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

December 28, 2020

Mr. Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NO. 1 - RE: REQUEST FOR RELIEF REQUEST N1-I5-NDE-002 INSERVICE INSPECTION ALTERNATIVE (EPID L-2020-LLR-0075)

Dear Mr. Stoddard:

By letter dated May 18, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20140A239), Virginia Electric and Power Company (VEPCO, Dominion Energy Virginia, the licensee), submitted Relief Request No. N1-15-NDE-002 for the North Anna Power Station, Unit 1 (NAPS1), to the U.S. Nuclear Regulatory Commission (NRC or Commission) for review and approval, pursuant to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(1).

Specifically, the proposed alternative is related to the successive inspection requirement of paragraph IWB-2420(b) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, “Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components,” for a “relevant condition.” The proposed alternative is related to the frequency requirements in Section XI of the ASME Code for inspecting the degraded cladding, which is only made accessible by the removal of the core barrel (normally removed once per ten-year inspection interval).

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that VEPCO has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) and demonstrated that the proposed alternative will provide a level of quality or safety. Therefore, the NRC staff authorizes the proposed alternative for North Anna, Unit 1, until the end of the Fifth Inservice Testing (IST) Program Test interval, which ends on April 30, 2029.

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

If you have any questions, please contact the Project Manager, Ed Miller at 301-415-2481 or via e-mail at Ed.Miller@nrc.gov.

Sincerely,

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2020.12.28 13:17:23 -05'00'

Michael T. Markley, Chief
Plant Licensing Branch 2-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338

Enclosure:
Safety Evaluation

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED INSERVICE INSPECTION ALTERNATIVE N1-15-NDE-002

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION ENERGY VIRGINIA)

NORTH ANNA POWER STATION, UNIT 1

DOCKET NO. 50-338

## 1.0 INTRODUCTION

By letter dated May 18, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20140A239), Virginia Electric and Power Company (Dominion Energy Virginia, the licensee), submitted Relief Request No. N1-15-NDE-002 for the North Anna Power Station, Unit 1 (NAPS1), to the U.S. Nuclear Regulatory Commission (NRC or Commission) for review and approval, pursuant to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(z)(1). Attachment 3 to this submittal, calculation C-4523-00-02-P, Revision 0, is withheld from public disclosure for being proprietary to Dominion Engineering Inc. An NRC staff determination to withhold this information was issued on July 16, 2020 (ADAMS Accession No. ML20178A321). A non-proprietary version is included in the submittal as attachment 4.

The licensee requested that the NRC authorize its proposed alternative to the successive inspection requirement of paragraph IWB-2420(b) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, “Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components,” for a “relevant condition.” The licensee’s proposed alternative is related to the frequency requirements in Section XI of the ASME Code for inspecting the degraded cladding, which is only made accessible by the removal of the core barrel (normally removed once per ten-year inspection interval).

The proposed alternative is applicable for the Fifth 10-Year Inspection Interval of the ISI Program for NAPS1, which is scheduled to end on April 30, 2029. In accordance with 10 CFR 50.55a(z)(1), the licensee submitted its proposed alternative based on its determination that the proposed alternative would provide an acceptable level of quality and safety.

## 2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ISI of ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system

pressure tests conducted during the First 10-Year ISI Interval and subsequent intervals comply with the latest edition and Addenda of Section XI of the ASME Code that was incorporated by reference in 10 CFR 50.55a(a)(1)(ii), 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

Regulation 10 CFR 50.55a(z) states that alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the NRC. A licensee's proposed alternative must be submitted to and authorized by the NRC prior to implementation. The licensee must demonstrate that:

(1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety; or

(2) *Hardship without* a *compensating increase in quality and safety.* Compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above regulatory requirements, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to propose and the Commission to authorize this alternative to the requirements of the ASME Code, Section XI.

## 3.0 TECHNICAL EVALUATION

### 3.1 Licensee's Proposed Alternative

Applicable Code Edition and Addenda

The applicable code for the NAPS1 Fifth 10-Year ISI Interval and ISI Program is the ASME Boiler and Pressure Vessel Code (BPV) Section XI, 2013 Edition with no Addenda. The NAPS1 Fifth 10-Year ISI interval started May 1, 2019 and ends April 30, 2029.

American Society of Mechanical Engineers (ASME) Code Components Affected

The affected component is the North Anna Power Station Unit 1 (NAPS1) reactor pressure vessel (RPV) interior surface:

- Examination Category - B-N-1, “Interior of Reactor Vessel”
- Examination Item Number - B13.10, “Vessel Interior”

ASME Code Requirement for Which Alternative Is Requested

IWB-2411, Inspection Program, requires visual examination of accessible areas of the interior of the RPV made accessible for examination by removal of components during normal refueling outages as identified in Table IWB-2500-1. These examinations are to be performed in each inspection period.

