NEXTera ENERGY DUANE ARNOLD

March 12, 2012

NG-12-0107
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Duane Arnold Energy Center
Docket No. 50-331
Renewed Op. License No. DPR-49

Update to License Amendment Request (TSCR-129): Application for Technical Specification Change Regarding Alternative Testing of Safety/Relief Valves Sections Affected: 3.4.3, 3.5.1, and 3.6.1.5 TAC ME7337 and TAC ME7336

References: 1) P. Wells (NextEra Energy Duane Arnold) to USNRC, "License Amendment Request (TSCR-129): Application for Technical Specification Change Regarding Alternative Testing of Safety/Relief Valves," NG-11-0371, dated September 29, 2011 (ML112720444).
2) T. Beltz (USNRC) to P. Wells (NextEra Energy Duane Arnold), "Duane Arnold Energy Center -Issuance of Amendment re: Adoption of Technical Specifications Task Force Traveler (TSTF-425) to Relocate Specific Surveillance Frequencies to a Licensee-Controlled Program (TAC NO. ME5744), dated February 24, 2012 (ML120110282).

In Reference 1, NextEra Energy Duane Arnold, LLC (hereafter NextEra Energy Duane Arnold) requested, pursuant to 10 CFR 50.90, revision to the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). The proposed amendment would revise the DAEC TS by modifying existing Surveillance Requirements (SR) regarding the various modes of operation of the Main Steam System Safety/Relief Valves (SRVs).

Subsequent to that application, the Staff has issued Amendment No. 280 to the DAEC Operating License (Reference 2), which modified these same SRs. To ensure proper integration of the Reference 1 application into the current TS, as modified by Reference 2, Attachment 2 to this letter contains updated, marked-up TS pages that reflect the desired final outcome. Attachment 3 contains the corresponding updated

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clean, typed pages. No technical changes are being made to the Reference 1 application. Please note that the original marked-up pages of the TS Bases, transmitted for information only in Attachment 4 of Reference 1, are not being updated, as they properly reflect the final desired outcome.

In addition, the original description and assessment of changes contained in Attachment 1 of Reference 1 is also updated to reflect the changes due to Reference 2, as well as to correct two typographical errors in the description of changes. The first corrects an incorrect numbering for SR 3.6.1.5.1, which was incorrectly referred to in the original application as "SR 3.6.1.5.3." The DAEC TS does not contain an SR 3.6.1.5.3 and no new SR with that number is being requested. The second corrects the description to match the actual wording changes to the SRs as reflected in the marked-up TS pages. Attachment 1 to this letter contains these revisions, with the changes highlighted by ~~strikethrough~~ deletions and underlined inserts, with accompanying revision bars in the margin.

A copy of this update, along with the corrected 10 CFR 50.92 evaluation of "No Significant Hazards Consideration," is being forwarded to our appointed state official pursuant to 10 CFR 50.91.

NextEra Energy Duane Arnold continues to request NRC review and approval of the proposed license amendment within one year of the original Reference 1 application in order to support the next refueling outage for the DAEC, with a 30 day implementation grace period to implement this license amendment.

This letter makes no new commitments or changes to any existing commitments.

If you have any questions or require additional information, please contact Steve Catron at 319-851-7234.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on March 12, 2012

Peter Wells
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Attachments: 1. Description and Assessment
2. Proposed Technical Specification Changes (Mark-ups)
3. Proposed Technical Specification Changes (Clean, typed)

cc: NRC Region III Offices
NRC Resident Inspector Office (DAEC)
NRC Licensing Project Manager (DAEC)
M. Rasmusson (State of Iowa)

TSCR-129: Technical Specification Change Regarding Alternative Testing of Safety/Relief Valves
Sections Affected: 3.4.3, 3.5.1, and 3.6.1.5

DESCRIPTION AND ASSESSMENT

1.0 SUMMARY DESCRIPTION

The proposed amendment would modify Technical Specifications (TS) by revising specific Surveillance Requirements (SR) dealing with the testing of the various modes of operation of the Main Steam System Safety/Relief Valves (SRVs).

The proposed amendment would modify the TS requirements for testing of the SRVs by replacing the current requirement to manually actuate each SRV during plant startup with a series of overlapping tests that demonstrate the required functions of successive valve stages. Elimination of the manual actuation requirement at low reactor pressure and steam flow is desirable to decrease the potential for SRV leakage and spurious SRV openings.

