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

September 6, 2011

Mr. Peter Wells
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER – ALTERNATIVE REGARDING PRESSURE TESTING REQUIREMENTS FOR MAIN STEAM SAFETY RELIEF VALVE PSV 4402 (TAC NO. ME5143)

Dear Mr. Wells:

By letter dated December 4, 2010, as supplemented by letter dated December 7, 2010, NextEra Energy Duane Arnold, LLC (the licensee) submitted a proposed relief request for Main Steam Safety Relief Valve (SRV) PSV 4402 for authorization to use, on a one-time-basis, a proposed alternative to the pressure test requirements specified in the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

As set forth above, the NRC staff concludes that the licensee has provided sufficient technical basis to find that compliance with the current requirements would cause an unnecessary burden on the licensee without a compensating increase in the level of quality and safety. Therefore, the licensee's proposed alternative pressure test requirements for the mechanical joints of PSV 4402 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) on a one-time-only basis at Duane Arnold Energy Center.

This closes the NRC staff's action on the above submittals. If you have any questions, please feel free to contact the project manager Mr. Karl Feintuch.

Sincerely,

R Pascarelli

Robert Pascarelli, 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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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION PROGRAM REQUEST FOR RELIEF FOR PSV 4402

DUANE ARNOLD ENERGY CENTER

NEXTERA ENERGY DUANE ARNOLD, LLC

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated December 4, 2010 (Accession No. ML103400074) as supplemented by letter dated December 7, 2010 (Accession No. ML103410547), NextEra Energy Duane Arnold, LLC (the licensee) submitted a proposed relief request for Main Steam Safety Relief Valve (SRV) PSV 4402 for authorization to use, on a one-time-basis, a proposed alternative to the pressure test requirements specified in the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The licensee stated that it intends to replace SRV PSV 4402, due to indications of leakage through the valve. Following replacement, the ASME Code requires that a system leakage test and VT-2 examination be performed to verify leak tightness of the mechanical joints (bolted connections). The test is required to be conducted at nominal operating pressure (approximately 1025 pounds per square inch gauge (psig)). In lieu of this ASME Code requirement, the licensee requested NRC authorization to perform the required VT-2 examination during a system leakage test performed at a minimum pressure of approximately 940 psig during the normal plant startup sequence.

The NRC staff has evaluated the licensee's request for relief pursuant to Title 10 of the *Code of Federal Regulations* 50.55a(a)(3)(ii) regarding compliance to the requirement that would result in hardship without a compensating increase in the level of quality and safety.

The NRC staff verbally authorized this relief on December 5, 2010, based on a preliminary response to an NRC request for additional information, which the licensee formally submitted on December 7, 2010.

2.0 REGULATORY REQUIREMENTS

Inservice inspection of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that (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.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements set forth in the ASME Code, Section XI, 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 interval, and subsequent 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 interval, subject to the limitations and modifications listed therein. The code of record for the Inservice Inspection Program and Repair/Replacement program for the Fourth 10-year interval at Duane Arnold Energy Center (DAEC) is ASME Section XI, 2001 Edition through 2003 Addenda. ASME Code Section XI, 1998 Edition is applicable for the pressure testing of mechanical joints included in Class 1, 2 and 3 repair and replacement activities, per 10 CFR 50.55a(b)(2)(xxvi).

3.0 TECHNICAL EVALUATION

As stated before, the component for which relief is requested is Main Steam Safety Relief Valve (SRV) PSV 4402.

3.1 ASME Code Requirements and the Licensee's Proposed Alternative

ASME Code, Section XI, 1998 Edition, IWA-4540(c), states, "Mechanical joints made in the installation of pressure retaining replacements shall be pressure tested in accordance with IWA-5211(a)."

ASME Code, Section XI, 1998 Edition, IWA-5211(a), states, "The pressure retaining components within each system boundary shall be subject to the following applicable system pressure tests under which conditions visual examination VT-2 is performed in accordance with IWA-5240 to detect leakages: (a) a system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature."

