UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

April 4, 2018

Mr. George Lippard III
Vice President
Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 800
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - NRC DESIGN BASES ASSURANCE INSPECTION (PROGRAMS) REPORT NUMBER 05000395/2018010

Dear Mr. Lippard,

On February 16, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Virgil C. Summer Nuclear Station, Unit 1, and discussed the potential findings with you and other members of your staff. On March 29, 2018, a re-exit meeting was conducted via teleconference to present the final inspection results to Mr. S. Zarandi and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented three findings of very low safety significance (Green) in this report. Three of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement; and the NRC resident inspector at the Virgil C. Summer Nuclear Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

Sincerely,

/RA/

Marvin D. Sykes, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-395
License Nos. NPF-12

Enclosure:
Inspection Report 05000395/2018010

cc: Distribution via ListServ

SUBJECT: VIRGIL C. SUMMER NUCLEAR PLANT - NRC DESIGN BASES ASSURANCE INSPECTION (PROGRAMS) REPORT NUMBER 05000395/2018010

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DATE	3/28/2018	3/8/2018	3/28/2018	4/2/2018	4/2/2018	4/2/2018	
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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number(s):	05000395
License Number(s):	NPF-12
Report Number(s):	05000395/2018010
Enterprise Identifier:	I-2018-010-0023
Licensee:	South Carolina Electric & Gas (SCE&G) Company
Facility:	Summer Nuclear Plant, Unit 1
Location:	Jenkinsville, SC
Inspection Dates:	January 29 to February 16, 2018
Inspectors:	T. Fanelli, Senior Reactor Inspector, M. Riley, Reactor Inspector, C. Franklin, Reactor Inspector
Approved By:	M. Sykes, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring licensee's performance by conducting a Design Bases Assurance Inspection (Programs) at Virgil C. Summer Nuclear Station, Unit 1, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. NRC findings, violations, and additional items are summarized in the table below.

List of Findings and Violations

"Failure to Justify Activation Energy for Valcor SOV XVX06050A"			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000395/2018010-01 Closed	None	71111.21N-Design Bases Assurance Inspection (Programs)
The NRC identified a Green finding and associated Non-cited Violation (NCV) of 10 CFR 50.49 (e)(5) when the licensee failed to justify the basis for the activation energy used for Valcor solenoid operated valve (SOV) XVX06050A in accordance with Regulatory Guide (RG) 1.89 Section C.5.c.			

"Failure to Verify the Seismic Qualification of Valcor Solenoid Operated Valve XVX06050A"			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000395/2018010-02 Closed	None	71111.21N-Design Bases Assurance Inspection (Programs)
The NRC identified a Green finding and associated Non-cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," when the licensee failed to verify the adequacy of design for the seismic qualification of valve XVX06050A in accordance with IEEE 344-1971.			

"Inadequate Radiation Harsh Environmental Qualification of Reactor Building Spray Pump A"			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000395/2018010-03 Closed	None	71111.21N-Design Bases Assurance Inspection (Programs)
The NRC identified a Green finding and associated NCV of 10 CFR 50.49 (e)(4) when the licensee failed to verify that RB spray pump A could perform its function under the radiation conditions expected during an accident in accordance with Section 2.1(3)(a) of NUREG 588.			

Additional Tracking Items

Unresolved Item (Open)	"Unjustified Qualified Life for ASCO Valves" URI 05000395/2018010-04	Section 71111.21N-Design Bases Assurance Inspection (Programs)

Unresolved Item (Open)	“Potential High Radiation Dose Areas with Unqualified Components” URI 05000395/2018010-05	Section 71111.21N-Design Bases Assurance Inspection (Programs)

Unresolved Item (Open)	“Potential Unjustified Activation Energy for Barton Transmitters” URI 05000395/2018010-06	Section 71111.21N-Design Bases Assurance Inspection (Programs)

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards."

