UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE N.E., SUITE 1200
ATLANTA, GEORGIA 30303-1200

December 22, 2020

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co. Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT – DESIGN BASIS ASSURANCE INSPECTION (PROGRAMS) INSPECTION REPORT 05000321/2020010 AND 05000366/2020010

Dear Ms. Gayheart:

On December 11, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Edwin I. Hatch Nuclear Plant. On July 16, 2020, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

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

A licensee-identified violation which was determined to be of very low safety significance is documented in this report. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the significance or severity of the violations documented in this inspection report, 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 Edwin I. Hatch Nuclear Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; and the NRC Resident Inspector at Edwin I. Hatch Nuclear Plant.

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 Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

Sincerely,

/RA/

James B. Baptist, Chief
Engineering Br 1
Div of Reactor Safety

Docket Nos. 05000321 and 05000366
License Nos. DPR-57 and NPF-5

Enclosure:
As stated

cc w/ encl: Distribution via LISTSERV®

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT – DESIGN BASIS ASSURANCE INSPECTION (PROGRAMS) INSPECTION REPORT 05000321/2020010 AND 05000366/2020010 dated December 22, 2020

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DATE	12/18/2020	12/15/2020	12/21/2020	12/18/2020	12/22/2020

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000321 and 05000366
License Numbers:	DPR-57 and NPF-5
Report Numbers:	05000321/2020010 and 05000366/2020010
Enterprise Identifier:	I-2020-010-0006
Licensee:	Southern Nuclear Operating Co. Inc.
Facility:	Edwin I. Hatch Nuclear Plant
Location:	Baxley, GA 31513
Inspection Dates:	May 18, 2020 to July 17, 2020
Inspectors:	T. Fanelli, Sr. Construction Inspector M. Greenleaf, Reactor Inspector J. Parent, Resident Inspector A. Rosebrook, Senior Reactor Analyst
Approved By:	James B. Baptist, Chief Engineering Br 1 Div of Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a design basis assurance inspection (programs) inspection at Edwin I. Hatch Nuclear Plant, 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. A licensee-identified non-cited violation is documented in report section: 71111.21N.02.

List of Findings and Violations

<table>
<tr><td colspan="4">Failure to Ensure 1D11-F050 Subcomponents were Replaced Prior to the End of the Qualified Life in Accordance with Procedure.</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Barrier Integrity</td><td>Green
NCV 05000321,05000366/2020010-01
Open/Closed</td><td>None (NPP)</td><td>71111.21N.02</td></tr>
<tr><td colspan="4">The inspectors identified a Green finding and associated Non-cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V when the licensee failed to ensure that subcomponents of solenoid-operated valve 1D11-F050 would be replaced or refurbished at the end of their designated life in accordance with procedure NMP-ES-016, Rev. 9.</td></tr>
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<tr><td colspan="4">Violation of Primary Containment Technical Specification 3.6.1.1</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Barrier Integrity</td><td>Green
NCV 05000321,05000366/2020010-02
Open/Closed</td><td>[P.1] - Identification</td><td>71111.21N.02</td></tr>
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<tr><td colspan="4">Failure to Seismically Qualify the Safety Functions of Primary Containment Isolation Butterfly Valves</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Barrier Integrity</td><td>Green
NCV 05000366/2020010-03
Open/Closed</td><td>None (NPP)</td><td>71111.21N.02</td></tr>
<tr><td colspan="4">The Inspectors identified a violation of 10 CFR 50, Appendix B, Criterion III for the licensee's failure ensure that the safety function of the primary containment isolation butterfly valves was seismically qualified in accordance with the licensing basis and Updated Final Safety Analysis Report (UFSAR).</td></tr>
</table>

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000366/2020-001-00	LER 2020-001-00 for Edwin I. Hatch Nuclear Plant, Unit 2, Primary Containment Penetration Exceeded Maximum Allowable Primary Containment Leakage Rate (La)	71111.21N.02	Closed

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. Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), inspectors were directed to begin telework. In addition, regional baseline inspections were evaluated to determine if all or portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In some cases, portions of an IP were completed remotely and on site. The inspections documented below met the objectives and requirements for completion of the IP.