IWB-2420(b), states, in part, “If a component is accepted for continued service in accordance with IWB-3132.3 or IWB-3142.4, the areas containing flaws or relevant conditions shall be re-examined during the next three inspection periods listed in the schedule of the Inspection Program of IWB-2400.”

Licensee's Proposed Alternative to the ASME Code

The licensee proposes to re-examine the areas with cladding damage found during the NAPS1 Spring 2018 refueling outage (N1 R26) no later than the next scheduled core barrel removal at the end of the Fifth ISI Inspection Interval, currently scheduled to occur in the fall 2028. Specifically, the ASME XI, IWB-2420(b), requirement for “Successive Examinations,” in the First Period and Second Period will be deferred. Pursuant to 10 CFR 50.55a(z)(1), the licensee determined that the proposed alternative will provide an acceptable level of quality and safety.

Description and Characterization of Indications Included in Proposed Alternative

The licensee provided the details of the three indications:

- Indication 1 - This indication was identified as an area of interest during a review of the 2009 B-N-1 RPV inspection video. This indication was located below the 55° irradiation specimen slot, and visual examination revealed an area of cladding degradation which exposed the underlying low-alloy steel. No indications of cracking were visible in the area surrounding this indication.

- Indication 2 - During the RPV interior examination, this indication was identified adjacent to the 270° Radial (Clevis) Support Keyway. Further investigation, which included an EVT-1 examination, revealed an area of cladding degradation which exposed the underlying low-alloy carbon steel. No indication of cracking was visible in the area surrounding this indication.

- Indication 3 - During the examination of the Lower Internals Support Ledge, there were areas identified with materials deformation at the base of each of the following Irradiation Specimen Slots: 45°, 55°, and 65°. This was a total of three areas of deformation: one area per Specimen Slot. No indication of cracking was visible in the area surrounding each indication.

The licensee determined that indications 1 and 2 are most likely fabrication flaws, and that the sizes and shapes of these indications suggest that they were a consequence of poorly adhered cladding material which was produced by a lack of fusion of the deposited clad. Further investigation, which included VT-1 examination, revealed an area of cladding degradation which exposed the underlying low-alloy carbon steel. No indications of cracking were visible in the areas surrounding these indications. The licensee stated that indication 3 is deformation of the cladding material most likely caused by the removal of the core barrel for the 10-year ISI B-N-1 vessel examination. The licensee suspects that there may have been a slight misalignment of the core barrel such that when the core barrel was extracted, the specimen features on the thermal shield contacted the specimen slots. No indication of cracking was visible in the area surrounding each indication.

Reason for Proposed Alternative

The licensee stated that IWB-2420(b), requires a VT-3 visual examination of the areas containing flaws identified during the fourth ISI interval, Third Period, B-N-1 examination, during the next three inspection periods. The licensee explained that an alternative is requested for the B-N-1 examination of degraded cladding in areas only made accessible by the removal of the core barrel, an evolution normally taking place once per 10-year inspection interval. Removal of the core barrel presents extensive radiological and industrial safety concerns, including:

- a significant heavy lift of a critical vessel internals component
- construction of a shield wall for the portion of the core barrel above the refueling pool water level when it is placed in the stand
- significantly higher dose rates for station personnel during the refueling outage
- the potential for further cladding damage when the core barrel is moved

Licensee's Basis for Proposed Alternative

The licensee provided its technical basis for the proposed alternative to the re-examination requirements of IWB-2420(b). The technical basis focused on areas of operating experience, potential degradation mechanisms, and structural analyses of the indications.

Operating Experience

The licensee stated that an external operating experience search was conducted for the potential for stress corrosion cracking (SCC) and inservice induced fatigue crack growth degradation mechanisms to affect pressurized-water reactor (PWR) RPV low-alloy steel material in the area of cladding defects. The results of this review and industry assessment concluded that these degradation mechanisms are not a significant concern for the case of cracked, damaged, missing, or removed cladding of the RPV.

Degradation Mechanisms

*Corrosion Evaluation*

The licensee stated that the maximum conservative estimate of corrosion loss of 0.31 inches (40 years) and 0.62 inches (80 years) have been shown to be structurally acceptable. The RPV has a structural margin for the probable depth of the indications and the extreme, worst-case corrosion even after an 80-year life. Plant operating experience supports the above conclusions that the subject cladding defects are acceptable for long-term future operation. Furthermore, several cases at other PWRs with areas of damaged, missing, or intentionally removed RPV cladding are described in WCAP-15338-A, and in all cases, the cladding defects were reported to be inconsequential.