NextEra Energy Duane Arnold requests NRC review and approval of the proposed license amendment within one year of this submittal in order to support the next refueling outage for the DAEC, which coincides with the next schedule performance of these SRs.

2.0 DETAILED DESCRIPTION

The proposed changes modify SR 3.4.3.2, SR 3.5.1.9, and SR ~~3.6.1.5.3~~ 3.6.1.5.1 to provide an alternative means for testing the dual function SRVs. The proposed changes will allow demonstration of valve capability by requiring that the valve actuator be manually stroked during each refueling outage without lifting the main valve seat.

Currently these TS SRs state, "Verify each [SRV/ADS valve/LLS valve] opens when manually actuated." The proposed amendment would change these SRs to "Verify each [SRV/ADS valve/LLS valve] actuator strokes when manually actuated.~~is capable of being opened.~~"

The current Frequency for these SRs is "~~24 months~~In accordance with the Surveillance Frequency Control Program;" this would be changed to state, "In accordance with the Inservice Testing Program."~~[1]~~

These SRs are each modified by a NOTE that states, "Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to

~~[1] NextEra Energy Duane Arnold has submitted a license amendment request TSCR-120 (ML110550570) to adopt generic TS change TSTF-425, Rev. 3, which would modify this SR Frequency to be "In accordance with the Surveillance Frequency Control Program." The proposed revision herein is intended to supersede that requested change, regardless of the order of approval of the two applications.~~

perform the test." This allowance would no longer be needed, and thus, is proposed to be deleted.

TS Bases associated with these Surveillance Requirements will be revised to describe the new testing method as discussed below. Revised Bases pages are attached for information only and do not require NRC approval. The final TS Bases pages will be submitted with a future update in accordance with TS 5.5.10, "Technical Specifications (TS) Bases Control Program."

3.0 TECHNICAL EVALUATION

3.1 System Description

SRVs installed at DAEC are Target Rock model 7467F three-stage safety/relief valves. Six SRVs are installed on the main steam lines between the reactor vessel and the inboard Main Steam Isolation Valves (MSIVs). Each SRV discharges via a separate tailpipe to a point below the minimum allowable water level in the suppression pool. SRVs open:

- In the safety mode on high reactor pressure, to provide primary system overpressure protection for the reactor coolant pressure boundary (RCPB), as discussed in Section 5.2.2 of the DAEC Update Final Safety Analysis Report (UFSAR).
- For four of the six SRVs, in the relief mode when actuated by the Automatic Depressurization System (ADS) logic of the Emergency Core Cooling Systems (ECCS). The ADS function is to rapidly reduce reactor pressure to within the capacity of low pressure ECCS pumps in the event of a small or intermediate break Loss of Coolant Accident with the High Pressure Coolant Injection System (HPCI) unable to maintain level due to equipment failure or break size (UFSAR Section 6.3.2.2.2).
- Two of the six SRVs (non-ADS valves), in the relief mode when actuated by the Low-Low Set (LLS) Logic. The function of the LLS valves is to mitigate high frequency hydraulic loads on the primary containment (torus) and thrust loads on the SRV tailpipes and discharge lines into the suppression pool during SRV operations. This reduces the possibility of a SRV tailpipe rupture occurring inside the torus above the suppression pool water level; thereby creating a bypass of the pressure suppression function. The LLS System automatically controls reactor pressure by opening and closing the LLS SRVs in the relief mode over a wider band of reactor pressure than the safety mode. The LLS valves are the two SRVs with the lowest safety mode pressure relief setpoints. This reduces the number and frequency of SRV actuations allowing the SRV discharge line

vacuum relief valves time to clear the discharge lines of water, thus lowering the thrust loads. (UFSAR Section 5.4.13)
- In the relief mode when manually actuated by individual control switches in the Main Control Room(all six SRVs), or by individual control switches on the Remote Shutdown Panel (selected SRVs only).