REACTOR SAFETY

71111.21N.02 - Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements

POV Review (IP Section 03) (8 Samples)

The inspectors:

a. Determined whether the sampled POVs are being tested and maintained in accordance with NRC regulations along with the licensee's commitments and/or licensing bases.
Specific Guidance
b. Determined whether the sampled POVs are capable of performing their design-basis functions.
c. Determined whether testing of the sampled POVs is adequate to demonstrate the capability of the POVs to perform their safety functions under design-basis conditions.
d. Evaluated maintenance activities including a walkdown of the sampled POVs (if accessible).

(1) 2T48F319, 2T48F320 Drywell Main Exhaust Inboard and Outboard
(2) 1E41-F001, Steam Supply Shutoff
(3) 1E41-F003, Steam Supply Outboard CIV
(4) 1E11-F006B, RHR Shut Down Cooling Suction
(5) 2E11-F068B, RHR Hx Service Water Discharge
(6) 2E51-F008, Steam Supply Outboard CIV
(7) 1E41-F051, Pump Suction Torus Inboard Isolation
(8) 1D11-F050, Fission Prod Mon CIV, Target Rock

INSPECTION RESULTS

Failure to Ensure 1D11-F050 Subcomponents were Replaced Prior to the End of the Qualified Life in Accordance with Procedure.

Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000321,05000366/2020010-01 Open/Closed	None (NPP)	71111.21N.02

The inspectors identified a Green finding and associated Non-cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V when the licensee failed to ensure that subcomponents of solenoid-operated valve 1D11-F050 would be replaced or refurbished at the end of their designated life in accordance with procedure NMP-ES-016, Rev. 9.

Description: Valve 1D11-F050 is a Target Rock, 1-inch, globe valve that is normally-closed, held-open. The valve is held open by a power signal (energized) until it is de-energized to close as part of the station's containment isolation during postulated design basis accidents.

Valve 1D11-F050 is required to be environmentally qualified in accordance with 10 CFR 50.49 "Environmental qualification of electric equipment important to safety for nuclear power plants." Since the valve normally energized, the licensee needed to account for the increased thermal degradation the valve's components would endure due to self-heating from the power applied to the solenoid, as described in the licensee's calculation SINH-91-001 (Revision 3). In the calculation it was determined that the electrical compartment components (e.g. rectifier, terminal board, and position switches), and non-metallic seals and discs should be replaced every 10 years. The continuously energized solenoid assembly was determined to have a qualified life of 20 years.

The station's current environmental qualification preventative maintenance (EQPM) activities for 1D11-F050 are EQPM-1D11F050-SV-002 (with a 20 year PM frequency) and EQPM-1D11F050-SV-003 (with a 10 year PM frequency). The 20 year PM requires replacement of electrical components (e.g. solenoids and switches) while the 10 year PM only requires replacement of the elastomers (e.g. O-rings and gaskets) internal to the valve.

Preventative maintenance of EQ equipment is required to be completed in accordance with a schedule to be determined by the demonstrated life of the equipment. Due to this fact, equipment should not remain in service beyond their qualified life without the PMs being performed. Unlike other maintenance activities at a nuclear power plant, there is no “grace period” for performing the required maintenance and, in general, equipment installed beyond its qualified life is not expected to perform its safety function when subjected to the deleterious effects of the harsh environment created during the design basis accident.

Inspectors questioned whether the current PMs or any other procedures or actions would replace the electronics with a 10 year qualified life prior to their qualified life expiring (within 4 months of the onsite inspection). The licensee performed a review and determined that there existed no PM or other mechanism that would have ensured the equipment would have been replaced prior to the expiration of their qualified life. The licensee also could not provide a justification for extending the qualified life of the subcomponents via analysis.

Based on the station’s PMs not replacing the electronics in the valve with a periodicity required by the valve’s EQ maintenance requirements, the valve’s internal components would have exceeded their qualified life prior to their replacement. Equipment installed beyond its qualified life is not expected to satisfactorily perform its specified safety function if called upon during a design basis accident. The valve was replaced in October of 2010 and the scheduled PM replacement of the electronic components would not have occurred until October of 2030 (10 years after the expiration of their qualified life). Without the inspectors' identification of the missing EQPM, the valve's qualification would have been no longer valid after October of 2020.