REACTOR SAFETY

71111.21N - Design Bases Assurance Inspection (Programs)

Programs (Environmental Qualification) (10 Samples)

The inspectors evaluated environmental qualification program implementation by reviewing the licensed environmental qualification requirements for the following components and a sample of their associated subcomponents from January 29, 2018, to February 16, 2018:

1) XVG01611A 20-2 Chicago Fluid Power, Feed Water Isolation / C-2233-AB

2) XVM02801C 20C Chicago Fluid Power, Main Steam Isolation / 321X-21

3) IPT00456 Barton Pressurizer Level / 763

4) MPP00038A General Electric Reactor Building Spray Pump / K Horizontal Frame 8210S 5K821055C44

5) IPCV00445A Pressurizer Power Operated Relief Valves (ASCO)

6) XVX06050A 20 Hydrogen Sampling Valves Valcor / V526-5292-20

7) RMG00007 Radiation Monitor Victoreen / 877-1

8) IPV02020 20A Main Steam Power Operated Relief Valves ASCO / WP-HVA-206-381-3RVU

9) XRP00010\I C-42 D.G. O'Brien / C-42 Plugs Low Voltage Power/Control

10) IFV03331 20B-HC NAMCO / EC210-34001, EC210-44010/20

INSPECTION RESULTS

"Failure to Justify Activation Energy for Valcor SOV (XVX06050A)"			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000395/2018010-01 Closed	None	71111.21N-Design Bases Assurance Inspection (Programs)

Introduction: The NRC identified a Green finding and associated Non-cited Violation (NCV) of 10 CFR 50.49 (e)(5) when the licensee failed to justify the basis for the activation energy used for Valcor solenoid operated valve (SOV) XVX06050A in accordance with Regulatory Guide (RG) 1.89 Section C.5.c.

Description: The qualification of the Valcor SOVs, completed in 1979, used the 10ºC rule to determine the accelerated aging rate, which was equivalent to a 0.831 eV activation energy derived for Valcor's ethylene propylene rubber (EPR). The inspectors determined that 0.831 eV for EPR, although realistic, it was not the most limiting identified for EPR. Valcor originally qualified the SOVs for 40 years at 120ºF, however many of the valves are normally energized and will see temperatures exceeding 120ºF. The SOV, XVX06050A, is a normally energized open valve that de-energizes to close on a containment isolation phase 'A' signal and opened post-accident for hydrogen analyzing in the reactor building. In 1988, Impell Corporation, the licensee's contractor, reanalyzed the qualification and determined that DuPont Tefzel insulation was the most limiting component instead of EPR and that a 50% loss of tensile strength was the limiting failure mechanism at 0.95 eV activation energy. To extrapolate a new activation energy, Impell estimated data points from a rudimentary log life plot that did not have any actual test data points. Impell obtained the plots from a DuPont Tefzel design handbook which also contained the log life plot for the elongation to break failure parameter of Tefzel, which appeared more limiting than tensile strength. Because the new activation energy extrapolation did not use actual test data, the extrapolation of that data was less limiting than the original qualification activation energy, and the elongation to break failure parameter was not evaluated, the team determined the new activation energy was not justified.

FSAR Section 3.11.2.1.3 stated that the environmental qualification of Class 1E equipment is in conformance with RG 1.89, Rev. 1. Section C.5.c of the RG stated that "the aging acceleration rate and activation energies used during qualification testing and the basis upon which the rate and activation energy were established should be defined, justified, and documented." The licensee did not find the original qualification activation energy to be in error or non-conservative. The licensee chose to develop an activation energy from less limiting log life plots, which was non-conservative. In addition, without actual data for the log life plots, the licensee was unable to demonstrate acceptable margins for uncertainty. The team determined that the valve would have exceeded its qualification based on the original qualification and unjustified use of the new activation energy.

