testing of the solenoid valve while detached from the SRV. Automatic valve actuation is proven by Logic System Functional Tests which include verification that the SOV is energized by the automatic signal. The actuator to main body joint is inspected during an inservice inspection VT-2 exam performed prior to startup. The above proposed surveillance, testing and inspection of the SRVs and associated components provide reasonable assurance of adequate valve operation and readiness.

DEAC proposes to implement Code Case OMN-17 that requires a 72-month test interval for Class 1 pressure relief valves with a minimum of 20% of the SRV group being tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 72-month interval, if they exist. The test interval for any individual valve that is in service shall not exceed 72 months except that a six month grace period is allowed to coincide with refueling outages to accommodate extended shutdown periods. The removed main steam relief valves will be sent for as-found testing to an off-site test facility. Testing of the valves will be performed as provided in Code Case OMN-17. Each main steam relief valve will then be disassembled and inspected for abnormal wear and the specific concerns documented in General Electric Company Service Information Letters No. 196, Supplement 17 and No. 646 respectively. The post maintenance tests required by Mandatory Appendix I, Section I-3310 will be conducted at the offsite testing facility. As part of implementation of this request, DAEC will institute measures to assure that each main steam relief valve will be disassembled and inspected prior to being placed on the new 72-month interval.

3.2 NRC Staff Evaluation

3.2.1 ASME OM Code, Sections ISTC-5113 and ISTC-5114

In lieu of the ASME OM Code requirements, the licensee proposed to test the SRVs at an off-site steam test facility, and compare the test results to a pre-established reference value of known good performing valves. The NRC staff reviewed the licensee's request and finds that the main valve stroke time is adequately measured and compared to a comparable pre-established reference value of known good performing valves in the proposed alternative. Currently, the ASME OM Code required SRV stroke testing is satisfied by manually stroking open each SRV with the reactor at system operating pressure. As noted in the licensee's submittal, the proposed alternative will allow testing of main valves and pilot valves at an offsite facility. A representative solenoid actuator used at the plant will be installed at the test facility for testing the main valve and pilot valve. The test conditions at the test facility are similar to those at normal operating conditions in the plant. Since the test facility cannot duplicate the electrical control system at the plant, actuation of the valve at the test facility is accomplished through a simplified electrical actuation. Observation of the end of the operating stroke at the test facility is indirect based on evidence of steam flow and pressure as it is at the plant.

Although these observations may result in minor differences in measured stroke time compared to those measured in the plant, the stroke times measured at the test facility, when compared to a pre-established reference value of known good performing valves, will provide indirect but comparable test results to detect changes or any abnormality in valve performance. Therefore, the NRC staff finds that the proposed alternative captures the valve stroke testing and the stroke time test data required by ISTC-5113 and ISTC-5114 of ASME OM Code, and therefore is acceptable.

3.2.2 ASME OM Code, Mandatory Appendix I, Paragraphs I-1320(a) and I-3410(d)

Mandatory Appendix I, Paragraph I-1320(a) requires that Class 1 pressure relief valves shall be tested at least once every five years. Mandatory Appendix I, Paragraph I-3410(d) requires that each valve that has been maintained or refurbished in place, removed for maintenance and testing, or both, and reinstalled shall be remotely actuated at reduced or normal system pressure to verify open and close capability of the valve before resumption of electric power generation. In lieu of the five-year test interval required by Paragraph I-1320(a), the licensee proposes to implement Code Case OMN-17, which allows a test interval of six years plus a six-month grace period. In lieu of testing the SRVs as required by Paragraph I-3410(d), the licensee proposes to test the main valve and pilot valve at a test facility and verify the actuator by separate surveillance tests.

The NRC staff reviewed the licensee's request and finds that the functional capability of the valves is adequately verified with the proposed alternative testing of the SRVs. In lieu of the current ASME OM Code required test, a manual actuation and valve leakage test will be performed at a steam test facility using test conditions similar to those for the installed valves in the plant, including valve orientation, ambient temperature, and steam conditions. Following SRV installation, the licensee's proposed testing includes verifying proper electrical connections and actuator performance. Although the tests of the SRVs at the steam test facility are not performed with the actual valve solenoids installed in the plant, the solenoid actuators are adequately tested and verified by separate surveillance tests.