Corrective Actions: The licensee generated condition report (CR) 10712920. In the CR, the licensee identified that the valve had been completely replaced in October of 2010 and at the time of discovery (June of 2020), these components had not yet exceeded their qualified life. The licensee plans to implement a work order to replace these components prior to the expiry of their qualified life in October.

Corrective Action References: CR 10712920

Performance Assessment:

Performance Deficiency: The failure to ensure that the electrical components with a demonstrated life of approximately 10 years would be replaced in accordance with procedures prior to the expiry of their demonstrated life was a performance deficiency. Specifically, section 3.3 of procedure NMP-ES-016, Rev. 9, states that EQ program owners are responsible, in part, for the inhiation and processing of preventative maintenance change requests (PMCRs) as necessary. Contrary to this, the licensee failed to initiate a PMCR to include the necessary maintenance activities to maintain qualification of valve 1D11-F050 past October of 2020.

Screening: The inspectors determined the performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, without discovery of this issue by the NRC, the valve electrical components would have been installed beyond their demonstrated life - challenging the capability and reliability of the valve to perform its safety function when called upon in a harsh environment during a design basis accident.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined that the finding was of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, a failure of containment isolation system (logic and instrumentation), a failure of containment pressure control equipment (including SSCs credited for compliance with Order EA-13-109), or a failure of containment heat removal components; and did not involve an actual reduction in function of hydrogen igniters in reactor containment.

Cross-Cutting Aspect: Not Present Performance. No cross cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.

Enforcement:

Violation:

Criterion V of Appendix B of 10 CFR Part 50 requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings and shall be accomplished in accordance with these instructions, procedures or drawings.

Contrary to the above, since 2014, the licensee failed to ensure that activities affecting quality

for the replacement or refurbishment of components at the end of their qualified life was accomplished in accordance with procedure NMP-ES-016, Rev. 9, which required appropriate schedules to accomplish these tasks. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Violation of Primary Containment Technical Specification 3.6.1.1			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000321,05000366/2020010-02 Open/Closed	[P.1] - Identification	71111.21N.02
The Inspectors identified a violation of the Technical Specification (TS) 3.6.1.1 for the licensee's failure to enter the limiting condition for operations and be in mode 4 in 36 hours.			

Description: On October 22, 2019 the licensee identified primary containment leakage. The licensee failed to determine that the leakage rate exceeded the allowable limits specified for a design basis Loss of Coolant Accident (LOCA) condition. The TS 5.5.12, "Primary Containment Leakage Rate Testing Program" stated, "the maximum allowable primary containment leakage rate, La, at Pa is 1.2% of primary containment air weight per day," where Pa is 43.7psig. The Updated Final Analysis report (UFSAR) Section 6.2.5.6, "Control of Combustible Gas Concentrations in Containment Following a LOCA," stated, in part, "HNP-2-FSAR section 15.3 provides offsite radiation exposure to the public for the design basis LOCA. In calculating the radiation exposures, a containment leakage rate of 1.5% (1.2% by weight) per day, which remains constant over the 30-day period following the accident, is assumed."

The leak continued unabated until January 4, 2020 when the licensee identified that primary containment purge and vent valves 2T48-F319 and 2T48-F320 were leaking to atmosphere. On January 4, 2020, Unit 2 primary containment leakage was determined to exceed allowable containment leakage (La) which resulted in a loss of the containment safety function

The inspectors reviewed licensee information Documentation of Engineering Judgment (DOEJ) -HRSNC1068903-M001, "Evaluation of Unit 2 Drywell Leakage." The inspectors were unable to verify the evaluation results. Errors in the evaluation could have resulted in an increased leakage rate. Due to the leakage rate, the licensee was unable to use Maintenance and Test Equipment to quantify the leakage rate. The licensee could not provide records of any examinations of the valve internals after removal that could be used to quantify the leakage rate. However, interviews with licensee personnel indicated that visual inspections could observe that the disc and seat were not seated against each other. The licensee provided further information in a third-party calculation documented in SNC1109456, "Leakage from Isolation Valves 2T48-F319 and 2T48-F320." The calculation indicated that the leakage rate could have been approximately 89% of containment volume per day.

