UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-1257

June 10, 2016

Mr. David R. Vineyard
Vice President
Southern Nuclear Operating Company, Inc.
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT - NRC REACTIVE INSPECTION REPORT 05000321/2016009

Dear Mr. Vineyard:

On April 1, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed its initial assessment of the circumstances associated with the Hatch Unit 1 safety relief valves (SRVs), which occurred during valve testing on March 30, 2016, at NWS Technologies. During that testing, three SRVs failed to fully re-close following actuations on the test stand. The cause of the failure of the SRVs to re-close during testing had not been determined. Based on this initial assessment, the NRC sent an inspection team to NWS Technologies and your site on April 4, 2016.

On April 27, 2016, the NRC completed its special inspection and the NRC inspection team discussed the results of this inspection with you and other members of your staff. The inspection team documented the results of this inspection in the enclosed inspection report.

The NRC inspectors did not identify any findings or violations of more than minor significance.

In accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's “Rules of Practice,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS).

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Sincerely,

/RA/

Shane Sandal, Chief
Reactor Projects Branch 2
Division of Reactor Projects

Docket Nos.: 50-321
License Nos.: DPR-57

Enclosures: IR 05000321/2016009
w/Attachment: Supplemental Information

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Letter to David R. Vineyard from Shane Sandal Dated June 10, 2016.

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT - NRC REACTIVE INSPECTION REPORT 05000321/2016009

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REGION II

Docket Nos.:	50-321
License Nos.:	DPR-57
Report No.:	05000321/2016009
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Edwin I. Hatch Nuclear Plant
Location:	Baxley, Georgia
Dates:	April 4 through April 27, 2016
Inspectors:	D. Hardage, Senior Resident Inspector D. Retterer, Resident Inspector J. Billerbeck, Mechanical Engineer E. Torres, Reactor Operations Engineer R. Bernhard, Senior Reactor Analyst
Approved by:	Shane Sandal, Chief Reactor Projects Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000321/2016009, April 4, 2016, through April 27, 2016; Edwin I. Hatch, Unit 1, Special Inspection to evaluate Hatch nuclear plant Unit 1 safety relief valve testing failures; Inspection Procedure 93812, “Special Inspection.”

A five-person U.S. Nuclear Regulatory Commission (NRC) team, comprised of two resident inspectors, a mechanical engineer, a reactor operations engineer and a regional senior reactor analyst, conducted this Special Inspection. The NRC’s program for overseeing the safe operations of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 5.

REPORT DETAILS

Summary of the Degraded Condition

In February 2016, during the Hatch Unit 1 refueling outage 1R27, all 11 safety relief valves (SRVs) were removed and replaced. The SRVs were Target Rock Model 0867F 3-stage valves that had been installed during the previous refueling outage (1R26). Following the 1R27 outage the SRVs were sent to NWS Technologies (NWS) for testing to verify the safety function lift setpoints as required by Hatch Unit 1 Technical Specification (TS) Surveillance Requirement (SRS) 3.4.3.1. Three SRVs failed to fully re-close following test actuations on the test stand at NWS. These valves had been installed in the ‘A’, ‘D’, and ‘H’ positions during the last Unit 1 operating cycle.

Special Inspection Charter

Based on the deterministic and conditional risk criteria specified in Management Directive 8.3, “NRC Incident Investigation Program,” a special inspection was initiated in accordance with NRC Inspection Procedure 93812, “Special Inspection Team.” The inspection focus areas included the following special inspection charter items:

1. Develop a detailed sequence of events from the time the licensee received the A, D, and H SRVs from the vendor until their subsequent test failure at NWS. Include any work, maintenance or corrective action program entries for the SRVs during that time.

2. Review and evaluate the licensee’s operability determination related to the SRVs currently in-service on Unit 1 and Unit 2. Include a review of justification for continued operation, immediate corrective actions, compensatory measures, and risk reduction actions the licensee has taken in response to the test failures.

3. Review and evaluate the licensee’s processes to remove, package and ship these SRVs to the testing laboratory. Determine if the SRVs that stuck open during the test at NWS had previously undergone testing at the NTS/Wyle (NTS) Laboratory facility.

4. Evaluate potential causes for the SRV test failures including variations in test methodology, non-conforming conditions, or evidence of valve degradation.

5. Review and assess the licensee’s maintenance practices related to the maintenance/inspection of the SRVs.

6. Review and assess the testing laboratory’s practices related to the receipt, storage and handling of the SRVs prior to testing.

7. Review and evaluate the licensee’s practices against vendor recommendations regarding maintenance and testing of 3-stage SRVs.

8. Review and verify the licensee's reportability determination was in accordance with the reportability criteria in 10 CFR 50.72 and NUREG-1022.

9. Assess the licensee's actions resulting from NRC generic communications, vendor technical bulletins, and industry operating experience related to 3-stage Target Rock SRVs.

10. Collect data necessary to support completion of the significance determination process, if applicable.

11. Identify any potential generic safety issues and make recommendations for appropriate follow-up action (e.g., Information Notices, Generic Letters, and Bulletins).

4. OTHER ACTIVITIES

4OA5 Other Activities – Special Inspection (IP 93812)

.1 Develop a detailed sequence of events from the time the licensee received the A, D, and H SRVs from the vendor until their subsequent test failure at NWS. Include any work, maintenance or corrective action program entries for the SRVs during that time.

The Target Rock 0867F 3-stage SRVs removed from Unit 1 in February 2016 were the first full set of Target Rock 0867F SRVs that had experienced a 24 month in-service operating cycle at Hatch Nuclear Plant. NTS Laboratories had refurbished all of the 0867F valves installed in Unit 1 during 1R26 in late 2013 on purchase order (PO) SNG10060848 (except for valve 274). Valve 274, which was installed in the plant in the 'H' position during 1R26, was previously refurbished at NTS Laboratories on PO SNG10046102 in late 2012. Final certification testing was also performed by NTS Laboratories before the valves were shipped to Hatch Nuclear Plant. The following summary highlights events from the time the licensee received the 'A', 'D', and 'H' SRVs from NTS Laboratories until their subsequent test actuations at NWS:

Date	Event
Valve 270 ('A')	
01/08/2014	Valve 270 received at Hatch Nuclear Plant.
02/13-21/2014	Valve 270 installed during refueling outage 1R26 in position 'A' (1B21-F013A).
03/03-10/2014	Hatch Unit 1 startup and power ascension to Rated Thermal Power (RTP).