A major difference between the current tests required by Mandatory Appendix I, Paragraph I-3410(d) and the proposed alternative is that the proposed alternative allows a series of overlapping tests to individually test SRV components. However, with the proposed alternative, all of the components necessary to manually actuate the SRVs will continue to be tested to demonstrate the functional capability of the valves, without the need to stroke test the valves on-line with system steam pressure. The NRC staff notes that the current testing requirements could result in seat leakage of the SRVs during power operation. Excessive seat leakage could result in excessive suppression pool temperature and level or unidentified drywell leakage. In addition, the staff notes that the proposed alternative is consistent with the 2004 Edition of the ASME OM Code (this Edition was incorporated into 10 CFR 50.55a in 2009), which no longer requires in-situ SRV testing. Therefore, the NRC staff finds that the proposed alternative to Paragraph I-3410(d) is acceptable.

Another difference between the current testing required by Paragraph I-1320(a) and the proposed alternative is that the alternative results in less frequent testing of the SRV components. Instead of testing each SRV every five years, the proposed alternative of Code Case OMN-17 allows extension of the test frequency from 60 months to 72 months plus a six-month grace period. The code case imposes a special maintenance requirement to disassemble and inspect each valve to verify that parts are free from defects resulting from time-related degradation or maintenance-induced wear prior to the start of the extended test frequency. The purpose of this maintenance requirement is to reduce the potential for set pressure drift. Code Case OMN-17 has not been added to Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," or included in 10 CFR 50.55a by reference. However, the NRC has allowed licensees to use OMN-17 provided all requirements in the code case are met. Consistent with the special maintenance requirement in Code Case OMN-17, each SRV will be refurbished to a like-new condition prior to the start of each 6 year test interval. Critical components will be inspected for wear and defects, and the critical dimensions will be measured during the inspection. Components will be reworked to within the specified tolerance or replaced if found to be worn or outside of specified tolerances. Furthermore, Code Case OMN-17 is performance based in that it requires SRVs be tested more frequently, if test failures occur. For example, the OMN-17 requires that two additional valves be tested when a valve in the initial test group exceeds the set pressure acceptance criteria. All remaining valves in the group are required to be tested if one of the additional valves tested exceeds its set pressure acceptance criteria. Therefore, the SRV test frequency would be equivalent to the current test frequency, if test failures occur. In addition, the licensee has had no setpoint failures of the valves to stroke open since 1999. Therefore, the NRC staff finds that the proposed testing frequency, with additional OMN-17 requirements, provides adequate periodic verification of valve operation.

The NRC staff determined that the proposed alternative testing conditions, locations and frequencies of the SRVs and associated components provide reasonable assurance that the valves will continue to operate when called upon to perform their safety-related function and is acceptable.

4.0 CONCLUSION

As set forth above, the NRC staff finds that the proposed alternative described in request VR-02 provides an acceptable level of quality and safety for main steam SRVs PSV4400, PSV4401, PSV4402, PSV4405, PSV4406, and PSV4407. 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)(i), and is in compliance with ASME OM Code requirements.

The NRC staff authorizes the alternative described in request VR-02 for the remainder of the DAEC fourth 10-year IST program interval which began on February 6, 2006, and is currently scheduled to end on February 5, 2016.

All other ASME OM Code requirements for which relief was not specifically requested and approved in the subject request remain applicable.

Principal Contributor: John Huang

Dated: July 11, 2012

July 11, 2012

Mr. Richard L. Anderson
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER – EVALUATION OF RELIEF REQUEST VR-02 REGARDING INSERVICE TESTING OF SAFETY/RELIEF VALVES (TAC NO. ME7336)

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The NRC staff has reviewed the subject request and finds, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and remains in compliance with ASME OM Code requirements.

This closes the NRC staff's action on the above submittal. If you have any questions, please contact Mr. Karl Feintuch at (301) 415-3079 or via e-mail at Karl.Feintuch@nrc.gov.

Sincerely,

/RA/

Istvan Frankl, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure: Safety Evaluation

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