Corrective Actions: The licensee replaced one valve and repaired the second valve. The licensee submitted LER 2020-001-00 and performed a root cause analyses to determine the technical and human performance deficiencies

Corrective Action References: CR 10657734, CR 10675186, CR 10675289, LER 2020-001-00, CAR 277277
<u>Performance Assessment</u>: Performance Deficiency: The licensee's failure to promptly identify and correct conditions adverse to quality and evaluate those conditions for operability. Operators failed to identify and later correct that containment leakage was in excess of technical specification limits from October 22, 2019, until January 4, 2020, which was a performance deficiency. This resulted in primary containment being inoperable for greater than its allowed outage time. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Human Performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to recognize leakage that exceeded TS limits degraded one of the plant's principle safety barriers which resulted in a loss of safety function, for more than three months, that is needed to control the large early release frequency (LERF) of radioactive material. Significance: The inspectors assessed the significance of the finding using Appendix H, "Containment Integrity SDP." The inspectors evaluated this performance deficiency using IMC 0612 Appendix B, "Additional Issue Screening Guidance." This performance deficiency was determined to be more than minor because the issue affects the Configuration Control Attribute of the Barrier Integrity Cornerstone and adversely impacts the associated cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Per IMC 0609 Attachment 4, this issue affects the Barrier Integrity cornerstone and is routed to IMC 0609 Appendix A The Significance Determination Process (SDP) for Findings At-Power. IMC 0609 Appendix A Exhibit 3 Barrier Integrity Screening Questions Section C Reactor Containment states, "Does the finding represent an actual open pathway in the physical integrity of reactor containment (valves, airlocks, etc), failure of containment isolation system (logic and instrumentation), failure of containment pressure control equipment (including SSCs credited for compliance with Order EA-13-109), failure of containment heat removal components, or failure of the plant's severe accident mitigation features (AP1000)? If YES screen to IMC 0609 Appendix H, Containment Integrity Significance Determination Process. A regional senior reactor analyst conducted a risk assessment using IMC 0609, Appendix H. The issue is considered a Type B finding which only affects LERF and has no impact on CDF. Hatch is a BWR with a Mark I containment and this issue affects Vent and Purge System Valves. Using table 7.1, Phase 1 Screening-Type B Findings at Power, this issue screen to Phase 2. Using table 7.2, Phase 2 Risk Significance -Type B Findings at Power since this is a BWR Mark 1 containment leakage from drywell to environment is considered significant if leakage was >100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems. Initially the licensee's calculations did not support their conclusions about containment leakage. The licensee contracted a third party who calculated containment leakage to be approximately 88.7% of containment volume. The NRC concluded that there was not enough data to challenge the conclusions of the third party calculation. Therefore, in accordance with Table 7.2, this issue is characterized as very low safety significance (Green).

Cross-Cutting Aspect: P.1 - Identification: The organization implements a corrective action program with a low threshold for identifying issues. Individuals identify issues completely, accurately, and in a timely manner in accordance with the program. The licensee had indication of excessive leakage on October 22, 2019 but failed to perform troubleshooting/investigative actions and a proper leak rate calculation until January 4, 2020.
Enforcement: Violation: Technical Specification 3.6.1.1, “Primary Containment,” required, that the Primary Containment Shall be Operable in Modes 1, 2, and 3 or be in mode 4 in 36 hours if the Required Action to Restore Containment Operability in a 1 hour Completion Time is not met. Contrary to the above, the Primary Containment Operable leakage allowable limits of 1.2% (by Weight) La was exceeded for longer than Technical Specification 3.6.1.1 allowed, from October 22, 2019 until the leak was isolated on January 3, 2020, without entering mode 4. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Seismically Qualify the Safety Functions of Primary Containment Isolation Butterfly Valves			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000366/2020010-03 Open/Closed	None (NPP)	71111.21N.02
The Inspectors identified a violation of 10 CFR 50, Appendix B, Criterion III for the licensee’s failure ensure that the safety function of the primary containment isolation butterfly valves was seismically qualified in accordance with the licensing basis and Updated Final Safety Analysis Report (UFSAR).			
Description: The licensing basis and UFSAR Chapter 3, “Design of Structures, Components, Equipment, and Systems,” specified the seismic requirements for classifying and qualifying seismic Category 1 components. • Title 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Section VI. "Application to Engineering Design," required, in apart, that " in addition to seismic loads, including aftershocks, applicable concurrent functional and accident-induced loads shall be taken into account in the design of these safety-related structures, systems, and components.” …”The engineering method used to insure that the required safety functions are maintained during and after the vibratory ground motion associated with the Safe Shutdown Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.” • The UFSAR Section 3.2.1, “Seismic Classification,” specified that Seismic Category 1 components are those that must function... for activity confinement following a loss-of-coolant accident to ensure that the public is protected in accordance with 10 CFR 100 guidelines. "			