03/16/2014	With the plant stable, pilot stage temperature decreased from 494 to 445 ^{o}F, second stage temperature deceased from 540 to 539 ^{o}F, and downstream tailpipe temperature increased from 184 to 191 ^{o}F. Within a shift, temperatures turned and returned to pre-event values. Condition Report (CR) 787589
08/15/2014	Pilot temperature fluctuating/erratic. CR 854347
09/25/2014	Pilot temperature indication failed high. CR 871308
01/06/2016	Pilot stage temperature intermittently erratic, appears to have noise in the signal. CR 10164951
02/08 - 03/03/2016	Refueling outage 1R27. All 11 SRVs removed and replaced.
03/15/2016	SRVs removed during refueling outage 1R27 shipped to NWS. Work Order (WO) SNC700951
03/31/2016	As found testing performed at NWS. Valve passed TS SR 3.4.3.1 over pressure safety function lift test but on the second test, the valve failed to close. CR 10204447, CR 10207459

<u>Valve 92 (‘D’)</u>

01/08/2014	Valve 92 received at Hatch Nuclear Plant.
02/13-21/2014	Valve 92 installed during refueling outage 1R26 in position ‘D’ (1B21-F013D).
03/03-10/2014	Unit 1 startup and power ascension to RTP.
03/17/2014	Pilot stage temperature decreased from 435 to 420.9 ^{o}F, second stage temperature decreased from 537.7 to 537.2 ^{o}F, and downstream tailpipe temperature increased from 167.4 to 171.9 ^{o}F over an hour period. Temperatures stabilized/return to normal. CR 788177, Corrective Action Report (CAR) 209673
08/03/2014	Annunciator "SAFETY/BLOWDOWN VLV PILOT/SEAT LEAKING" received due to 1B21F013D pilot temperature decreasing from 420.3 to 420 ^{o}F. Licensee determined decrease in the pilot temperature was due to a change in the ambient temperature in the drywell. CR 848547, CR853250, CAR 211584
01/19/2015	During reactor shutdown at approximately 200 pounds per square inch gage (psig), SRV ‘D’ downstream tailpipe temperature increased from 150 to 185 ^{o}F in about an hour. Temperature stayed at 185 ^{o}F for about an hour, then decreased to follow temp trend of other SRVs. CR 10014204, CAR 249601

01/29/2015	SRV pilot temperature slowly decreased after plant startup. The pilot temperatures reduced approximately 7 °F while the second stage has reduced 0.5 °F. The downstream tailpipe temperature did not increase during this time period. CR 10018772
03/09/2015	Pilot stage temperature increased six degrees on both thermocouples and a 1.3 °F increase on the second stage temperature. No change in downstream tailpipe temperature. CR 10038362
02/08 - 03/03/2016	Refueling outage 1R27. All 11 SRVs removed/replaced.
03/15/2016	SRVs removed during refueling outage 1R27 shipped to NWS. WO SNC700951
03/30/2016	As found testing performed at NWS. Valve failed TS SR 3.4.3.1 over pressure safety function lift test (valve lifted low at 1113 psig) and on the second test, the valve failed to close. CR 10204045, CR 10207459

<u>Valve 274 (‘H’)</u>

01/05/2014	Valve 274 received at Hatch Nuclear Plant.
02/13-21/2014	Valve 274 installed during refueling outage 1R26 in the ‘H’ position (1B21-F013H).
03/03-10/2014	Unit 1 startup and power ascension to RTP
02/08 - 03/03/2016	Refueling outage 1R27. All 11 SRVs removed and replaced.
03/15/2016	SRVs removed during refueling outage 1R27 shipped to NWS. WO SNC700951
03/30/2016	As found testing performed at NWS. Valve passed TS SR 3.4.3.1 over pressure safety function lift test but on the second test, the valve failed to fully re-close. CR 10204447, CR 10207459

Except for installation and removal, no maintenance was performed on valves 92, 270, or 274 at Hatch Nuclear Plant. There were no operational or test strokes of the SRVs during the in-service time of these valves. No condition reports were generated on the ‘H’ valve during operation. Condition reports generated on the ‘A’ and ‘D’ valves during operation were all related to pilot leakage except for CR 10014204 which documented indications of main seat leakage on the ‘D’ SRV at low pressure during a mid-cycle shutdown. The team concluded the issues documented in these conditions reports were not directly associated with the degradation observed in the main valve bodies during post-operational testing at NWS.

.2 Review and evaluate the licensee's operability determination related to the SRVs currently in-service on Unit 1 and Unit 2. Include a review of justification for continued operation, immediate corrective actions, compensatory measures, and risk reduction actions the licensee has taken in response to the test failures.

a. Inspection Scope

The team reviewed and evaluated the licensee's operability determination related to the SRVs currently in-service on Unit 1 and Unit 2 including a review of justification for continued operation, immediate corrective actions, compensatory measures, and risk reduction actions the licensee performed in response to the test failures.

b. Findings and Observations

Hatch had 11 SRVs, manufactured by Target Rock, located on the steam lines inside the primary containment. They were three-stage dual function valves that operated in a safety mode or a relief mode. In the safety mode, the spring loaded pilot valve opens when steam pressure at the valve inlet expands the bellows to the point that the bellows force overcomes the force holding the pilot valve closed. Opening the pilot valve allows steam to pass to the second stage operating piston which causes the second stage disc to open. This vents the chamber over the main valve disc to the downstream side of the valve, which causes a pressure differential to develop across the main valve piston and opens the main valve. Seven of the SRVs comprise the automatic depressurization system (ADS), which is designed to depressurize the reactor, in the event the high pressure coolant injection (HPCI) system cannot maintain reactor water level during certain postulated accidents so that the low pressure emergency core cooling system (ECCS) can inject water. Four of the 11 SRVs are equipped to provide the low level set (LLS) function. The LLS logic causes the LLS valves to be opened at a lower pressure than safety mode pressure setpoints and stay open longer, so that reopening more than one SRV is prevented on subsequent actuations. Therefore, the LLS function prevents excessive short duration SRV cycles with valve actuation at the relief setpoint. The SRVs are required to be operable at reactor coolant system pressures greater than 150 psig.