*Stress Corrosion Cracking (SCC) Evaluation*

The licensee stated that the typical PWR service and shutdown Reactor Coolant System (RCS) chemistry contains oxygen and chloride levels that are significantly below the threshold levels required to initiate either intergranular SCC (IGSCC) or transgranular SCC (TGSCC) in the cladding, even if the base metal were exposed. Furthermore, the degree of corrosive attack and wastage due to operation is insignificant, as evidenced by operational histories and analyses based on corrosion tests. Therefore, the licensee determined that SCC is not a feasible degradation mechanism even if the base metal were exposed.

Structural Analysis

Indications 1 and 2 are evaluated in the following structural analysis. Indication 3 is not structurally analyzed, as no indication of impact to the carbon steel vessel is apparent. The governing stresses experienced at these locations are less than the stresses at indications 1 and 2.

*Indication #1*

The licensee stated that the primary local membrane plus bending stress intensity for the intact RPV transition region was calculated to be 18.3 ksi, a factor of 2.2 below the 1.5 Sm limit of 40.05 ksi, which is greater than the conservative estimate of 1.23 for an increase in local stress due to corrosion. Furthermore, the fatigue usage factor for the intact RPV transition region, reported is to be 0.0083, which is well below the allowable value of 1.0.

*Indication #2*

The licensee stated that the primary local membrane plus bending stress intensity is a factor of 1.58 below the allowable value, which is greater than the estimate of 1.25 for the increase in local stress due to corrosion conservatively based on the wall thickness of the bottom head. The license also stated that the primary plus secondary stress intensity range is a factor of 1.96 below the allowable value, which is greater than the estimate of 1.25 for the increase in local stress due to corrosion conservatively based on the wall thickness of the bottom head.

Furthermore, the thickness of the undisturbed region of the bottom head is more than 1.0-inch greater than the required 3.868-inch thickness calculated on the basis of the primary membrane stress due to the design pressure. The licensee explained that it applied a 1.0-inch flaw to the fatigue analysis with a fatigue reduction factor of 4.0 to model the flaw as if it is the root of a thread of a bolt. Applying this factor increases the fatigue usage factor to 0.436, which is still below the allowable limit of 1.0 and therefore acceptable.

Duration of Proposed Alternative

The proposed alternative will be used for the Fifth 10-Year Inspection Interval of the ISI Program for NAPS1, which is scheduled to end on April 30, 2029.

## 3.2 NRC Staff Evaluation

Degradation Mechanisms

*Stress Corrosion Cracking Evaluation*

Based on a review of PWR operating experience with RPV cladding indications shows SCC does not occur of pressure vessel steels (e.g., low-alloy steels) under normal RCS conditions (i.e., oxygen-deficient PWR primary coolant environment during at-power conditions, which is maintained by the plant's water chemistry program). Furthermore, the NRC staff noted that WCAP-15338-A (ADAMS Accession No. ML083530289) addressed, in part, cracking and separation of a portion of the clad weld metal resulting in the exposure of the base metal to the primary water. The NRC staff noted that the mechanisms that are applicable include intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC). IGSCC of the clad metal can occur if the weld is sensitized (chromium depleted grain boundaries) and is exposed to oxygenated water. In addition, TGSCC can occur in the cladding only in the presence of a chloride environment. The NRC staff noted that the normal RCS conditions and shut down RCS chemistry at PWRs contains oxygen and chloride levels that are significantly below the threshold levels required to initiate either IGSCC or TGSCC. Thus, the NRC staff finds it reasonable that SCC is not a concern to the structural integrity of the RPV and is not an active degradation mechanism even if the base metal were exposed.

*Localized and Flow-Accelerated Corrosion*

The licensee explained that aggressive localized environments can potentially form in areas of cracked, damaged, missing, or removed cladding and could lead to localized forms of corrosion, such as crevice corrosion, pitting, and galvanic corrosion. The NRC staff noted that PWR primary coolant during power operation is deaerated to the extent where the electrochemical potential is governed by the hydrogen concentration; thus, the potential gradient to cause corrosion is not present. As such, the NRC staff finds that the time and temperature at oxygenated conditions are limited in PWRs, limiting the relevance of aerated conditions and the potential for localized forms of corrosion in areas of cracked, damaged, missing, or removed cladding. Furthermore, the NRC staff noted that a review of PWR operating experience with RPV cladding indications shows that localized forms of corrosion has not been an issue for small regions of exposed low-alloy steel due to the nearly oxygen-deficient PWR primary coolant environment during at-power conditions and the absence of aggressive anion species.

