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

August 22, 2023

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 NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 295 AND 278 RE: LICENSE AMENDMENT REQUEST TO REMOVE THE REFUELING WATER CHEMICAL ADDITION TANK AND CHANGE THE CONTAINMENT SUMP PH BUFFER (EPID L-2022-LLA-0164)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 295 and 278 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (North Anna), Unit Nos. 1 and 2, respectively. These amendments are in response to your application dated November 3, 2022 (Reference 1), as supplemented by letter dated April 13, 2023 (Reference 2).

The amendments revise the North Anna, Unit Nos. 1 and 2, Technical Specifications to eliminate the Refueling Water Chemical Addition Tank and allow the use of Sodium Tetraborate Decahydrate (NaTB) to replace Sodium Hydroxide (NaOH) as a chemical additive (buffer) for containment sump pH control following a loss-of-coolant accident at North Anna, Units 1 and 2. This change will also eliminate active components from the Quench Spray System.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions, please contact me at (301) 415-2481 or Ed.Miller@nrc.gov.

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 295 to NPF-4
2. Amendment No. 278 to NPF-7
3. Safety Evaluation

cc: Listserv

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 295
Renewed License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated November 3, 2022, as supplemented by letter dated April 13, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to paragraph 2.C (2) of Renewed Facility Operating License No. NPF-4, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A, as revised through Amendment No. 295, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented by the completion of the spring 2024 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T.
Markley

Digitally signed by Michael T. Markley
Date: 2023.08.22 06:58:53 -04'00'

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-4
and Technical Specifications

Date of Issuance: August 22, 2023

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 278
Renewed License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated November 3, 2022, as supplemented by letter dated April 13, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to paragraph 2.C (2) of Renewed Facility Operating License No. NPF-7, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented by the completion of the fall 2023 refueling outage

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FOR THE NUCLEAR REGULATORY COMMISSION

Michael T.
Markley

Digitally signed by Michael T. Markley
Date: 2023.08.22 06:59:31 -04'00'

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-7
and Technical Specifications

Date of Issuance: August 22, 2023

ATTACHMENT TO

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

LICENSE AMENDMENT NO. 295

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND LICENSE AMENDMENT NO. 278

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
NPF-4, page 3	NPF-4, page 3
NPF-7, page 3	NPF-7, page 3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.6.8-1	3.6.8-1
3.6.8-2	3.6.8-2

(2) Pursuant to the Act and 10 CFR Part 70, VEPCO to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or component; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

Technical Specifications contained in Appendix A, as revised through Amendment No. 295 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material, without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or component; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations as set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 278 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Additional Conditions</u>

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the insurance of the condition or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission:

a. If VEPCO plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Power Station, the

3.6 CONTAINMENT SYSTEMS

3.6.8 Chemical Addition System

LCO 3.6.8 The Chemical Addition System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Chemical Addition System inoperable.	A.1 Restore Chemical Addition System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify that each sodium tetraborate decahydrate basket is unobstructed, in place and intact.	In accordance with the Surveillance Frequency Control Program
SR 3.6.8.2	Verify that the sodium tetraborate decahydrate baskets collectively contain $\geq$ 16,013 lbm and $\leq$ 22,192 lbm of sodium tetraborate decahydrate.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.3	Verify that a sample from the sodium tetraborate decahydrate baskets provides adequate pH adjustment of borated water.	In accordance with the Surveillance Frequency Control Program

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 295 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AND

AMENDMENT NO. 278 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By application dated November 3, 2022 (Reference 1), as supplemented by letter dated April 13, 2023 (Reference 2), Virginia Electric and Power Company (the licensee) submitted a license amendment request (LAR) requesting changes to the Technical Specifications (TSs) for North Anna Power Station, Units 1 and 2 (North Anna). The proposed changes would revise the TSs to eliminate the Refueling Water Chemical Addition Tank and allow the use of Sodium Tetraborate Decahydrate (NaTB) to replace Sodium Hydroxide (NaOH) as a chemical additive (buffer) for containment sump pH control following a loss-of-coolant accident (LOCA) at North Anna, Units 1 and 2. This change would also eliminate active components from the Quench Spray System.

The supplement dated April 13, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 24, 2023 (88 FR 4219).