3.2 Operating Experience

Experience in the industry and at DAEC has shown that manual actuation of SRVs during plant operation (start-up) leads to valve seat leakage. In particular, manual actuation testing has been the principle cause of main stage seat leakage at DAEC. Such SRV leakage is discharged to the suppression pool via the discharge lines, where the increased energy and fluid volume additions to the suppression pool require more frequent operation of the Residual Heat Removal (RHR) System in the suppression pool cooling and pool pump-down modes in order to maintain those parameters within their TS limits (LCO 3.6.2.1 and 2, respectively). This causes the Low Pressure Coolant Injection (LPCI) mode of RHR, its primary safety function, to become inoperable. Main stage seat leakage also tends to mask the indications of pilot stage (or 2nd stage) seat leakage; pilot stage (or 2nd stage) leakage can cause misoperation of the SRV, including spurious actuation and/or failure to re-close after actuation. Excessive leakage of either stage requires plant shutdown to replace the leaking SRV.

The Boiling Water Reactor Owners' Group (BWROG) Evaluation of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item I1.K.3.16, "Reduction of Challenges and Failures of Relief Valves," recommends that the number of SRV openings be reduced as much as possible and that unnecessary challenges should be avoided. NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications" also recommends reducing the number of challenges to the SRVs. The proposed changes in testing are consistent with those recommendations.

NUREG-1482, Rev. 1, "Guidelines for Inservice Testing at Nuclear Power Plants," Paragraph 4.3.2.1 states, "In recent years, the NRC staff has received numerous requests for relief and/or TS changes related to the stroke testing requirements for BWR dual-function main steam safety/relief valves (SRVs). Both Appendix I to the ASME OM Code and the plant-specific TS require stroke testing of SRVs after they are reinstalled following maintenance activities. Several licensees have determined that in situ testing of the SRVs with reactor pressure can contribute to undesirable seat leakage of the valves during subsequent plant operation and have received approval to perform testing at a laboratory facility coupled with in situ tests without reactor pressure and other verifications of actuation systems as an alternative to the testing required by the ASME OM Code and TS."

3.3 Technical Evaluation

The manual actuation test currently prescribed in TS SRs 3.4.3.2, SR 3.5.1.9, and SR ~~3.6.1.5.3~~3.6.1.5.1 provides demonstration of the mechanical operation of the SRVs, and overlaps with other testing to demonstrate that the functions of the SRVs can be performed. The manual actuation test is performed once every 24 months, which corresponds to refueling outages.

The proposed revision to the SRs deletes the requirement to demonstrate the capability of the relief valves to open using steam pressure and substitutes a requirement to demonstrate that the valve actuator strokes when manually actuated. In addition, the proposed revisions to the TS Bases will describe the testing that will occur to verify the opening capability of the valve. The combination of testing the valve actuator and the verifications of the capability of the valve to open provides a complete verification of functional capability. This testing is described in more detail below.

S/RV Valve Actuator Testing

For the S/RVs, the actuator test will be performed by energizing a solenoid that pneumatically actuates a plunger. The plunger depresses the second stage disc located within the main valve body. Actuation of the plunger during plant operation allows pressure to be vented from the top of the main valve piston. This allows reactor pressure to lift the main valve piston, which opens the main valve disc. The test will verify movement of the plunger in accordance with vendor recommendations. However, since this test will be performed prior to establishing the reactor pressure needed to overcome main valve closure spring force, the main valve will not stroke during the test.

This test does not disturb the safety-mode first stage pilot valve. This is desirable, since leakage through the first stage pilot valve can mask main valve seat leakage after steam is applied to the valve.

Valve Testing

Valve testing will be performed at a steam test facility, where the valve (i.e., main valve and pilot valve) and an actuator representative of the actuator used at the plant will be installed on a steam header in the same orientation as the plant installation. The test conditions in the test facility will be similar to those in the plant installation, including ambient temperature, valve insulation, and steam conditions. The valve will then be leak tested, functionally tested to ensure the valve is capable of opening and closing (including stroke time), and leak tested a final time. Valve seat tightness will be verified by a cold bar test, and if not free of fog, leakage will be measured and verified to be below design limits. In addition, for the safety mode of S/RVs, an as-found setpoint verification and as-found leak check are performed, followed by verification of set pressure, and delay time. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components. A receipt inspection will be performed in accordance with the requirements of the NextEra Energy Quality Assurance Program. The storage requirements in effect ensure the valves are protected from physical damage. Prior to installation, the valve will again be inspected for foreign material and damage. The valve will be installed, insulated, and pneumatically and electrically connected. Proper connections will be verified per procedure.