The licensee explained that flow-accelerated corrosion (FAC) of carbon steels is dependent on factors including temperature, material composition (chromium, copper, and molybdenum contents), fluid velocity, turbulence, steam quality, and water chemistry (including dissolved oxygen and pH). The licensee stated that similar to other PWRs, the NAPS Unit 1 reactor inlet temperature is in the range from 285 to 293°C (545 to 560°F). The NRC staff noted that although FAC depends, in part, on temperature, the upper bound for applicability of this degradation mechanism is 280°C (536°F). The licensee explained that the NAPS Unit 1 reactor vessel was fabricated using a SA-508 Class 2 low-alloy steel forging, which has a nominal chromium content of 0.33 wt% and a molybdenum content of about 0.6 wt%. Furthermore, the licensee stated the shell-to-head transition region of the NAPS Unit 1 reactor vessel comprises SA-508 Class 2 low-alloy steel forging material welded to SA-533 Grade B Class 1 plate material, which has a molybdenum content of about 0.5 wt%. The NRC staff noted that studies and test data indicate that these alloying contents in the fabrication materials for the region of the RPV with the cladding indications provide a resistance to FAC and aid in reducing the rate

of flow acceleration corrosion. Thus, based on the temperature of the reactor coolant being above the upper bound for applicability of FAC, and the alloying contents in fabrication materials of the RPV, the NRC staff finds it reasonable that FAC is not expected to be a relevant degradation mechanism, specifically for the intervening time being request as part of the licensee's proposed alternative for either subject cladding defects.

*Corrosion Evaluation*

The licensee explained that that the stainless cladding protects the underlying low-alloy steel from corrosion; however, in areas of cladding loss, corrosion of the low-alloy steel may occur. Thus, the licensee performed an evaluation to determine the amount of corrosion that can occur to the exposed RPV steel as a result of the cladding defects.

The NRC staff reviewed the corrosion evaluation in the licensee's submittal (i.e., Dominion Engineering calculation C-4523-00-01) and noted that the highest corrosion rate reported for all immersion tests (i.e., general corrosion) applicable to each chemical environment were assumed (i.e., refueling water, cold shutdown, startup/shutdown, and normal operation). The NRC staff noted that conditions known to lead to substantial amounts of low-alloy steel corrosion at cladding defects (i.e., ongoing air in leakage during normal operation and highly concentrated boric acid associated with pressure boundary leakage or evaporation of a limited volume of liquid) were not accounted for in the corrosion evaluation because they are not credible environments for the reactor vessel interior. The NRC staff reviewed the corrosion rates used in the corrosion evaluation in the licensee's submittal and finds them reasonable and representative for the plant operating conditions being assessed because they were based on a wide range of laboratory experiments that considered key factors, such as boron concentration, lithium concentration, pH value and oxygen content of solution, and temperature, that was performed to characterize boric acid corrosion rates in various environments.

The licensee assumed that plant operating conditions were constant during each of the explicitly considered operating periods (i.e., refueling water, cold shutdown, startup/shutdown, and normal operating conditions) and were conservatively developed based on the procedures from the water chemistry program at NAPS Unit 1. Furthermore, the licensee's evaluation assumed the corrosion rate affecting the reactor vessel was constant, and as the highest reported rate for each operating condition. The NRC staff finds these assumptions conservative because water chemistry conditions fluctuate within established thresholds that impact the corrosion rates, rather than remaining constant at the upper bound for chemistry conditions and corrosion rates. Furthermore, the NRC staff noted the duration of cold shutdown conditions and startup/shutdown conditions were conservatively assumed to last for seven days, when typically, these operating conditions only last for one to three days.

The licensee's corrosion evaluation assessed several scenarios to assess the potential defect depth over the next 10 years, 20 years (i.e., end of 60-year renewed license), and 40 years (i.e., end of an 80-year renewed license if licensee decides to pursue). In addition, the licensee assessed the potential defect depth for the entire service time of 80-years, assuming the licensee decides to pursue a subsequent renewed license. The NRC staff noted that assessment of the potential defect depth for the entire service time of 80-years is appropriate because it is believed the cladding defects are most likely fabrication flaws that they were a consequence of poorly adhered cladding material which was produced by a lack of fusion of the deposited clad.