2.0 REGULATORY EVALUATION

Regulations

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," paragraph (c) states, in part, that technical specifications will include "Limiting conditions for operation [(LCOs)]" and in 10 CFR 50.36(c)(2), it states "[l] imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." When an LCO is not met, the the licensee shall shut down the reactor or follow any remedial action permitted by the TS until the condition can be met. The regulations in 10 CFR 50.36(c)(3) states that Surveillance requirements (SRs) are requirements relating to test, calibration, or inspection to assure the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” states that the emergency core cooling system (ECCS) must be designed so that its calculated core cooling performance following postulated loss-of-coolant accidents (LOCAs) of different sizes, locations, and properties.

10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” states, in part, licensees shall establish a program for qualifying the electric equipment important to safety as defined in section 50.49(b). The regulation in 10 CFR 50.49(e) states that the electric equipment qualification program must include and be based on the following: temperature and pressure, humidity, chemical effects, radiation, aging, submergence, synergistic effects, and margins.

10 CFR 50.67, “Accident source term,” provides, in part, requirements for licensees who seek to revise the current accident source term used in their design basis radiological analyses.

The regulations in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria [GDC] for Nuclear Power Plants,” establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The NRC issued construction permits for North Anna, Units 1 and 2, before the GDC were issued on May 21, 1971. The North Anna Updated Final Safety Analysis Report (Reference 25), states that the construction permits were based on the design being in conformance with the draft GDC published in 1966. The SRM for SECY-92-223, “Resolution of Deviations Identified during the Systematic Evaluation Program,” dated September 18, 1992 (Reference 3) identified that plants with CPs issued before May 21, 1971, such as North Anna, were not subject to the requirements in 10 CFR Part 50, Appendix A. However, to facilitate initial licensing review, the North Anna Updated Final Safety Analysis Report (UFSAR) (Reference 25) discusses conformance of the draft GDC to the current 10 CFR Part 50 Appendix A GDC. The draft GDC (referred to as AEC (Atomic Energy Commission) criteria in the North Anna UFSAR) are generally equivalent to the current 10 CFR Part 50 Appendix A GDC, applicable to this LAR include:

> *Criterion 1 (*GDC 1), “Quality standards and records,” states, Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, accuracy, and sufficiency, and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

> *Criterion 4* (GDC 4), “Environmental and missile design bases,” states, Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal

operation, maintenance, testing, and postulated accidents, including LOCAs. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

The General Design Criteria 4 (GDC-4) has undergone significant changes. The revised GDC-4 ([North Anna UFSAR] References 14 and 15) approved the use of leak-before-break technology for eliminating the dynamic effects of postulated pipe ruptures in high energy piping including primary coolant piping from the design basis of pressurized water reactor's (PWR). Implementation of the revised rule permits the removal of pipe whip restraints, jet impingement barriers, and other related changes. The rule clearly allows removal of plant hardware which it is believed negatively affects plant performance and safety. However, as stated in the Federal Register/Vol. 15, No. 70/ of April 11, 1986, and subsequently in broad scope rule in the Federal Register/Vol. 52, No. 207/ of October 7, 1987, containment design, emergency core cooling, and environmental qualification requirements are not influenced by the revised rule.

*Criterion 14 (*GDC 14), "Reactor coolant pressure boundary [RCPB]," states, The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

Criterion 16 (GDC 16), "Containment design," states, Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 38 (GDC 38), "Containment heat removal," states, A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 41 (GDC 41), "Containment atmosphere cleanup," States systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to ensure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 50 (GDC 50), “Containment design basis,” states, The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Regulatory Guidance

Regulatory Guide (RG) 1.183, “Alternative Radiological Sources Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, July 2000, (Reference 4).

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition,” (SRP) Section 3.11 “Environmental Qualification of Mechanical and Electrical Equipment,” Revision 3, (Reference 5) provides guidance on EQ of mechanical and electrical equipment.

NUREG-0800, Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,” Revision 3, (Reference 6) addresses determination of rupture locations and dynamic effects associated with the postulated rupture of piping inside and outside containment.

NUREG-0800, Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, (Reference 7).

NUREG-0800, Branch Technical Position (BTP) 3-3, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” Revision 3 (Reference 8).

NUREG-0800, BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” Revision 3 (Reference 9).

NUREG-0800, Section 6.2.2, “Containment Heat Removal,” Revision 5, (Reference 10) addresses containment heat removal under postaccident conditions.

NUREG-0800, Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, (Reference 11) addresses containment spray and the spray additive or pH control systems.