Corrective Actions: On February 19, 2018, the licensee entered this issue into their corrective action program as CR 18-00754 and performed an immediate determination of operability to verify that the valve could still perform its intended safety function.

Corrective Action Reference: CR 18-00754

Performance Assessment: The failure to justify the basis upon which the activation energy of Valcor SOV XVX06050A was established in accordance with RG 1.89 Section C.5.c was a performance deficiency (PD). The PD was determined to be more than minor because it adversely affected the SSC and Barrier Performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to justify the activation energy used for Tefzel adversely affected the reliability of the solenoid to maintain its qualification over the entire 40 year qualified life of the plant. The team used inspection manual chapter (IMC) 0609, Att. 4, "Initial Characterization of Findings," issued December 7, 2016, for barriers, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, and heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. Since the underlying cause of the issue occurred in 1988, the team determined that no crosscutting aspect was applicable because the finding was not indicative of current licensee performance.

Enforcement: Title 10 CFR 50.49 (e)(5) states "Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life." Contrary to the above, since August 30, 1988, the licensee failed to age Valcor SOV XVX06050A to its end of life condition and to replace the equipment at the end of its designated life. This violation is being treated as an NCV, consistent Section 2.3.2 of the Enforcement Policy.

"Failure to Verify the Seismic Qualification of Valcor Solenoid Operated Valve XVX06050A"			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000395/2018010-02 Closed	None	71111.21N-Design Bases Assurance Inspection (Programs)

Introduction: The NRC identified a Green finding and associated Non-cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," when the licensee failed to verify the adequacy of design for the seismic qualification of valve XVX06050A in accordance with IEEE 344-1971.

Description: Calculation VCS-0423-DC-1, "Valcor Voltage and Current Reducing Resistors," Rev. 0, dated September 10, 1981, located in Tab E1 of EQDP-H-VO4-V01 for solenoid operated valve XVX06050A, indicated a 300 ohm resistor was in series with the valve and that it reduced the voltage in the coil to approximately 32VDC at minimum conditions. The team questioned if the valve was seismically qualified at the lower voltage since the seismic qualification in test report QR 52600-515, Section 4.2.5, "Seismic Vibrations," stated that it was performed at 108VAC. The team noted that the Valcor SOV was not installed in the same configuration that it was seismically qualified. The failure to ensure the valve was seismically qualified, as configured, did not ensure that damage would not occur during a seismic event. FSAR Section 3.10 stated that seismic qualification must be done in

accordance with IEEE 344-1971. Section 3.2.2.2 of IEEE 344-1971 states the device being tested should demonstrate its ability to perform its intended safety function and sufficient monitoring equipment should be used to evaluate its performance. The team determined that the licensee did not demonstrate the seismic qualification of valve XVX06050A in its current plant configuration at reduced voltage.

Corrective Actions: On February 15, 2018, the licensee entered this issue into their corrective action program as CR 18-00686 and performed an immediate determination of operability to verify that the valve could still perform its intended safety function.

Corrective Action Reference: CR 18-00686

Performance Assessment: The licensee's failure to verify the adequacy of the seismic design and qualification of valve XVX06050A in accordance with IEEE 344-1971 was a performance deficiency (PD). The PD was determined to be more than minor because it adversely affected the Design Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to verify the adequacy of design for seismic qualification of the valve resulted in the valve being installed in an unqualified configuration. The team used inspection manual chapter (IMC) 0609, Att. 4, "Initial Characterization of Findings," issued December 7, 2016, for barriers, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, and heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. Since the underlying cause of the issue occurred on August 30, 1988, the team determined that no crosscutting aspect was applicable because the finding was not indicative of current licensee performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III "Design Control," requires, in part, that "The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, since August 30, 1988, the licensee failed to verify valve XVX06050A was seismically qualified in its current configuration in accordance with IEEE 344-1971. This violation is being treated as an NCV, consistent Section 2.3.2 of the Enforcement Policy.