- The UFSAR Sections 3.7A, “Seismic Design,” 3.9 “Mechanical Systems and Components,” and 3.10, “Seismic Design of Seismic Category 1 Instrumentation and Electrical Equipment,” specified that the valves must be seismically qualified by dynamic seismic analysis and testing in accordance with IEEE 344-1975, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. The inspectors noted that the IEEE standard specified, in part, that an analysis should demonstrate an equipment's ability to perform its required function including component alignment (if critical to proper operation).

The qualification of record for 18” Primary Containment Isolation Butterfly Valves (PCIVs) in Unit 2 was a generic seismic analysis SX18951, “Seismic Analysis on 6", 16", 18", 20", & 04" Butterfly Valve Assemblies for Georgia Power Company.” The unit 2 construction permit was issued on December 27, 1972. The evaluation was prepared by Fisher Controls company June 14, 1971. The evaluation was resubmitted to the licensee on Mach 28,1973. The analysis stated, the internals were “Non-Critical Areas,” because previous calculations have shown that the stresses due to seismic loading on the internal parts of the valve (disc, shaft, etc.) are insignificant. Therefore, these specific parts are not analyzed in this report.” The analysis was a static analysis that did not meet the requirements stated above to provide adequate conservatism that the valves would perform their required function and did not evaluate the alignment of the disc and seat that is critical to proper operation. The inspectors noted that the purchase order (PO) for these valves (PEH-2-145) and attached valve specifications SS-2102-107, “Furnishing, Fabricating and Delivering Butterfly Valves for Edwin I. Hatch Nuclear Plant - Unit No. 2 of Georgia Power Company,” was dated March 7, 1972. The PO did not specify the Hatch seismic licensing basis or UFSAR specifications stated above. The inspectors determined that the resulting seismic analysis did not meet the Hatch licensing basis, UFSAR specifications, nor reasonable assurance that the valves would perform their function to seal containment after seismic and other concurrent loadings.

Corrective Actions: The licensee entered this into the corrective action program and performed an immediate determination of operability and determined the valves were functional.

Corrective Action References: CR10731466

Performance Assessment:

Performance Deficiency: The failure to meet the seismic requirements for butterfly valves in seismic analysis SX18951, as specified by the Unit 2 licensing basis and UFSAR, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the SSC and Barrier Performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors determined the performance deficiency was more than minor because it was associated with the Barrier Performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to seismically qualify the safety function of containment isolation valves in accordance with the UFSAR failed to

provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Significance: The inspectors assessed the significance of the finding using Appendix H, “Containment Integrity SDP.” The inspectors reviewed the finding using IMC 0609, Attachment 4, “Initial Characterization of Findings,” Barrier Integrity Cornerstone, which directed the review to IMC 0609 Appendix H. Using Appendix H, “Containment Integrity Significance Determination Process,” Section 07.01, Approach for Assessing Type B Findings at Power, and Table 7.2, the inspectors determined that the finding was of very low safety significance (Green). Cross-Cutting Aspect: Not Present Performance. No cross cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.
Enforcement: Violation: Title 10 CFR 50, Appendix B, Criterion III, required, in part, "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 Mach 28,1973, the Hatch Unit 2 design control measures failed to 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. Specifically, the licensee failed to verify that the Butterfly Valves purchased under Purchase Order PEH-2-145 and evaluated by analysis SX18951 met the Unit 2 seismic qualification licensing basis and UFSAR specifications. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Licensee-Identified Non-Cited Violation	71111.21N.02
This violation of very low safety significance was identified by the licensee and has been entered into the licensee corrective action program and is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.	
Violation: The licensee identified that valves which are limit switch and torque switch controlled the site has been determining the margin based on limit switch control. The site correctly recognized a concern with leak tightness with this method of control. This as an issue for valves with LLRT requirements as identified by the Corporate Project Manager and documented in Condition Report 10705537. Technical Evaluation 1066456 in the Hatch CAP process. Corporate procedure NMP-ES-017-002 Section 4.9 MOV Control Logic, stated, in part “the review SHALL verify that the control scheme is appropriate for the MOV and its required function. Section 4.10 Close Direction Logic – Rising Stem Valves, stated in part, “Torque Switch Bypass …For certain valves, the torque switch can be bypassed until after flow isolation to provide maximum margin capability. Adherence to this procedure could have prevented the violation. No operability concerns were identified. Any actions needed, such as changing MOV test	

frequencies based on Torque Switch trip margin, will be determined under TE 1066456 due July 4, 2020.

10 CFR Part 50, Appendix B, Criterion V requires, in part that activities affecting quality be accomplished in accordance with procedures.

Contrary to the above, the station failed to follow corporate procedure NMP-ES-17-002 when determining MOV logic control.

Significance/Severity: Green.

EXIT MEETINGS AND DEBRIEFS

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

- On July 16, 2020, the inspectors presented the design basis assurance inspection (programs) inspection results to Cheryl A. Gayheart and other members of the licensee staff.
- On December 8, 2020, the inspectors presented the Final Exit inspection results to Sonny Dean and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.21N.02	Calculations	CD 71-59	Seismic Analysis - 6", 16", 18", 20" & 24" Butterfly Valve Assemblies for Georgia Power Company.	12/23/1971
		CD 72-292	Seismic Analysis of 6", 16", 18", 20" & 24" Butterfly Valve Assemblies for Bechtel Corporation Agent for Georgia Power Company.	03/28/1973
		S-52630	Motor Operated Valves - 20" 300lb 5206WE Gate Valve - Thrust & Torque Calculations - MPL 1E11F006A,B,C,D & 1E11F010	03/20/1991
		SENH-11-002	Unit 2 Station Auxiliary System Study	Rev. 3
		SENH-11-003	Diesel 2A and 2C Dynamic Diesel Generator Loading Analysis	Rev. 2
		SENH-13-007	Unit 1 Station Auxiliary System Study - Final	Rev 5
		SENH-15-004	Unit 1 Degraded Grid Timing Analysis	Rev. 1
		SENH-93-011	Diesel Generator 1A, 1B, 1C Dynamic Loading Analysis	Rev. 2
		SINH 91-001	Qualified Life of Electronics for TRC SOVs	Rev. 3
		SMNH 89-051	Determine Qualified Lives for EQ Equipment In The Drywell, Steam Chase & Personnel Access	Rev. 13
		SMNH-04-004	Motor Operated Valve Torque Switch Setting Guide	Rev. 17
		SMNH-11-006	Jog Motor Operated Valve Classifications for GL 96-05 Gate Valves	Rev. 1
		SMNH-11-073	HPCI Reference Summary	Rev. 2
		SMNH-12-021	Unit 1 RHR Reference Summary for Design Basis Retrieval	Rev. 2
		SMNH-12-022	Unit 2 RHR Reference Summary for Design Basis Retrieval	Rev. 2
		SMNH-89-051	Determine Qualified Lives for EQ Equipment In The Drywell, Steam Chase & Personnel Access	Rev. 13
		SMNH-89-052	Determine Qualified Lives For EQ Equipment in Drywell, Steam Chase & Personnel Access	Rev. 10
		SMNH-91-018	RHRSW Flow Controller Valve	Rev. 1
		SMNH-93-004	Motor Operated Valve Differential Pressure Calculations	Rev. 6
		SMNH-99-013	Pressure Locking and Thermal Binding for Gate Valves	Rev. 6