While the SRVs satisfactorily opened during the setpoint test actuations at the testing facility, disassembly revealed varying levels of damage to the main valve stage internals. In its normal condition, the main valve stem is threaded into the main piston, torqued, and secured with a stem nut and locking tab. During disassembly, technicians noted on multiple valves that the stem nut was not tightly secured and the locking tab was slightly rotated out of its locked position. Damage to some internal valve main stage parts was noted. Specifically, valves 270 and 274 had significant degradation of the threads at the piston-to-stem interface. The thread degradation for these valves had progressed to the point that the piston rotated freely by hand on the stem. The piston was captured on the stem by intact threads and stem nut above the piston. Five other SRVs exhibited some minor thread damage. Four SRVs had no thread damage. Six of the valves also had grooves worn in the main operating cylinder liner where the piston rings rest while the valve is in its fully closed position.

Similar damage was observed in SRVs removed from Pilgrim Nuclear Power Station in March 2015. Two 0867F SRVs had failed to open on demand at Pilgrim in 2013 and 2015 (NRC Special Inspection Report, Pilgrim Nuclear Power Station - ADAMS ML15147A412). Following these events, the valve's manufacturer, Curtiss-Wright Flow Control Corporation (Curtiss-Wright), Target Rock Division, issued a 10 CFR Part 21 Report (ADAMS ML15077A422) due to the potential to induce a defect during the testing of the relief valve Model 0867F. Target Rock determined excessive valve impact loads during limited flow testing on the test stand can relieve the torque applied to the piston-to-stem interface (de-torqueing) leading to the creation of clearance between the piston and the main valve disc (de-shouldering). This loss of shoulder-to-shoulder contact allows relative motion between the main piston and main disc. If the excessive impact load damages the lock nut or lock tab and shortens the length of the main valve spring, in-service plant vibratory loads can allow the piston to rotate circumferentially and rock relative to the stem shoulder further increasing the clearance between the piston and the stem. This mechanism is time dependent. Increasing the amount of time the piston is exposed to these conditions will increase the propensity for fretting wear. The team concluded the degradation mechanism described in the Target Rock 10 CFR Part 21 report (i.e., de-torqueing and de-shouldering during preservice certification testing followed by vibration induced fretting of the stem piston interface and the piston rings/guide interface) was consistent with the damage observed in the Hatch Unit 1 main valve internals.

The 10 CFR Part 21 Report recommended valves currently installed be inspected to ensure proper piston-to-stem shoulder engagement based on plant-specific indications of the potential for fretting. At the time this recommendation was made, all the Hatch SRVs on both units had been converted to the 0867F 3-stage model. Unit 1 valves which were removed during 1R27 had been installed during the spring of 2014, and Unit 2 had just completed a refueling outage with had installed the 0867F SRVs. Hatch had four 0867F valves available for inspection that had operational time in service (and were not currently installed in the plant). Inspections were performed on these valves and none were de-shouldered or had indications of fretting wear. Based on this plant specific data with regard to the as-found condition of the piston/main disc shoulder and lack of fretting, the licensee concluded there was low potential for de-shouldering and subsequent in-plant fretting and did not schedule inspections of in-service valves prior to planned refueling outages.

Target Rock also recommended in the 10CFR Part 21 Report (for valves not yet installed) that additional inspections be performed on the main valve internals after the certification test. These inspections required the base assembly to be removed from the main body after as-left testing to inspect the threaded stem-to-main piston connection to ensure there was no de-shouldering caused by the testing. These inspections were performed on the valves currently installed in Hatch Unit 1.

The licensee's operability determination concluded that the augmented inspections performed on the currently installed Unit 1 SRVs provided reasonable assurance that the SRVs were operable. The team noted that the post certification testing inspections on the valve internals of the SRVs currently installed in Unit 1 should minimize the conditions necessary for vibration-induced fretting and prevent the degradation mechanism that was observed in the valves removed from Unit 1 during 1R27.

The team also assessed the licensee's evaluation that addressed potential Unit 2 failure mechanisms based on the observed degradation of the Unit 1 SRVs removed during 1R27.

Failure to close:

The team noted the cause of the failure of the three SRVs to fully reseat on the test stand was due to limited steam flow in the testing configuration, shortening of the main disc spring, and increased friction due to wear/fretting in the piston/guide area. Limited flow means a gag device is used in the discharge throat of the SRV during testing. This configuration minimizes the capacity requirements of the steam system at the test facility but the flow through the valve and the differential pressure across the main disc is different than would be experienced in the plant. Deformation and shortening of the main disc spring had previously been noted in Target Rock SRVs and does not alone prevent the valve from closing on the test stand. Fretted grooves in the main operating cylinder liner where the piston rings rest while the valve is in its closed position and the wear at the piston-to-stem interface (which allowed piston wobble) created additional resistance to valve movement. The weakened spring combined with internal valve damage resulted in the failures of the valve to fully close on the limited steam flow test stand. The licensee's operability determination noted that during normal plant operation, higher steam flow would provide the additional force necessary to close the valve. Target Rock calculated the in-plant closing force on the valve to be approximately 12,000 pounds above the closing force on the test stand. This force, combined with the observation that the valves which failed to fully close could be moved by hand, demonstrated that there is reasonable assurance that the Unit 2 SRVs would close when demanded.

Potential failure of SRVs to open:

The team noted two Hatch SRVs had significant degradation of the threads attaching the piston to the stem (i.e., the piston rotated freely on the stem but was held in place by the jam nut which was in place on intact but degraded threads). In this condition, failure of the valve (due to the piston cocking and jamming in the guide or by the jam nut working off the stem and piston coming free of the stem) was possible. The licensee stated in the operability determination that excessive fretting can cause binding, but this condition had not been known to prevent Hatch SRV's from opening. The licensee noted that cycles on the test stand at rated pressure were performed with no indication of binding preventing or inhibiting the valves from cycling open at high system pressures. Additionally, the licensee contracted an independent engineering firm to evaluate the potential for valve binding. The engineering analysis concluded the potential for binding in the open direction was low because (with the disc in the seat) the seat and guide provide relatively tight control on the stem. This minimized the achievable radial offset of the piston. Even if the piston cross threaded due to piston-to-stem fretting wear, the report concluded there was adequate clearance to prevent valve binding. The licensee concluded there was reasonable assurance that the Unit 2 SRVs would perform their over-pressure protection safety function and meet Technical Specification surveillance requirements.