"Inadequate Radiation Harsh Environmental Qualification of Reactor Building Spray Pump A"			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000395/2018010-03 Closed	None	71111.21N-Design Bases Assurance Inspection (Programs)

Introduction: The NRC identified a Green Finding and associated NCV of 10 CFR 50.49 (e)(4) when the licensee failed to verify that RB spray pump A could perform its function under the radiation conditions expected during an accident in accordance with Section 2.1(3)(a) of NUREG 588.

Description: During the review of EQDP-H-MO1-G03 for RB spray Pump A, the team noted that the pump was qualified for a maximum harsh environment of $1x10^6$ radiation absorbed dose (rad); however, the total integrated dose (TID) was expected to be greater than $6.1x10^6$ rad TID over its 40 year life. Tab F1 of the EQDP, containing the equipment qualification report of the motors dated June 1977, stated that the maximum integrated radiation dose justified by the report over the 40 year operating life of the motor was $1x10^6$ rads. The EQDP stated that component data shows that "all components are suitable for the rated $1x10^6$ rads integrated dose with the exception of (a) unfilled polyester resin and (b) the Dacron felt. In all cases, the polyester resins are filled to various degrees with glass or similar products. Such filling of the resin results in a significant increase in the radiation resistance of the combination -- as high as $9x10^8$ rads. The Dacron felt by itself, at a threshold resistance of $8.6x10^5$ rads, approaches the required radiation resistance but the felt is designed to be saturated with the impregnating epoxy resin and occurs only in this state. No specific data is available on the radiation resistance of the combination (Dacron filled epoxy), but the evidence indicates that the combination will exceed the required $1x10^6$ rads." The team noted that the expected TID dose over the 40 year life of the RB spray pump A motor exceeded the original qualification provided in this test report.

In order to ensure the pump was qualified for its radiation environment, the licensee had Impell Corporation perform Calculation 0980-036-030, "Qualified Radiation Levels for GE Motors," Rev. 0, in August 31, 1988, which concluded that the motor was qualified for $1.5x10^7$ rads. The re-analysis was not based on partial type testing of the motor or a similar motor in accordance with NUREG-0588, but only reinterpreted the same material information previously provided by GE. The team noted that the reanalysis made different assumptions than GE did on the material characteristics of an unknown polyester resin fill material and Dacron felt. For the polyester resin, Impell could not determine what the fill material was or how much fill was used, but determined that it had a higher radiation resistance. For the Dacron felt, Impell assumed that the Dacron would not be a weak link in radiation resistance because of the epoxy. These assumptions were used to justify increasing the radiation qualification of the RB spray pump motor. The team determined that the original qualification of $1x10^6$ rads was appropriate and was not proven to be inadequate by Impell because of the uncertainties documented by GE, and the lack of actual type testing information for the motor to support the Impell assumptions.

FSAR Section 3.11.2 states that the licensee is committed to NUREG 588 Category II requirements. Section 2.1.2 of NUREG 588 states "The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached." Section 2.1(3)(a) of NUREG 588 states "Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident." The team determined that the basis for raising the radiation qualification was not justified and that the qualification test report did not demonstrate that RB spray pump A was qualified over its 40 year operating life.

Corrective Actions: On February 16, 2018, the licensee entered this issue into their corrective action program as CR 18-00707 and performed an immediate determination of operability to verify that the pump could still perform its intended safety function.

Corrective Action Reference: CR 18-00707

Performance Assessment: The licensee's failure to justify that RB spray pump 'A' could perform its function under the radiation conditions expected during an accident in accordance with Section 2.1(3)(a) of NUREG 588 was a PD. The PD was determined to be more than minor because it adversely affected the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the failure to qualify the pump to expected radiation conditions adversely affects the pump's capability to perform its intended safety function during a design basis accident. The team used inspection manual chapter (IMC) 0609, Att. 4, "Initial Characterization of Findings," issued December 7, 2016, for mitigating systems, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the qualification of a mitigating structure, system, and component (SSC), and the SSC maintained its operability. Since the underlying cause of the issue occurred on August 31, 1988, the team determined that no crosscutting aspect was applicable because the finding was not indicative of current licensee performance.