The combination of the steam testing of the valve at the test facility and the valve actuator testing at the site will provide a complete check of the capability of the valves to open and close. Therefore, the proposed changes will allow the testing of the SRVs such that full functionality is demonstrated through overlapping tests, without cycling the valves under steam pressure with the valves installed.

As discussed in the referenced Relief Request (Ref. 6.1), ASME OM Code requirements for testing of main steam pressure relief valves are satisfied by the above testing. The requirement of Section I-3410(d) for manual actuation testing following reinstallation is to be exempted, and relief has been requested from that requirement based on the overlapping tests described above and the maintenance controls involved in reinstallation. The Relief Request provides additional discussion of Code Case OMN-17.

Another potential reason for in-situ testing of the relief valves is to verify that the discharge line is not blocked. The probability of blocking a relief valve discharge line and preventing the valve function is considered to be extremely remote. As implemented at DAEC, the NextEra Energy Foreign Material Exclusion program provides the necessary requirements and guidance to prevent and control introduction of foreign materials into structures, systems, and components. This program minimizes the potential for debris blocking a relief valve discharge line.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

10 CFR 50.36 requires in part that the operating license of a nuclear production facility include technical specifications. Paragraph (c)(2)(ii) of that part requires that a Limiting Condition For Operation (LCO) of a nuclear reactor must be

established for each item meeting one or more of four criteria. The SRV functions identified in LCOs 3.4.3, 3.5.1, and 3.6.1.5 all meet Criterion 3, "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." Paragraph (c)(3) further requires the establishment of Surveillance Requirements, "relating to test, calibration, or inspection to assure ... that the limiting conditions for operation will be met." As discussed above, the proposed changes in the SRs for the SRVs are sufficient to demonstrate the safety and relief modes operation of the SRVs, and therefore, are sufficient to ensure that these LCO are met.

4.2 Precedent

Similar changes in TS SRV testing have been approved for Dresden and Quad Cities (ML042600571), and Peach Bottom power plants (Ref. 6.2), which use three-stage Target Rock SRVs similar to the DAEC.

4.3 No Significant Hazards Consideration Determination

NextEra Energy Duane Arnold has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) using the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes modify TS SRs 3.4.3.2, SR 3.5.1.9, and SR ~~3.6.1.5.3~~3.6.1.5.1 to provide an alternative means for testing the main steam SRVs, ADS valves, and LLS relief valves. Accidents are initiated by the malfunction of plant equipment, or the catastrophic failure of plant structures, systems, or components. The performance of SRV testing is not a precursor to any accident previously evaluated and does not change the manner in which the valves are operated. The proposed testing requirements will not contribute to the failure of the SRVs nor any plant structure, system, or component. NextEra Energy Duane Arnold has determined that the proposed change in testing methodology provides an equivalent level of assurance that the SRVs are capable of performing their intended safety functions. Thus, the proposed changes do not affect the probability of an accident previously evaluated.

The performance of SRV testing provides confidence that the relief valves are capable of depressurizing the reactor pressure vessel (RPV). This will protect the reactor vessel from overpressurization and allow the combination of the Low Pressure Coolant Injection and Core Spray Systems to inject into the RPV as designed. The LLS relief logic causes two LLS relief valves to be opened at a lower pressure than the relief mode pressure setpoints and causes the LLS relief valves to stay open longer, such that reopening of more than one valve is prevented on subsequent actuations. Thus, the LLS relief function prevents excessive short duration SRV cycling, which limits induced thrust loads on the SRV discharge line for subsequent actuations of the relief valve. The proposed changes do not affect any function related to the safety mode of the dual function SRVs. The proposed changes involve the manner in which the subject valves are tested, and have no effect on the types or amounts of radiation released or the predicted offsite doses in the event of an accident. The proposed testing requirements are sufficient to provide confidence that these valves are capable of performing their intended safety functions.

In addition, an inadvertent opening of an SRV is an analyzed event in the DAEC UFSAR (Section 15.1.7.2), as well as the assumption of a single SRV failure to open on demand in other transients and accidents, as appropriate (e.g., one ADS valve failure in the LOCA analysis). Since the proposed testing requirements do not alter the assumptions for any analyzed transient or accident, the radiological consequences of any accident previously evaluated are not increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not affect the assumed accident performance of the main steam SRVs, nor any plant structure, system, or component previously evaluated. The proposed changes do not install any new equipment, and installed equipment is not being operated in a new or different manner. The proposed change in test methodology will ensure that the valves remain capable of performing their safety functions due to meeting the testing requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, with the exception of opening the valve following installation or maintenance for which a relief request has been submitted (Ref. 6.1), proposing an acceptable alternative. No setpoints are being changed which would alter the dynamic response of plant equipment. Accordingly, no new failure modes are introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

Overpressure protection of the RCPB is based on the SRVs' setpoints and total relief capacity. The setpoints are verified at an offsite testing facility; this requirement is not altered by the proposed change. The relief capacity of each SRV is determined by the valve's geometry, which is also not altered by the proposed test methods.