The engineering analysis also evaluated the ability of the as-found Hatch SRVs to open at 150 psig reactor coolant pressure (minimum pressure required for operability of ADS valves). In 2013 and 2015 at Pilgrim Nuclear Power Station, two different SRVs failed to open at low pressure (one valve for each event). The analysis noted that the Pilgrim valves had deeper fretting grooves and steeper ramp angles than was observed in the Hatch Unit 1 SRV's. This significantly contributed to the amount of additional force that would have been necessary to open the Pilgrim valves. The engineering report determined the fretting wear on the Hatch Unit 1 valves would not have prevented the SRVs from opening at low reactor coolant pressure. The available piston differential pressure force would have been more than sufficient to initiate the open stroke and drive the piston rings through the fretting wear bands. The licensee concluded there was reasonable assurance that the Unit 2 SRVs would open at low reactor coolant pressure. The licensee determined that the Unit 2 SRVs were operable but in a degraded/nonconforming condition due to the potential for continued in-service vibration wear.

The team concluded that the Hatch SRVs were susceptible to fretting as described in the Target Rock 10 CFR Part 21 Report. The team also noted the vibration-induced fretting was a time dependent degradation mechanism. The licensee conducted a mid-cycle maintenance shutdown of Unit 2 on May 20, 2016, to replace all 11 SRVs and inspect the main valve internals in accordance with vendor recommendations discussed in the Part 21 report. The post certification testing inspection of the replacement SRVs installed in Unit 2 has been completed and the licensee determined that the newly installed replacement Unit 2 SRVs were operable.

.3 Review and evaluate the licensee's processes to remove, package, and ship these SRVs to the testing laboratory. Determine if the SRVs that stuck open during the test at NWS had previously undergone testing at the NTS Laboratory facility.

a. Inspection Scope

The inspectors reviewed Southern Nuclear Operating Company (SNC) procedure 52GM-B21-005-0, "Main Steam Safety Relief Valve Maintenance," Revision 24, which prescribed SNC processes to remove, package and ship SRVs to NTS and NWS. The inspectors reviewed SNC Quality Assurance Topical Report (QATR), Revision 13, to understand quality assurance program requirements and basis. In addition, the inspectors reviewed PO SNG10060848, Revision 2, dated January 30, 2014, from SNC to Curtiss-Wright for refurbishment as needed, disassembly and decontamination, component inspection/repair and re-assembly as necessary for Hatch 3-stage SRVs installed in Unit 1 during 1R26. The PO included the performance of steam certification testing at NTS for main valve serial numbers: 92, 123, 216, 270, 272, 273, 276, 309, 312, and 1245. The inspectors reviewed TR-FRSDB-12322-001-00, "Target Rock Field Service Data Book," dated January 25, 2013. The data book included the rework and steam certification testing performed on main valve serial number 274 performed at NTS.

b. Findings and Observations

The inspectors determined that procedure 52GM-B21-005-0 provided adequate instructions for the removal, packaging and shipping of Hatch's SRVs to prevent damage or deterioration. The procedure steps through the removal of the valve major components including quality control hold points to perform in-service inspection (ISI) in accordance with the American Society of Mechanical Engineers (ASME) Section XI code. The inspectors verified that SNC QATR handling, storage and shipping requirements were imposed to Curtiss-Wright on PO SNG10060848 for the return of operable SRVs that were installed in Hatch Unit 1 during 1R26.

The inspectors determined during the review of PO SNG10060848 and TR-FRSDB-12322-001-00, that NTS successfully completed steam certification testing on Hatch's 3-stage SRVs that failed to re-close, main valve serial numbers: 92, 270 and 274. SRVs 92 and 270 were tested on December 18, 2013. The valves were returned to SNC on January 8, 2014. SRV 274 was tested on December 28, 2012. The valve was returned to SNC January 5, 2013.

.4 Evaluate potential causes for the SRV test failures including variations in test methodology, non-conforming conditions, or evidence of valve degradation.

a. Inspection Scope

The team inspected SRV valve internals at NWS. Two of the valves were shipped to NTS for destructive disassembly due to degradation of the stem and piston threads. The team reviewed and evaluated the testing and inspection results from these valves. The team interviewed licensee, test facility and Target Rock engineers. In addition the team reviewed an engineering analysis the licensee had performed by an external engineering contractor.

b. Findings and Observations

The team reviewed a vendor evaluation that showed there is very little closing force on the test stand (i.e., the limited flow test condition due to the gag used in the valve discharge) because steam pressure above and below the main disc and above and below the piston are essentially equal. In that case, it is only the spring above the piston which contributes closing force to the disc. That same calculation also showed that under actual in-service main steam conditions (i.e., the full flow condition wherein no test gag exists), there is a steam pressure gradient through the valve which produces higher steam pressures both above the piston and over the main disc delivering approximately 12,000 pounds of additional closing force that would not be present on the test stand.

The team concluded that a degraded valve (shortened spring, de-shouldered stem-to-piston interface with attendant fretting and grooving of the main guide) may not fully re-close because the weakened spring does not contribute enough force to overcome the additional friction due to the piston rings sliding across the fretted grooves in the guide. This conclusion was supported when the team reviewed several test plots for the affected valves which showed the position of the main disc/stem/piston assembly as a function of time. The team noted that the plots indicated that the assembly would proceed towards the closed position until the point where the lower piston ring encountered the first groove in the guide (worn by the upper piston ring) and then stall. The team determined that a valve in good operating condition should normally close on the test stand with only spring force to assist it.