Enforcement: Title 10 CFR 50.49 (e)(4) requires, in part, that the electric equipment qualification program must include and be based on radiation, and the radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects. Contrary to the above, since August 31, 1988, the licensee failed to qualify RB spray pump 'A' to the total dose expected during normal operation over the installed life of the pump and during the most severe DBA. This violation is being treated as an NCV, consistent Section 2.3.2 of the Enforcement Policy.

Unresolved Item (Open)	"Unjustified Qualified Life for ASCO Valves" URI 05000395/2018010-04	71111.21N-Design Bases Assurance Inspection (Programs)

Description: The NRC opened a Unresolved Item (URI) to determine if a performance deficiency was more than minor. In 1993, the licensee's contractor, Impell Corporation, re-analyzed the qualified life established by ASCO qualification report AQR-67368 and a field notification from ASCO dated 10/27/1989. Impell erroneously used the heat rise temperatures from the field notification for both the AQR-67368 test samples accelerated aging temperature and the actual service temperatures in various plant locations. Replacing the actual test specimen's documented accelerated aging temperature with an assumed temperature was not justified. As a result, when using the actual temperature identified in the qualification report, many of these solenoids are currently beyond their qualified lives.

The licensee provided an alternate heat rise test report less limiting than the ASCO testing to justify that the ASCO valves were within their service lives, report 8058-001-2000-RA-0001-R00, Environmental Qualification Temperature Test of ASCO 206 and NP Series Solenoid Valves, dated June 2000. The team's evaluation must determine whether the alternate report is applicable to the licensee, and, if so, whether the test report indicated that the ASCO testing was invalid to conclude that the valves are currently within their qualified lives.

NUREG-0588 Section 4(6) and Regulatory Guide 1.89, Rev. 1, Regulatory Position 5.c, required, in part, that the aging acceleration rate and the basis upon which it was established be described, documented, and justified. The team determined that the failure to justify the aging acceleration rate was a performance deficiency. However, a review of the additional information is warranted to determine if the performance deficiency is more than minor. The licensee entered the performance deficiency into their corrective action program as CR-18-00175 and determined that preliminary calculations indicated that the ASCO valves are currently operable based on the additional information provided for review.

Unresolved Item (Open)	"Potential High Radiation Dose Areas with Unqualified Components" URI 05000395/2018010-05	71111.21N-Design Bases Assurance Inspection (Programs)

Description: The NRC opened a URI to determine if a performance deficiency exists. The licensee did not perform analysis to determine the radiation exposure to shielded components adjacent to electrical and blank penetrations on the outboard side through containment. As a result, many mild environment components may be adversely affected. The inboard side of the penetrations is exposed to rad levels approaching $9X10^7$ rads and the out board side is shielded by thin steel plates with electrical pass-thru holes. The inspectors noted that there were many areas of the plant identified as mild environments with unanalyzed penetrations. For example, the inspectors observed that the two trains for the plant service water were adjacent to unanalyzed penetrations. The components adjacent to the outboard side of the penetrations may be unqualified for service conditions expected during the most severe DBA as required by 10 CFR 50.49(e)(4).

NUREG-0588 Section 1.4 "Radiation Conditions Inside and Outside Containment," required, in part, that "(8) Shielded components need be qualified only to the gamma radiation levels required…" and that "(12) Equipment that may be exposed to radiation doses below 10^4 rads should not be considered to be exempt from radiation qualification, unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values." The licensee provided a white paper for this issue that asserts that consideration of radiation streaming was not part of their licensing basis, thus enforcement would be addressed through a backfit analysis in accordance with 10 CFR 50.109. The team must determine whether the site licensing basis required consideration of radiation streaming and whether a backfit analysis would be appropriate in lieu of enforcement. The licensee captured this issue in their corrective action program as CR-18-00684 and determined that "the process for qualification of equipment used was found acceptable per the VCS SER. Further evaluation will be performed under this CR but currently all components are qualified to their expected operating conditions and will perform their design functions. At worst, the EQ life of components may be reduced. All equipment in penetration areas are operable."