Based on its review, the NRC staff finds the results from the corrosion evaluation are conservative because of the assumptions used to develop the corrosion rates, as discussed above, and the lack of visually discernible low-alloy steel corrosion in the inspection photos of the as-found cladding defects. Additionally, the NRC staff finds that the corrosion evaluation supports the licensee's assumption to use corrosion features with a depth of 1.0 inch in the structural analyses (discussed below).

Structural Evaluation

Indications 1 and 2 were evaluated in licensee's structural analysis. Indication 3 did not warrant an analysis since no damage to the base metal was apparent.

*Indication #1*

The licensee stated that the primary local membrane plus bending stress intensity for the intact RPV transition region was calculated to be 18.3 ksi, a factor of 2.2 below the 1.5 Sm limit of 40.05 ksi, which is greater than the conservative estimate of 1.23 for increase in local stress due to the corrosion feature; the NRC staff finds that the reactor vessel wall in the vicinity of indication #1 still conforms to the design bases in meeting the ASME allowable. Furthermore, the fatigue usage factor for the intact RPV transition region, is reported to be 0.0083, which is well below the allowable limit of 1.0. Thus, the NRC staff finds that the reactor vessel wall in the vicinity of indication #1 still conforms to the design bases in meeting the design life.

*Indication #2*

Since the primary plus secondary stress intensity range is a factor of 1.96 below the allowable value, which is greater than the estimate of 1.25 for the increase in local stress due to the corrosion feature, conservatively based on the wall thickness of the bottom head; the NRC staff finds that the reactor vessel wall in the vicinity of indication #2 still conforms to the design bases in meeting the ASME allowable.

Furthermore, the thickness of the undisturbed region of the bottom head is more than 1.0-inch greater than the required 3.868-inch thickness calculated on the basis of the primary membrane stress due to the design pressure. The licensee explained that it applied a 1.0-inch flaw to the fatigue analysis with a fatigue reduction factor of 4.0 to model the flaw as if it is the root of a thread of a bolt. Applying this factor increases the fatigue usage factor to 0.436, which is still below the allowable limit of 1.0. Thus, the NRC staff finds that the reactor vessel wall in the vicinity of indication #2 still conforms to the design bases in meeting the design life.

Technical Conclusion

Based on its review of the licensee's corrosion and structural evaluation, the NRC staff has determined that there is reasonable assurance the cladding defects in the reactor vessel of NAPS1 will not affect the structural integrity of the reactor vessel that would adversely affect reactor safety during normal plant operations through the end of the Fifth ISI Inspection Interval, which is scheduled to end on April 30, 2029. Therefore, the NRC staff finds that the presence of the cladding defects is acceptable for continued service with one re-examination of its condition to occur no later than the end of the Fifth ISI Inspection Interval, which is scheduled to end on April 30, 2029.

## 4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative to the requirements of paragraph IWB-2420(b) of the ASME Code, Section XI, provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes proposed alternative N1-15-NDE-002 at NAPS1 for the Fifth 10-Year Inspection Interval of the ISI Program for NAPS1, which is scheduled to end on April 30, 2029.

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

Principal Contributor: O. Yee, NRR
M. Breach, NRR

Date: December 28, 2020

SUBJECT: NORTH ANNA POWER STATION, UNIT NO. 1 - RE: REQUEST FOR RELIEF REQUEST N1-I5-NDE-002 INSERVICE INSPECTION ALTERNATIVE (EPID L-2020-LLR-0075) DATED DECEMBER 28, 2020

**DISTRIBUTION**:
PUBLIC
PM File Copy
RidsACRS_MailCTR Resource
RidsNrrDexEmib Resource
RidsNrrDnrlNvib Resource
RidsNrrPMNorthAnna Resource
RidsNrrLAKGoldstein Resource
RidsRgn2MailCenter Resource
MBreach, NRR
OYee, NRR

**ADAMS Accession No. ML20350B806** *Via SE Input NRR-028

| OFFICE | NRR/DORL/LPL2-1/PM | NRR/DORL/LPL2-1/LA | NRR/DEX/EMIB/BC | NRR/DNRL/NVIB/BC* |
|---|---|---|---|---|
| NAME | GEMiller | KGoldstein (BAbeywickrama for) | ABuford* | HGonzalez |
| DATE | 12/23/2020 | 12/22/2020 | 12/10/2020 | 12/10/2020 |
| OFFICE | NRR/DORL/LPL2-1/BC | | | |
| NAME | MMarkley | | | |
| DATE | 12/28/2020 | | | |

**OFFICIAL RECORD COPY**