NUREG-0800, BTP 6-1, “pH for Emergency Coolant Water for Pressurized Water Reactors,” (Reference 12) addresses the minimum value of pH in post-accident containment sprays.

Inspection and Enforcement Bulletin (IEB) 79-01B, “Environmental Qualification of Class 1E Equipment.” (Reference 13)

Technical Guidance

“Roark’s Formulas for Stress & Strain,” 6th Edition, Warren C. Young. (Reference 14)

American Institute of Steel Construction (AISC Manual of Steel Construction, 9th Edition. (Reference 15)

American Society of Civil Engineers (ASCE)-8-90, “Specification for the Design of Cold-Formed Stainless Steel Structural Members.” (Reference 16)

“The Behavior of Welded Joints in Stainless and Alloy Steels at Elevated Temperatures,” Oak Ridge National Lab Report Number ORNL-4781, August 1972. (Reference 17)

3.0 TECHNICAL EVALUATION

3.1 Background

A chemical addition tank (CAT) at North Anna, Units 1 and 2, is used to add NaOH to the quench spray (QS) and recirculation spray (RS) systems to reduce the amount of radioiodine released during a postulated LOCA. According to NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” (Reference 18) iodine released from the damaged core to the containment after a LOCA is composed of 95 percent cesium iodide, which is a highly ionized salt and soluble in water. The function of the NaOH additive is to maintain the pH of the containment sump water in the basic range, which mean a pH above 7 at a reference temperature of 25 degrees Centigrade (°C) (77 degrees Fahrenheit or °F). A basic pH minimizes the conversion of water-soluble cesium iodine to elemental iodine, which can be re-evolved as a gas into containment and potentially released to the atmosphere. The guidance in NUREG/CR-5950, “Iodine Evolution and pH Control,” (Reference 19) describes acids and bases in containment and their relationship to iodine chemical forms and evolution.

The LAR proposes using baskets of soluble NaTB on the containment floor, rather than NaOH from an active spray system, to maintain a basic sump pH during a postulated LOCA. The guidance in SRP Section 6.5.2 and RG 1.183 (Appendix A) identify a pH of 7 as the value below which molecular iodine should be assumed to evolve from the sump water. The staff also evaluated the changes to TS Section 3.6.8, which currently describes the NaOH spray additive requirements but would be modified to contain the requirements related to the NaTB baskets.

The resolution of Generic Safety Issue (GSI) -191, “Assessment of Debris Accumulation on PWR Sump Performance,” showed that the sump pH buffer affects the type and amounts of chemical precipitates that may form in postulated post-LOCA recirculating water. Chemical precipitates are a result of interaction between materials in containment (e.g., insulation and metallic materials) and the sump fluid, and they could degrade the performance of the emergency core cooling system (ECCS) by contributing to blockage of sump strainers and fuel assemblies, and the loss of heat transfer. Studies of these “chemical effects” have included both NaTB and NaOH. WCAP-16530-NP-A (Reference 20) provides additional references for GSI-

191 chemical effects testing and evaluation. The licensee's response to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," describes the chemical effects analysis for North Anna, Units 1 and 2. (Reference 21)

In addition, the pH of the sump fluid may affect corrosion of ECCS components. To reduce the likelihood of stress corrosion cracking (SCC) in austenitic stainless steel, BTP 6-1 provides a minimum pH criterion of 7.0 and states that the likelihood of SCC decreases with increasing pH between 7.0 and 9.5. It also notes that aluminum corrosion and the associated hydrogen gas evolution should be considered for pH greater than 7.5.

3.2 Containment Sump pH Buffer

3.2.1 Containment Sump pH Buffer, Description

The licensee proposes installing eight stainless steel baskets containing NaTB decahydrate in containment for adjusting the post LOCA sump pH. The SRs would be revised to require the NaTB baskets to be unobstructed, in place, and intact, and to collectively contain between 16,013 pounds (lbm) and 22,192 lbm of NaTB decahydrate that shall provide adequate pH adjustment of the borated water. Each basket will be marked to indicate the minimum acceptable level of NaTB. In response to a request for additional information, the licensee clarified that the maximum mass of NaTB allowed by the proposed TS will not be exceeded due to the physical size of the baskets and the maximum volume to which they could be filled.

The maximum dissolution time for the NaTB was determined for both single-train and full Engineered Safety Feature (ESF) conditions. Conservative inputs such as minimum rate of rising water and minimum temperature profile were used to determine the maximum time required to dissolve all of the NaTB.