Unresolved Item (Open)	"Potential Unjustified Activation Energy for Barton Transmitters" URI 05000395/2018010-06	71111.21N-Design Bases Assurance Inspection (Programs)

Description: The contractor, Impell Corporation, changed the activation energy for the Barton transmitters from 0.5 eV to 0.78 eV. The 0.78 eV was based upon an academic paper documenting experimental work, apparently, performed for the early space program and apparently first published in 1965. The paper cautioned the reader that the methods used were experimental and were not validated. A 0.5 eV activation energy for electronics was documented by the Electric Power Research Institute (EPRI) report NP-1558, which attributed

it to electron migration of aluminum. The report was available to the licensee at the time of the change. Reports published by the Institute of Electrical and Electronics Engineers (IEEE) indicated that activation energies for various electronic failure modes could range from 0.5-0.66. Impell did not document an independent failure modes and effects analysis to justify the activation energy that they used.

The licensee did not find the original qualification activation energies to be in error or non-conservative. The licensee chose to use less limiting activation energies that may not have been proven to be justified. In addition, the licensee was unable to demonstrate acceptable margins for extrapolation uncertainty. FSAR Section 3.11.2.1.3 stated that the environmental qualification of Class 1E equipment is in conformance with RG 1.89, Rev. 1. The RG in Section C.5.c stated that "the aging acceleration rate and activation energies used during qualification testing and the basis upon which the rate and activation energy were established should be defined, justified, and documented." NUREG 0588 Section 5(2), "Qualification Documentation," specified, in part that "a certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program." The licensee captured this issue in their corrective action program as CR-18-00500, and determined that the "NRC challenged the qualified life for Barton installed as IPT00456 based on an activation energy. VC Summer engineering does not agree with the NRC, nor do the OEMs Barton, Weed/Foxboro and Rosemount who have reviewed their prior research and state that it is suitable and adequate for our applications." The team must determine whether the activation energy used for the Barton transmitters was appropriate and, if not, whether the licensee had the responsibility to verify the information provided by their vendors and contractors.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

The inspectors confirmed that proprietary information was controlled to protect from public disclosure.

- On February 16, 2018, the inspector presented the Program inspection results to Mr. G. Lippard, and other members of the licensee staff. On March 29, 2018, a re-exit meeting was conducted via teleconference to present the final inspection results to Mr. S. Zarandi and other members of the licensee's staff.

DOCUMENTS REVIEWED

Corrective Action Documents written as a result of the inspection

CR-18-00175, NRC Inspector for the EQ DBAI questioned the qualified life qualified life of ASCO solenoid valve XVG09684B-20-CC

CR-18-00194, During the NRC DBAI (EQ) Inspection, the NRC questioned the basis for extending O-ring replacement schedule beyond the vendor recommended change frequency of 5 years for Allied 321X solenoid valves.

CR-18-00322, NRC DBA EQ program Inspection selected 10 sample EQDP for inspection. Engineering review of the sample EQDPs found two where the component replacement schedule of EQDB report 985 are not currently reflected in CMMS PMs

CR-18-00411, NRC DBAI EQ Program Inspection selected Namco hermetic connector, model EC-210, equipment id: IFV03331-20B-HC-FW, installed in IB-436-02 known as EQ Zone IB-07. Engineering reviewed the aging calculation DC0011A-091 and determined it is not conservative.

CR-18-00461, NRC DBAI EQ Inspection – During EQ document reviews a minor typo error (unit error) was discovered in an aging calculation.