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	Corrective Action Documents	10031840		
		10035026		
		10133018		
		10138053		
		10150041		
		10185014		
		10186473		
		10462195		
		10471961		
		10547227		
		10547659		
		10555769		
		10657734		
		10675186		
		10675289		
		10694714		
		10695410		
		10696032		
		1087258		
		206008		
		269167		
		277167		
		277193		
		277277		
		482310		
		TE 1051813		
		TE 1054901		
		TE 1057577		
		TE 1057579		
		TE 1063153		
		TE 465659		
		TE 525257		
	Corrective Action	10711923	MIDAS MOV Program Update Needed	05/28/2020

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	Documents Resulting from Inspection	10711978	2020 NRC POV Inspection	05/28/2020
		10712473	EQ Target Rock Documentation Issue - 1-QDP10-EQ	06/01/2020
		10712920	Missed EQPM Component Replacements	06/02/2020
	Drawings	A-52123	Protective Relay Set Point Data Sheet - Sheet 41	Rev. 4
		CD05547	Pressure Seal Gate Valve Assembly	B
		H-16173	Fission Products Monitoring System P&ID Sheet No. 1	Rev. 14
		H-16329	R.H.R System P&ID Sht. 1	Rev. 84
		H-17011	Single Line Diagram - Reactor Building 600 V. Essential MCC "1B" Sh. 1 MPL R24-S012	Rev. 43
		H-17205	Residual Heat Removal System E11 Local Racks H21-P018, H21-P021 & Misc. External Connection Diagram	Rev. 46
		H-17227	600V Reactor Building Essential MCC 1B-R24-S012 External Connection Diagram Sheet 2 of 4	Rev. 30
		H-17774	Residual Heat Removal System E11 Elementary Diagram Sheet 15 of 25	Rev. 25
		H-19573	Remote Shutdown System (C82) Elementary Diagram Sheet 2 of 9	Rev. 20
		H-19611	Remote Shutdown System (C82) Elementary Diagram Sheet 4 of 8	Rev. 19
		H-19613	Remote Shutdown System (C82) Elementary Diagram Sheet 6	Rev. 14
		H-21039	R.H.R. Service Water System P. & I.D.	Rev. 48
		H-26084	2T48F319 and 2T48F320 Simplified Diagram Primary Containment Ventilation & Purge System	Rev. 41.0
		H-27013	Single Line Diagram - Reactor Building 600/280V AC Essential MCC 2B Sheet 1 MPL 2R24-S012	Rev. 47
		H-27596	Edwin I. Hatch Nuclear Plant Unit No.2 Primary Containment Isolation Sys. 2C61 Elementary Diagram Sheet 1	26.0
		H-27643	Residual Heat Removal Sys. 2E11 Elementary Diagram Sheet 9	Rev. 20
		H-27650	Residual Heat Removal System 2E11 Elementary Diagram Sheet 16	Rev. 37
		H27989	Edwin I. Hatch Nuclear Plant Unit NO.2 Nuclear Steam	Rev. 16

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			Supply Shut Off Sys. 2A71 Elementary Diagram SHT. 16 of 20.	
		S-52534	Residual Heat Removal Service Water System Flow Control Valves - Drag Valve - Outline	Rev. 3
		S-59880	Project Control Dwg 1" Solenoid Operated Globe Valve Energize to Open SW Ends	Rev. 1
	Engineering Changes	SNC99451	Hatch Unit 1 JOG MOV Modification	Rev. 7.0
	Engineering Evaluations	DOEJ-HRSNC1068903-M001	Evaluation of Unit 2 Drywell Leakage	01/06/2020
		S-55910	Evaluation of Solenoid Valve Part Interchangeability And Environmental Qualification Report No. 6691	Rev. 3
		TERI-006	Technical Evaluation of Replacement Item Bridge Rectifier	05/09/1991
	Miscellaneous		1E41F003 Test Traces	05/03/2019
			2B21-F022C AF-AL Traces	02/11/2019
		1-QDP01	Limitorque Valve Motor Operators	Rev. 1
		1-QDP10	Target Rock Solenoid Valves Models 73K and 75F	Rev. 1
		2-QDP01	Limitorque Valve Motor Operators	Rev. 1
		IEEE 344-1971	IEEE Trail-Use Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations	09/16/1971
		IEEE 344-1975	IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.	12/20/1974
		LER 2020-001-00	Primary Containment Penetration Exceeded Maximum Allowable Primary Containment Leakage Rate (La).	03/02/2020
		NMP-ES-014-H-AOV-2B21F022C	Flow Scanner Datasheet Preparation for Testing, Datasheet 1, Significant Parameters	Rev. 3.0
		S-31443	Limitorque Corp. Valve Operator SMB0/H3BC	06/17/1977
		S-52722	Residual Heat Removal Service Water System Operation & Maintenance Instruction Manual	Rev. 3
		S-52853	RHRSW System - Flow Control Valves Valve Design	Rev. 2