The licensee's basis for relief as delineated in the December 4, 2010, submitted is reproduced below:

> *The start-up sequence at the DAEC has been aborted to replace pilot assembly or complete pilot assembly and valve body of a Safety/Relief Valve (SRV) (PSV 4402). The SRV is connected to the main steam piping with a bolted, mechanical joint. Replacing the SRV for maintenance is considered a Repair-Replacement activity under the rules of ASME Code Section XI, 2001 Editions, including Addenda through 2003 (the current code of record for DAEC Repair/Replacement Program). Following the replacement, a system leakage test and VT-2 examination are required. The system leakage test is required to be performed at the nominal pressure associated with the reactor at 100% power (approximately 1025 psig).*

Several conditions associated with such testing represent an imposition on personnel safety, personnel radiation exposure, and challenges to the normal mode and manner of equipment operation. The SRV is not isolable from the reactor vessel; in order to perform this test, the primary system would need to be pressurized to the inboard system isolation valves. A leakage test and inspection at about 1025 psig cannot be performed during a normal plant startup, due to the excessive temperature and radiological exposure conditions to which licensee personnel would be exposed in the primary containment during the required VT-2 inspection. Extensive valve manipulations, alternative system lineups and procedural controls would be required for heating and pressurizing the primary system to establish the necessary test pressure, while complying with the Technical Specification (TS) requirements for Pressure-Temperature (P/T) Limits, without withdrawal of control rods, i.e. without using nuclear heat.

The licensee proposed to perform the system leakage test and VT-2 examination of the mechanical joints on SRV PSV 4401 during the normal operational start-up sequence at a minimum pressure of approximately 940 psig, in lieu of the nominal operating pressure associated with 100 percent reactor power (approximately 1025 psig). The VT-2 examination would be performed following the hold time required by the ASME Code. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in October 2012), another inspection of the bolted connection would be performed to look for any evidence of leakage at a minimum pressure of approximately 940 psig. This alternative was proposed on a "one-time-only" basis following the repair/replacement of the SRV planned for December 2010.

The licensee considered other available methods to reach nominal operating pressure required to perform the system leakage test and VT-2 examination. The licensee's discussion of these methods is reproduced below.

Pressurizing System Without Withdrawing Control Rods

NextEra Energy Duane Arnold cannot isolate PSV 4402 from the reactor vessel. Thus, NextEra Energy Duane Arnold would have to manipulate numerous valves, change system lineups, and establish procedural controls for heating and pressurizing the primary system in order to perform the system leakage test and VT-2 examination of the mechanical joints of the SRV without withdrawing control rods, while maintaining compliance with the TS P/T limits. The reactor pressure vessel (RPV) would need to be filled with coolant and the steam lines flooded to the inboard main steam isolation valves (MSIVs) to provide a water-solid condition. The pressure increase would be obtained by balancing the flow into the vessel, which is provided by the control rod drive (CRD) system, with the flow out of the vessel provided by the reactor water cleanup (RWCU) system via the dump flow control valve and flow controller. This is the method used during refueling outages to complete the RPV system leakage test.

This test typically takes about two days to accomplish, and the additional valve lineups and system reconfigurations necessary to support this test impose an additional challenge to the affected systems. After completion of the test, system lineups must be restored to support startup.

<u>*Pressurizing System During Normal Startup*</u>

Using normal startup procedures, the allowed pressure range for conducting the test would typically not be reached until a high power level (greater than 75% of rated). If access to the primary containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to current station ALARA practices. Establishing the 1025 psig test condition at a more moderate power level and in the manner needed to address radiation concerns would require a deviation from the method in which the primary system pressure control system (Electro-Hydraulic Control (EHC) Pressure Set) is normally used, as discussed below.

During a typical plant startup, after achieving criticality, the operating procedure directs the Operator to heat up and pressurize the reactor vessel (while maintaining the heat up rate within TS limits) by withdrawing additional control rods or raising EHC Pressure Set to maintain a turbine bypass valve within a specified "percent open" range. Adjustments to EHC Pressure Set are stopped, by procedure, when reactor pressure reaches 940 psig. The reactor power at that point is typically between 5 and 10% of rated.