CR-18-00500, NRC identified NRC DBAI EQ Program inspection challenged the qualified life for Barton installed as IPT00456 and the calculation use of 0.78 electron Volts (eV) for the activation energy.

CR-18-00501, NRC identified NRC DBAI EQ Program Inspection challenged EQ qualified life for the seals and O-rings in Allied MSIVs.

CR-18-00592, NRC DBA EQ Program inspection noted difficulty correlating the qualification of cable and connector interfaces with the tested equipment configurations

CR-18-00593, NRC DBA EQ Program inspection challenged the EQ qualified life for the O-ring in CE cable connector (on the core exit thermocouples).

CR-18-00645, NRC DBAI EQ Program Inspector asked about the Thermal Aging Analysis for the GE Reactor Building Spray Pump Motor.

CR-18-00654, NRC DBAI EQ Inspector questioned if the temperature measured was conservative in the Chicago Fluid Power (CFP) test report, Appendix C of report CCL A-795-89 in EQDP-H-VO4-C16-1, Tab F1.

CR-18-00677, NRC DBA EQ inspection team asked for justification as to why the use of the ASCO FV heat rise drawing values are representative of the actual temperature that the test specimens were exposed to during thermal aging.

CR-18-00684, NRC Design Basis Assurance Inspection questioned whether post-accident radiation streaming has been modeled at Environmental Zones that include electrical penetration areas

CR-18-00686, NRC DBA EQ inspection team questioned the adequacy of seismic qualification testing for Valcor solenoid valve XVX06050A-HR

Drawings

B-208-054, Post Accident H2 Removal Loop A (IRB) XVX-6050A

B-210-324, Wiring Diagram – Reactor Building, Rev.3

D-302-011, Main Steam, Rev. 40

D-302-661, Reactor Building Spray Steam, Rev. 35

D-302-861, Post Accident Hydrogen Removal and Alternate Purge System, Rev. 33

Equipment Qualification Packages
EQDP-H-C05-D01-1, DG O'Brien Triax Connector, Rev.2
EQDP-H-C06-N01, Namco Hermetic Connectors, Rev.2
EQDP-H-CA6-C06, CETS MI Cable System, Rev.1
EQDP-H-MO1-GO3, GE EFW and RB Spray Pump Motors, Rev. 0
EQDP-H-V04-C16-1, Chicago Fluid Power Solenoid Valve, Rev. 0
EQDP-H-VO4-A13, Allied Solenoid Valves, Rev. 1
EQDP-H-VO4-VO1, Valcor Solenoid Valves, Rev. 1
EQDP-IN6-V05, Victoreen Radiation Monitoring System, Rev.0
EQDP-H-VO4-A11 - ASCO SOLENOID VALVES, Rev 4
EQDP-H-IN1-B05-1 2-D, Barton Transmitters, Rev 2

Procedures
EMP-405.003, Termination and De-Termination of Cables 480 Volts and Below, Rev.17
ES-382, Inventory Management, Rev. 3
ICP-235.042, Feedwater Isolation Valves XVG01611 A, B, &C Rebuild Procedure, Rev. 6
PR-01, Development of Shelf Life Criteria, Rev. 2
VCS-ES-0324, Establishment of Shelf Life Criteria, Rev. 0
VCS-ES1-0381, Materials Management, Rev. 0

Work Orders
1007346 1410298
1007505
1110070
1410214

Miscellaneous Documents
883929-MPS-5EFPR-001, Installation and Maintenance Manual for Mineral Insulated Triaxial Cable Assemblies, Rev. 00
ECR 50799, Reactor Coolant Pump Seal Replacement, Rev. 5
ECR 50897, Replacement Reactor Service Structure-Integrated Head Assembly (IHA), Rev. 5
PO 9086255, Mineral Insulated Triaxial Cable Certificate of Conformance, Rev. A