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			Report	
		S-52854	RHRSW System Flow Control Valves Seismic Analysis Report	Rev. 2
		S-53012	Test Report - Qualification Testing of Namco Limit Switches & Seals - Six Year	02/20/1992
		S-63313	RHRSW HX Control Valves 2E11F068A/B Data Sheet and CV Curve	Rev. 1
		S-70585	Qualification Test Report - Aging, Seismic, & Accident Simulation Test 1" Solenoid Valve Model 76HH-002	10/28/1985
		S-71042	Envir. Qual. Similarity Analysis Report Vlv Model 73K-002-1, 73K-003-1, 73K-004-1, 75F-005-1, 75F-008-1 & 75F-010-1 Solenoid Operated Globe Vlv	Rev. 1
		S-71068	Report No. 5194B, Project Bo. 89AA EQ Replacement Parts List for Solenoid Operated Globe Valves Models 73K and 75F	Rev. B
		S53012	Test Report - Qualification Testing of Namco Limit Switches & Seals - Six Year	02/20/1992
		S70468	Limitorque Valve Actuator Qualification for Nuclear Power Station Service - Report B0058 - Test Conducted Per IEEE 382-1972, 323-1974, 344-1975	1/11/80
		SINH-93-002	Patel Conduit Seal Test	02/09/1993
		SS-6902-173	Nuclear Solenoid Valves	Rev. 2
	Procedures	34-SV-E11-002-1	RHR "B" Loop Valve Operability	Rev. 21.6
		34-SV-SUV-027-2	Reactor Building Isolation Logic System FT	Rev. 1.3
		34AB-T22-001-2	Primary Coolant System Pipe Break in Reactor Building	Rev. 0.7
		34SO-E11-010-1	Residual Heat Removal System	Rev. 45
		34SV-R43-004-1	Diesel Generator 1A Semi-Annual Test	Rev. 16.1
		34SV-R43-004-1	Diesel Generator 1A Semi-Annual Test	Rev. 16
		34SV-R43-020-1	Diesel Generator 1A LOCA/LOSP LSFT	Rev. 2.3
		34SV-SUV-008-1	Primary Containment Isolation Valve Operability	Rev. 15.24
		42SV-TET-003-2	Primary Containment Integrated Leak Rate Test	7.7
		A43830	Motor Operated Valve Torque Switch Setting Guide	Rev. 31
		NMP-AD-025-F02	Hatch Nuclear Plant MOV Regulatory Scope	Rev. 2
		NMP-ES-016	Environmental Qualification Program	Rev. 8

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		NMP-ES-017-001-H	Hatch Nuclear Plant MOV Regulatory Scope	Rev. 2
		NMP-ES-017-006	Motor-Operated Valve Design Database Control and Design Data Sheet Activities	Rev. 1
		NMP-GM-002	Corrective Action Program	Rev. 15.2
		NMP-GM-002-001	Corrective Action Program Instructions	Rev. 39.0
		NMP-GM-008	Operating Experience Program	Rev. 22.0
		NMP-GM-008-006	Leveraging Internal Operating Experience	Rev. 5.1
		NMP-GM-008-GL01	Guideline for Searching for Relevant OE	Rev. 5.0
	Work Orders	SNC1056280		
		SNC340178		
		SNC399169		
		SNC641858		
		SNC641858		
		SNC806506		
		SNC974077		0
		SNC990381		