The licensee used an analytical model to determine equilibrium sump conditions. The model was benchmarked to site-specific buffer testing using the same buffer material that will be installed in the plant. Conservative inputs were used to determine the minimum and maximum pH values. For instance, the low pH calculation assumed high levels of hydrochloric acid generated by irradiation of cable insulation and high levels of nitric acid generated by irradiation of water. Maximum quantities of boric acid and maximum core iodine release were also assumed for the low pH case. The opposite assumptions were made when determining the high pH case. The analysis showed that with the conservatisms included in the modelling, a pH of greater than 7.0 is acquired from the time when recirculation spray is credited for iodine removal up to 30 days and that the upper pH limit of 9.0 ($t \leq 20$ minutes) and 8.5 ($t > 20$ minutes) is not exceeded.

The licensee states that testing of the buffer will be required each refueling outage to ensure that the chemical composition of the buffer and its buffering ability do not change over time. A sample will be taken from each of the 8 baskets and tested to ensure that it maintains its original buffering capacity. The baskets will also be surveilled to ensure that the NaTB maintains a loose consistency and does not clump and solidify due to environmental conditions.

3.2.2 Containment Sump pH Buffer, NRC Staff Evaluation

The NRC staff reviewed the LAR to determine if the proposed amount of buffer (NaTB) is sufficient to prevent iodine re-evolution by raising the pH to at least 7.0 prior to the beginning of recirculation and maintaining it above 7.0 for the 30-day post-LOCA period, without causing it to exceed 9.0. As part of its review, the staff performed independent calculations related to basket geometry and capacity, pH at the onset of recirculation, pH after 30 days following the postulated LOCA, and the amounts of strong acids (nitric and hydrochloric) generated in the post-LOCA environment.

Based on its review of the licensee's methodology, inputs, and analyses, the NRC staff finds the licensee met the criteria in SRP Section 6.5.2 and RG 1.183 for maintain a pH of at least 7.0 (at 25 °C) in the sump fluid from the time of recirculation to 30 days after the start of the postulated LOCA. Therefore, the NRC staff finds the proposed changes meet 10 CFR 50.67 for evaluating DBA consequences and GDC 41 as it relates to pH control for preventing post-LOCA iodine re-evolution.

Additionally, with an achieved sump pH of 7.0 or greater using NaTB existing dose-related safety margins would not be altered by this amendment. Therefore, the NRC staff finds the proposed changes would not affect the licensee's compliance with 10 CFR 50.67 for evaluating DBA dose consequences and GDC 41 as it relates to pH control for preventing post-LOCA iodine re-evolution.

3.3 ECCS Strainer Blockage

3.3.1 ECCS Strainer Blockage, Description

Section 3.1.3 of the LAR addresses the effect of the pH buffer change on the potential for ECCS strainer blockage due to formation of chemical precipitates in the sump fluid (chemical effects). The licensee considers the current chemical precipitate evaluation bounding and did not submit a new evaluation for the proposed NaTB buffer. This is based on the current chemical effects analysis attributed mostly to aluminum corrosion, reduction in the aluminum corrosion for NaTB buffer compared to NaOH in the licensee's chemical effects methodology, the unchanged amount of chemical effects source materials, absence of additional chemical effects associated specifically with NaTB, and current margin between the chemical precipitate quantity used in strainer testing and the quantity predicted by the chemical effects methodology. The proposed change would eliminate the NaOH injection spray which is currently the most corrosive post-LOCA buffering option with respect to aluminum corrosion. In addition, the long-term pH with NaTB would be lower than with NaOH, which is less corrosive for aluminum.

3.3.2 ECCS Strainer Blockage, NRC Staff Evaluation

The NRC staff evaluated the licensee's existing North Anna, Units 1 and 2, chemical effects analysis considering the changes proposed in the LAR. The amount of chemical precipitate in the licensee's analysis is determined primarily by aluminum corrosion. The licensee's methodology predicts the aluminum corrosion rate, and, therefore, the amount of chemical precipitate decreases with decreasing pH over the range for NaOH and NaTB sump pH buffers. Because the use of NaTB would eliminate the NaOH injection spray phase and reduce the long-term pH of the sump solution, the licensee's methodology would predict less aluminum corrosion and, therefore, less chemical precipitate. In addition, there are no chemical effects specific to NaTB or the stainless-steel basket materials. Based on these factors, the NRC staff finds that the licensee's existing chemical effects analysis remains bounding and that the proposed changes would meet 10 CFR 50.46 as it relates to the North Anna, Units 1 and 2, chemical effects analysis.