The proposed changes will allow testing of the valve actuation electrical circuitry, including the solenoid, and mechanical actuation components, without causing the SRV to open. The SRVs will be manually actuated prior to installation in the plant. Therefore, all modes of SRV operation will be tested prior to entering the mode of operation requiring the valves to perform their safety functions. The proposed changes do not affect the valve setpoint or the operational criteria that cause the SRVs to open during plant transients or accidents, either manually or automatically. There are no changes proposed which alter the setpoints at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NextEra Energy concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 <u>Conclusions</u>

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. NextEra Energy Duane Arnold has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

<u>Basis</u>

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in the 10 CFR 50.92 evaluation included in this exhibit, the proposed amendment does not involve a significant hazards consideration.

2. The proposed changes do not result in an increase in power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

3. The proposed changes do not result in changes in the level of control or methodology used for processing of radioactive effluents or handling of solid radioactive waste nor will the proposal result in any change in the normal radiation levels within the plant. There is no significant increase in individual or cumulative occupational radiation exposure.

6.0 REFERENCES

6.1 P. Wells (NextEra Energy Duane Arnold) to NRC, NG-11-0365, "Relief Request VR-02 Regarding In-Service Testing of Safety/Relief Valves," dated September 30, 2011.

6.2 Letter from M. C. Thadani (U. S. NRC) to G. D. Edwards (PECO Energy Company), "Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, Technical Specifications Revision Relating to the Surveillance of the Safety Relief Valves (TAC Nos. MA1741 and MA1742)," dated October 5, 1998

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Technical Specification Pages

(Updated Markups)

3 pages to follow

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the SRVs and SVs are as follows: Number of SRVs / Setpoint (psig) 1 — 1110 ± 33.0 1 — 1120 ± 33.0 2 — 1130 ± 33.0 2 — 1140 ± 33.0 Number of SVs / Setpoint (psig) 2 — 1240 ± 36.0 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	~~-----NOTE-----~~ ~~Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ ~~----------~~ Verify each SRV ~~opens~~ actuator strokes when manually actuated.	In accordance with the Inservice Testing Program ~~In accordance with the Surveillance Frequency Control Program~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	------------------NOTE------------------ Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.9	------------------NOTE------------------ ~~Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ -- Verify each ADS valve ~~opens~~ actuator strokes when manually actuated.	In accordance with the Inservice Testing Program ~~In accordance with the Surveillance Frequency Control Program~~

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.5.1	-------------------------NOTE------------------------- Not required ~~to be~~ performed ~~until~~ 12 hours after reactor steam ~~pressure~~ and flow are adequate ~~to~~ perform the test. -- Verify each LLS valve ~~opens~~ actuator strokes when manually actuated.	In accordance with the Inservice Testing Program ~~In accordance with the Surveillance Frequency Control Program~~
SR 3.6.1.5.2	-------------------------NOTE------------------------- Valve actuation may be excluded. -- Verify the LLS System actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program

Attachment 3 to
NG-12-0107

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Technical Specification Pages

(Updated Clean, typed)

3 pages to follow

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the SRVs and SVs are as follows: Number of SRVs / Setpoint (psig) 1 / 1110 ± 33.0 1 / 1120 ± 33.0 2 / 1130 ± 33.0 2 / 1140 ± 33.0 Number of SVs / Setpoint (psig) 2 / 1240 ± 36.0 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	Verify each SRV actuator strokes when manually actuated.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	-------------------------NOTE------------------------------- Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.9	Verify each ADS valve actuator strokes when manually actuated.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.5.1	Verify each LLS valve actuator strokes when manually actuated.	In accordance with the Inservice Testing Program
SR 3.6.1.5.2	-------------------------NOTE------------------------- Valve actuation may be excluded. -- Verify the LLS System actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program