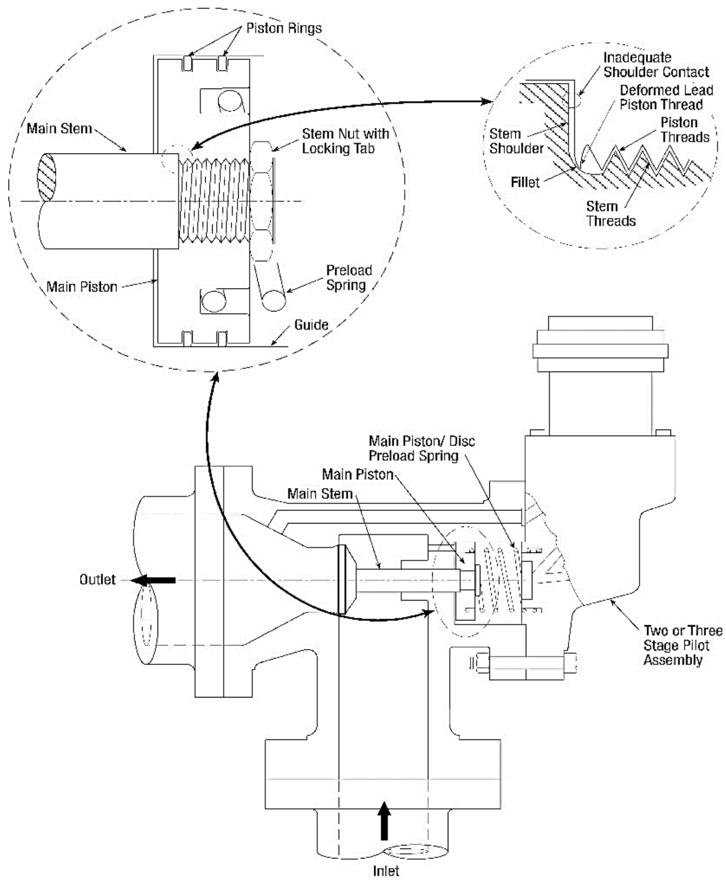

Figure 1 - BWR SRV (NRC Information Notice 2003-01, “Failure of Boiling Water Reactor Target Rock Main Steam Safety/Relief Valve,” ADAMS ML030140543)

The team noted that the test methodology could cause de-shouldering on the valve’s open stroke which is the predecessor to the subsequent fretting damage (at the stem-to-piston interface and the piston-to-guide interface) in a plant vibration environment. Target Rock determined the root cause of the Pilgrim main guide fretting (due to piston/ring wear) resulted from excessive impact load during limited flow testing which relieved the torque applied to the piston-to-stem interface leading to the creation of a significant clearance between the piston and the main disc (de-shouldering) as well as plastic deformation of the piston-to-stem threads. This loss of shoulder-to-shoulder contact allows relative motion between the main piston and main disc. If the excessive impact load also damages the jam nut or tab washer and shortens the main spring, plant vibratory loads can allow the piston to rotate circumferentially and rock relative to the stem shoulder further increasing the clearance between it and the stem. This action is time dependent in that increasing the amount of time the piston is exposed to these conditions will increase the propensity for fretting wear.

Target Rock is pursuing an enhanced 0867F design which will continue to meet in-service specification requirements for the valve yet be better able to mitigate the extreme loads encountered in limited flow testing. In the meantime, Target Rock has recommended that the number of test stand cycles be minimized for a given valve and that additional inspections be performed following test stand cycling and prior to placing the valve in-service.

The team noted that all vendor test facilities employed the use of a flow gag (i.e., a plate fitted in the discharge side of the main seat to block off most of the steam flow). This allows inlet pressure to be maintained during the stroke of the main valve. Without the flow gag, inlet pressure would decrease to zero instantaneously which would not allow for a valid steam certification test and may damage the main disc upon closing. In addition, when testing contaminated valves, the flow gag minimizes the amount of potentially radioactive steam exhausted from the valve. However, the flow gag also causes a reaction force with the underside of the main disc at the instant of opening when steam at inlet pressure rushes into the small cavity between the main disc and the gag. This extra opening force on the test stand is significantly greater than the force acting on the valve under plant conditions and can result in damage to the valve as described above. NRC staff members with the Component Performance, NDE, and Testing Branch (NRR/DE/EPNB) intend to pursue questions regarding use of this test methodology with the ASME OM Code Committee at their next meeting.

.5 Review and assess the licensee's maintenance practices related to the maintenance/ inspection of the SRVs.

a. Inspection Scope

The inspectors reviewed SNC procedure 52GM-B21-005-0, "Main Steam Safety Relief Valve Maintenance," Revision 24, which prescribes SNC processes for maintenance on Target Rock 3-stage safety relief valves (SRVs).

b. Findings and Observations

The inspectors concluded that procedure 53GM-B21-005-0 provided adequate instructions for the removal, packaging and shipping of Hatch's SRVs to prevent damage or deterioration. The procedure stepped though the removal of the valve major components including quality control hold points to perform in-service inspection (ISI) in accordance with the American Society of Mechanical Engineers (ASME) Section XI code. No other maintenance was performed on SRVs on site by the licensee.

.6 Review and assess the testing laboratory's practices related to the receipt, storage, and handling of the SRVs prior to testing.

a. Inspection Scope

The inspectors reviewed PO SNG 10129025 from SNC to NWS, dated February 26, 2016, to perform as-found testing of Hatch Unit 1 SRVs installed during 1R26, main valve serial numbers: 92, 123, 216, 270, 272, 273, 274, 276, 309, 312, and 1245. The inspectors reviewed NWS procedures for receipt, storage and handling, NWS-QA-P10-1, "Receipt Inspection," Revision 2, and NWS-QA-P13-1, "Handling, Storage, and Shipping," Revision 0. The inspectors conducted a walkdown at NWS of their receiving, storage, and decontamination spaces. The inspectors interviewed the vendor quality control inspectors that conducted the receipt inspections for SRVs that failed to re-close, main valves serial numbers: 92, 270, and 274.

The inspectors reviewed NTS SRV-TCR-001, "Receiving inspection – Target Rock 3-Stage Safety Relief Valve," dated July 2009, SRV-TCR-002, "Receiving inspection – Target Rock 3-Stage Pilot/Base," dated January 2010, and SRV-TCR-019, "Receiving inspection – Target Rock 3-Stage Safety Relief Valve Body," dated January 2010. The inspectors reviewed NTS test procedure 1129, "Target Rock Three Stage Pilot-Operated Relief Valves, Model No. 0867F-001/09G-001 for Southern Nuclear Company Hatch Nuclear Plant," Revision 0, which included storage and handling instructions for Hatch SRVs. The inspectors did not interview personnel or conduct walkdowns at the NTS facility because the inspection was conducted at the NWS test facility.

b. Findings and Observations

The inspectors determined that NWS receipt, storage and handling of the Hatch 3-stage SRVs was in accordance with PO SNG 10129025 requirements and in accordance with procedures NWS-QA-P10-1 and NWS-QA-P13-1. The inspectors verified that NWS-QA-P-10-1 and NWS QA-P13-1 complied with the receipt, storage and handling requirements established in ANSI N45.2.2-1978, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants." The inspectors determined that the procedures were adequate to prevent safety-related components damage or deterioration. The inspectors verified the use of adequate labels and tagging to reflect current status of components. In addition, adequate component segregation was observed though out the spaces. The inspectors determined during interviews with quality control inspectors that instructions and procedures for receipt, storage and handling of the Hatch SRVs were consistently followed.