While it is technically possible to manipulate these controls to establish the nominal system pressure of 1025 psig at lower power levels, doing so will affect core reactivity and could challenge plant safety systems, such as the reactor protection system (RPS). Changing the EHC settings outside of the normal range of operation for the purpose of performing this test at nominal operating pressure would pose an operational challenge, since this would be outside the normal operating parameters for startup. Procedural revisions would be required, as well as training provided to the Operators, to enable the EHC controls to be manipulated in a manner outside the norm.

The licensee concluded that the proposed alternative is the only reasonable approach, stating that:

Compliance with the Code-required system leakage test and inspection would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Application of this alternative test maintains reasonable levels of personnel safety and reduces the opportunity for the introduction of undesirable operational challenges. While NextEra Energy Duane Arnold does not expect that leakage will occur, any leakage at the bolted connection would be related to the differential pressure across the connection. The reduction in test pressure is less than 10%, and is not, therefore, expected to affect the ability of the VT-2 examination to detect leakage from the bolted connection. In the event that leakage would occur at the mechanical joint at the slightly higher pressure associated with 100% operating power, it would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. Leakage monitoring is required by the DAEC Technical Specifications. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in October 2012),

another inspection of the bolted connection will be performed to look for any evidence of leakage at a minimum pressure of approximately 940 psig.

The alternative will provide an acceptable verification of the integrity of the mechanical joint without unnecessary radiation exposure and operational challenges.

3.2 NRC Staff Evaluation

The NRC staff finds that testing main steam line SRV PSV 4402 at nominal operating pressure, in order to meet the above ASME Code requirements, would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee stated that performing the required testing of this safety relief valve could only be accomplished with unusual difficulty in that extensive valve manipulations, alternative system lineups and procedural controls would be required for heating and pressurizing the primary system to establish the necessary test pressure, while complying with technical specification requirements for pressure-temperature limits, without withdrawal of control rods, i.e., without using nuclear heat. In addition, testing this SRV at nominal operating pressure would require an at-power containment entry which represents an imposition on personnel safety due to excessive temperature and a tenfold increase in radiation exposure.

The NRC staff further concludes that performing the test and accompanying VT-2 examination at a minimum of 940 psig during the normal plant startup following the SRV repair/replacement activities provides reasonable assurance of structural integrity. This is because this test pressure is adequate to cause the bolted connection on the replacement SRV to leak during the test if a leak-tight connection has not been established. Should leakage occur later, DAEC Technical Specifications require monitoring of the reactor coolant system leakage using the drywell sump system and the primary containment air sampling system. In addition, the licensee committed in its letter of December 4, 2010, that if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in October 2012), another inspection of this mechanical joint will be performed to look for any evidence of leakage at a minimum pressure of 940 psig.

According to the licensee, the drywell monitoring systems would detect leakage if the mechanical joint were to leak at the higher pressures associated with nominal reactor power. These systems include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. The NRC staff agrees that monitoring such leakage provides additional assurance of the integrity of the component.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the licensee has provided sufficient technical basis to find that compliance with the current requirements would cause an unnecessary burden on the licensee without a compensating increase in the level of quality and safety. Therefore, the licensee's proposed alternative pressure test requirements for the mechanical joints of PSV 4402 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) on a one-time-only basis at Duane Arnold Energy Center.

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

Principal Contributor: Margaret Audrain

Dated: September 6, 2011

Mr. Peter Wells
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

September 6, 2011

SUBJECT: DUANE ARNOLD ENERGY CENTER – ALTERNATIVE REGARDING PRESSURE TESTING REQUIREMENTS FOR MAIN STEAM SAFETY RELIEF VALVE PSV 4402 (TAC NO. ME5143)

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Sincerely,
/RA/
Robert Pascarelli, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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