The inspectors determined that NTS receipt inspection procedures SRV-TCR-001, 002, and 019, were adequate to verify conformance with documented instructions, procedures, and drawings. The procedures verified the physical integrity of SRV components received, verified the components serial numbers and performed an inventory of items included in the shipping package. The inspectors determined that test procedure 1129 provided adequate guidance for the storage and handling of Hatch's SRVs at the NTS test facility.

.7 Review and evaluate the licensee's practices against vendor recommendations regarding maintenance and testing of 3-stage SRVs.

a. Inspection Scope

The team evaluated the licensee's implementation of maintenance practices versus vendor recommendations for 3-stage SRVs. The team reviewed vendor maintenance and inspection practices contained in the technical manual for SRV Model 0867F-001. The team additionally reviewed activities and documents relating to technical specification surveillances, pre-certification, augmented inspections, post-operation testing, refurbishment and recertification.

b. Findings and Observations

The team noted the following observations related to vendor recommended maintenance. First, the newer Target Rock 3-stage SRV model 0867F-001 had a limited operating history with documented mechanical issues. Specifically, Pilgrim Nuclear Power Station installed model 0867F valves and experienced mechanical fretting that resulted in one SRV's failure to open in February 2013 and another SRV failure to open in January 2015 (NRC Special Inspection Report, Pilgrim Nuclear Power Station - ADAMS ML15147A412). The team noted that the vendor technical manual for the model 0867F-001 recommends a 96-month inspection periodicity for the main stage seating surfaces, guide surfaces and piston rings while the licensee utilized a 24-month inspection maintenance frequency. The team noted that reducing the inspection frequency to the vendor recommended frequency could delay the identification of vibration induced fretting conditions in the SRV main stage assembly. The team noted that Hatch Technical Specifications did not require in-situ testing of safety relief valve main stages based, in part, on proven operating history of the previous 2-stage SRV design. The licensee utilized one of two testing laboratories to perform testing and refurbishment of the Hatch SRVs (NTS Laboratories and NWS Technologies). The inspectors concluded that the licensee's implementation of maintenance practices was consistent with vendor recommendations for 3-stage SRVs.

.8 Review and verify the licensee's reportability determination was in accordance with the reportability criteria in 10 CFR 50.72 and NUREG-1022.

a. Inspection Scope

The team assessed the licensee's reportability determination evaluation against the requirements and guidance contained in 10 CFR 50.72 and NUREG-1022 respectively. The team reviewed condition reports, operability determinations, technical specifications, design basis documents, the licensee's final safety analysis report (FSAR), and a vendor thermo-hydraulic evaluation regarding the plant design bases response to SRV's failing to re-close.

b. Findings and Observations

No findings or observations were identified.

.9 Assess the licensee's actions resulting from NRC generic communications, vendor technical bulletins, and industry operating experience related to 3-stage Target Rock SRVs.

a. Inspection Scope

The team evaluated the licensee's actions in response to Curtiss-Wright, Target Rock Division, 10 CFR Part 21 Report due to the potential to induce a defect during the testing of the relief valve model (3-stage Target Rock Model 0867F).

b. Findings and Observations

Curtiss-Wright Flow Control Company, Target Rock Division, issued a 10 CFR Part 21 Report due to the potential to induce a defect during the testing of the relief valve model (3-stage Target Rock Model 0867F) on June 30, 2015 (ADAMS ML15187A172). This report was issued due to Target Rock Model 0867F failing to open when demanded at Pilgrim in 2013 and again in 2015. Upon disassembly, valve main guide fretting damage due to piston ring wear was observed. Target Rock determined the root cause of the Pilgrim main guide fretting (due to piston/ring wear) resulted from excessive impact load during limited flow testing which relieved the torque applied to the piston/stem interface (de-torqueing) and led to the creation of a significant clearance between the piston and the main disc (de-shouldering). This loss of shoulder-to-shoulder contact allowed relative motion between the main piston and main disc. If excessive impact load damaged the lock nut or lock tab and shortened the main spring, plant vibratory loads could allow the piston to rotate circumferentially and rock relative to the stem shoulder further increasing the clearance between it and the stem. This action was time dependent in that increasing the amount of time the piston was exposed to these conditions would increase the propensity for fretting wear.

The team concluded that de-torqueing and de-shouldering during pre-service certification testing followed by vibration induced fretting of the stem-to-piston interface and the piston rings-to-guide interface was responsible for the damage to the Unit 1 main SRV valve internals observed at Hatch.

The Part 21 Report recommended that valves currently installed be inspected to ensure proper piston-to-stem shoulder engagement based on plant-specific indications of the potential for fretting. At the time this vendor recommendation was made, all Hatch SRVs on both units had been converted to the Model 0867F 3-stage design. Unit 1 valves had been installed during the spring of 2014, and Unit 2 had just completed a refueling outage that installed the Model 0867F SRVs. Hatch had four 0867F valves available for inspection that had operational time in service (and were not currently installed in the plant). Inspections were performed on these valves and none were de-shouldered or had indications of fretting wear. Based on this plant specific data with regard to the as-found condition of the piston/main disc shoulder and lack of fretting, the licensee concluded there was low potential for de-shouldering and subsequent in-plant fretting and did not schedule inspections of in-service valves prior to planned refueling outages.

Target Rock also recommended in the 10CFR Part 21 Report (for valves not yet installed) that additional inspections be performed on the main valve internals after the certification test. These inspections required the base assembly to be removed from the main body after as-left testing to inspect the threaded stem-to-main piston connection to ensure there was no de-shouldering caused by the testing. These inspections were performed on the valves currently installed in Hatch Unit 1 and the licensee concluded that the Unit 1 SRVs are operable.

The licensee conducted a mid-cycle maintenance shutdown of Unit 2 on May 20, 2016, to replace all 11 SRVs and inspect the main valve internals in accordance with vendor recommendations discussed in the Part 21 report. The post certification testing inspection of the replacement SRVs currently installed in Unit 2 has been completed and the licensee concluded that the newly installed replacement Unit 2 SRVs were operable.

.10 Collect data necessary to support completion of the significance determination process, if applicable.

a. Inspection Scope

A regional Senior Reactor Analyst used information that indicated three of the 11 SRVs from the prior Hatch Unit 1 operating cycle failed to fully re-close when tested on a test stand.

b. Findings and Observations

The failure of the SRVs to re-close was modeled in the NRC's risk model. A modified Hatch model was prepared where two stuck open valves would depressurize the plant without operator action per the licensee's thermal hydraulic analysis. Probabilities of zero, one, two or three valves sticking open were calculated by looking at a four valve opening combination that would be expected at the first lift pressure setpoint for the SRVs.

The NRC's SPAR model calculated a change in risk of about $1x10^{-5}$ per year of exposure, with the Small Loss of Coolant Accident - Operator Fails to Depressurize the Reactor - Human Reliability Analysis (HRA) failure rate significantly impacting the single valve stuck open sequences. For these cases, if HPCI or RCIC were injecting, the time available before core damage would be longer and reduce the failure rate by an order of magnitude (this was not credited in the model). A more realistic calculation result would be a mid to high 10^{-6}. Large Early Release Frequency (LERF) was estimated to exceed $1x10^{-6}$, but was estimated to be less than $1x10^{-5}$ The LERF result was in the SIT/AIT overlap region of Management Directive (MD) 8.3, "NRC Incident Investigation Program," and was the risk metric of choice for the NRC's initial follow-up inspection decision.

The team determined that a loss of the SRV's closure function could not be verified, due to the nature of the testing methodology. This invalidated the assumptions used in the initial MD 8.3 risk evaluation prior to the inspection. Without a confirmed loss of function, there is not a calculated increase in risk due to a valve failure to re-close upon opening.

.11 Identify any potential generic safety issues and make recommendations for appropriate follow-up action (e.g., Information Notices, Generic Letters, and Bulletins).

a. Inspection Scope

The team reviewed degraded SRVs for potential generic safety issues.

b. Findings and Observations

All Target Rock 2-stage and 3-stage SRVs have similarly designed main stage components. The main stage valve internals are assembled by screwing the main piston onto the main stem so that the piston moves inside the guide, installing a locking tab washer, and installing the stem nut against the washer's locking tab. The piston and stem nut are individually torqued to vendor specified values, and the locking tab is bent to capture both the stem nut and a groove in the piston. The stem has a shoulder that seats tightly against the piston shoulder and most of the valve actuation force is carried by the stem and piston shoulders.

Loss of tight shoulder-to-shoulder contact (i.e., "de-shouldering") can cause the onset of thread damage. Thread damage can begin with the first actuation on the test stand, resulting in a loss of torque. Over time, vibration from normal plant operations can cause fretting and wear of the valve stem shoulder and threads. The piston rocks in the guide and wears grooves where the piston rings contact the guide. Eventually the piston could significantly cock on the stem and wedge in the guide during valve actuation, which could prevent proper opening or closing of the valve.

NRC Information Notice 2003-01, "Failure of a Boiling Water Reactor Target Rock Main Steam Safety/Relief Valve," (ADAMS ML030140543) described de-shouldering and resultant main stage valve damage for Target Rock 2-stage valves. While the main stage damage described in the information notice is similar to the main stage damage observed in the Hatch Unit 1 3-stage valves, the root cause is apparently different. The root cause analysis for the earlier 2-stage design concluded that the lead thread of the piston was contacting the fillet of the stem shoulder, preventing tight shoulder-to-shoulder contact. Since the piston was not adequately attached to the stem, operational vibration and valve actuation caused thread damage and eventual valve failure. The valve vendor (Curtiss-Wright) subsequently developed changes to the inspection and refurbishment procedures and manufacturing tolerances to ensure proper shoulder-to-shoulder contact during valve assembly. Licensees also began to conduct more frequent inspection and maintenance activities for these valves. These corrective actions have been effective in addressing the de-shouldering concern for Target Rock 2-stage valves.

Target Rock reported that the root cause of de-shouldering in the newer Model 0867F series 3-stage valves is due to excessive impact load during limited flow testing on the test stand which relieves the torque applied to the piston-to-stem interface. When the impact load is much greater than the local yield strength, the preload on the joint is not only removed but clearance is established between the stem and piston shoulders. Additional cycles increase the clearance between the stem and piston shoulders and cause plastic deformation, rounding, and galling of the threads on the stem. This is different than the minor local yielding that occurs during normal limited flow cycling for all other two and 3-stage designs. There are a number of differences between the 0867F and earlier designs which incrementally reduce opening velocity, and therefore impact loads, on the earlier designs, during limited flow testing. These designs allow no plastic deformation of the piston shoulder and threads.

The potential generic safety issue is that after de-shouldering occurs, a piston could cock on the stem and wedge in the guide during valve actuation which could prevent proper opening or closing of the valve. The issue described above is limited to Target Rock Model 0867F three-stage valves. Target Rock's Part 21 report identified that Pilgrim, Fitzpatrick, Hatch and Hope Creek had either received or ordered Model 0867F SRVs. In their June 30, 2015, Part 21 report (ADAMS ML15187A172), Target Rock indicated that they are working with these sites to provide recommended inspection procedures and test fixtures to detect and correct de-shouldering for valves currently installed and for valves not yet installed.

In the Part 21 report, Target Rock also indicated that they are pursuing an enhanced Model 0867F design that will be better able to mitigate the extreme loads encountered in limited flow testing while continuing to meet the in-service specification requirements for the valves. The new design will be validated through both limited flow and full flow testing as well as through post-test inspections. Redesign and qualification testing is expected to be completed by the end of June 2016.

Following the original Pilgrim event in January 2015, the NRC screened the event into its operating experience process. This process provides a framework for evaluating issues or events that meet certain criteria, and for documenting the agency's actions. The staff is focused on overall industry response to 3-stage SRV performance and material condition. Specifically, the operating experience evaluation will look at activities to minimize the potential for damage to Model 0867F valves during limited flow testing. The NRC plans to perform a vendor inspection to verify implementation of corrective actions including design changes and modification to testing methods. In addition, the NRC continues to track operational and testing issues with the remaining Model 0867F valves installed in U.S. plants. At the conclusion of its operating experience evaluation, the staff will follow up with the appropriate regulatory response.

4OA6 Meetings, Including Exit

On April 27, 2016, the inspection team presented the inspection results to Mr. David Vineyard and other members of the licensee's staff. The inspectors confirmed that proprietary information was reviewed and controlled to prevent improper disclosure.

ATTACHMENTS: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

Key Points of Contacts

Licensee personnel:

B. Anderson, Health Physics Manager
G. Brinson, Maintenance Director
C. Collins, Principal Licensing Engineer
A. Giancatarino, Engineering Director
G. Johnson, Regulatory Affairs Manager
R. Spring, Plant Manager
M. Torrance, Design Engineering Manager
W. Williams, System Engineer

Vendor personnel:

J. Armstrong, NWS technologies Quality Control Inspector
R. Fleming, NWS technologies Quality Control Inspector
J. Gibson, NWS technologies Quality Assurance Manager
J. Ledsome, NWS technologies Quality Control Inspector
M. Ledsome, NWS technologies Radiation Safety Officer
T. Nederostek, NWS technologies Vice President of Operations
A. DiMeo, Target Rock Senior Manager Design Engineering

List of Documents Reviewed

Purchase Orders

SNC PO SNG101129025 to NWS technologies to conduct As-Found testing of 11 Target Rock 3-stage 0867F Safety Relief Valves, dated February 26, 2016

SNC PO SNG10060848 to Curtiss-Wright Flow Control Corp to refurbish, inspection/repair and perform steam certification testing for 10 Target Rock 3 stage 0867F Safety Relieve Valves, Revision 2, dated January 30, 2014

Procedures

NWS technologies procedure NWS-QA-P13-1, Handling, Storage, and Shipping, Revision 0, dated November 20, 1996

NWS technologies procedure NWS-QA-P-10-1, Receipt Inspection, Revision 2, dated May 14, 2008

NWS technologies procedure NWS-T-91, NWS Test Procedure Hatch Nuclear Plant Target Rock 0867F 3 Stage Main Steam Relief Valves, Revision 0, dated January 24, 2013

Wyle/NTS test procedure 1129, Test Procedure for Target Rock Three Stage Pilot-Operated Relief Valves Model No.0867F-001/09G-001 for Southern Nuclear Company Hatch Nuclear Plant, Revision 0, dated February 23, 2011

SNC procedure 52GM-B21-005-0, Main Steam Safety Relief Valve Maintenance, Revision 24, dated July 31, 2014

Target Rock procedure S64212, Target Rock packing and shipping Procedure 3 Stage Safety Relief Valves, Revision 2, dated October 17, 2014

Documents

NWS Safety Valve Test Traveler, Receipt Inspection Main Valve Body S/N 270, performed on March 24, 2016
NWS Safety Valve Test Traveler, Receipt Inspection Main Valve Body S/N274, performed on March 22, 2014
NWS Safety Valve Test Traveler, Receipt Inspection Main Valve Body S/N 92, performed on March 24, 2016
NWS Customer Equipment Anomalies Report 16-104, SRV Main Valve S/N 272 Failure to Close, dated April 5, 2016
NWS Customer Equipment Anomalies Report 16-103, SRV Main Valve S/N 274 Failure to Close, dated April 5, 2016
NWS Customer Equipment Anomalies Report 16-101, SRV Main Valve S/N 92 Failure to Close, dated April 5, 2016
NTS/Wyle Certification Test Report T71125-1, Steam Certification Testing Base Assembly S/N: 10; Main Body: 1245; Pilot: N/A; Air Operator: 111; Solenoid: 40, performed on December 18, 2013
NTS/Wyle Certification Test Report T71125-2, Steam Certification Testing Base Assembly S/N: 7; Main Body: 123; Pilot: N/A; Air Operator: 124; Solenoid: 45, performed on December 20, 2013
NTS/Wyle Certification Test Report T71125-3, Steam Certification Testing Base Assembly S/N: 5; Main Body: 312; Pilot: N/A; Air Operator: 71 Solenoid: 54, performed on December 20, 2013
NTS/Wyle Certification Test Report T71125-4, Steam Certification Testing Base Assembly S/N: 15; Main Body: 3/273; Pilot: N/A; Air Operator: 70; Solenoid: 58, performed on December 18, 2013
NTS/Wyle Certification Test Report T71125-5, Steam Certification Testing Base Assembly S/N: 38; Main Body: 4/216; Pilot: N/A; Air Operator: 78; Solenoid: 53, performed on December 18, 2013
NTS/Wyle Certification Test Report T71125-6, Steam Certification Testing Base Assembly S/N: 10; Main Body: 1/270; Pilot: N/A; Air Operator: 77; Solenoid: 51, performed on December 18, 2013
NTS/Wyle Certification Test Report T71125-7, Steam Certification Testing Base Assembly S/N: 3; Main Body: 92; Pilot: N/A; Air Operator: 74; Solenoid: 330, performed on December 18, 2013
NTS/Wyle Certification Test Report T71125-8, Steam Certification Testing Base Assembly S/N: 2; Main Body: 309; Pilot: N/A; Air Operator: 68; Solenoid: 55, performed on December 18, 2013
NTS/Wyle Certification Test Report T71125-10, Steam Certification Testing Base Assembly S/N: 40; Main Body: 8/272; Pilot: N/A; Air Operator: 129; Solenoid: 357, performed on December 19, 2013
NTS/Wyle Certification Test Report T71125-11, Steam Certification Testing Base Assembly S/N: 64; Main Body: 2/276; Pilot: N/A; Air Operator: 108; Solenoid: 38, performed on December 19, 2013
NTS/Wyle TR-FRSDB-12322-001-00, Target Rock Field Service Data Book Steam Certification Results for Base Assembly S/N: 18 Main Body S/N: 274, dated December 18, 2012.