3.4 Corrosion of Containment Materials

3.4.1 Corrosion of Containment Materials, Description

Sections 3.1.4 and 3.1.5 of the LAR address the criteria in NUREG-0800, Branch Technical Position 6-1, "pH for Emergency Coolant Water for Pressurized Water Reactors," indicating that the pH of the recirculating sump solution have a minimum pH of 7.0 to reduce the probability of stress corrosion cracking of austenitic stainless steel components, and that hydrogen generation from aluminum corrosion should be considered if the pH is greater than 7.5. The licensee stated that the proposed amount of NaTB buffer achieves a minimum long-term pH of 7.0. The licensee also stated that evaluation of hydrogen generation is not affected because the long-term pH range for the proposed amount of NaTB buffer maintains the pH between 7.0 and 8.5, which is consistent with the current licensing basis.

3.4.2 Corrosion of Containment Materials, NRC Staff Evaluation

The NRC staff evaluated the LAR to determine if the proposed containment sump pH will be in a range that does not cause SCC of austenitic stainless-steel components or an increase in the corrosion rate of aluminum. The licensee's pH calculations and the NRC staff's corresponding evaluation, which are discussed above in Sections 3.2 and 3.3 of this evaluation, respectively, indicate that the post-LOCA sump pH will remain above 7.0 and below 8.5 by the time recirculation spray mode is credited for iodine removal.

For austenitic stainless steel, the criteria in the BTP 6-1 guidance are that for a low probability of SCC, the pH should be 7.0 or greater, and that an increasing pH in the 7.0 to 9.5 range increases the assurance that SCC will not occur. For aluminum, BTP 6-1 includes a criterion that for pH greater than 7.5, consideration of hydrogen generation from aluminum corrosion should be considered. In Section 3.1.5 of the LAR, the licensee stated that hydrogen generation currently assumed would be unaffected by the proposed changes because the long-term predicted sump pH range of 7.0-8.5 is unchanged from the current design pH range.

Based on the predicted sump pH being at least 7.0 early in the post-LOCA period, and within the licensee's current analysis range of 7.0-8.5 for 30 days, the NRC staff finds the proposed NaTB pH buffer changes acceptable with respect to SCC of austenitic stainless steel at lower pH, and corrosion of aluminum at higher pH. Therefore, the NRC staff finds the proposed use of NaTB meets GDC 14 with respect to assuring the low probability of abnormal leakage or failure of the reactor coolant pressure boundary and safety-related structures.

3.5 Proposed TS Changes

3.5.1 Proposed TS Changes, Description

The proposed wording in TS 3.6.8 would delete all existing SRs, which are related to using NaOH as a pH buffer, and would replace them with SRs related to using NaTB as a pH buffer. Specifically, the new proposed SRs are:

- Revised SR 3.6.8.1 would state: "Verify that each sodium tertraborate decahydrate basket is unobstructed, in place and intact."

- Revised SR 3.6.8.2 would state: “Verify that the sodium tertraborate decahydrate baskets collectively contain ≥16,013 lbm and ≤22,192 lbm of sodium tertraborate decahydrate.”

- Revised SR 3.6.8.3 would state: “Verify that a sample from the sodium tertraborate decahydrate baskets provides adequate pH adjustment of borated water.”

No changes are proposed to the existing Limiting Condition for Operation (LCO) or ACTIONS.

3.5.2 Proposed TS Changes, NRC Staff Evaluation

The NRC staff reviewed the proposed TS changes to assess whether the TS required amount of NaTB is sufficient to maintain the sump pH at 7.0 or greater following a LOCA and that requirements for periodic sampling and testing of the buffer provide reasonable assurance it will function as required.

The NRC staff determined that the proposed changes to TS 3.6.8 are acceptable as NaTB will serve as an adequate buffer for post LOCA sump pH control as discussed above in Section 3.2 of this SE. Additionally, the staff finds it acceptable to maintain a combined weight of ≥16,013 lbm and ≤22,192 lbm of NaTB decahydrate, as this amount of buffer will be adequate to maintain the sump pH greater than 7.0 and less than 8.5, as discussed in the NRC staff’s technical evaluation above. The proposed changes include required periodic testing of the NaTB stored in containment to confirm the NaTB buffering capabilities are within its design limits. Based on the above, the NRC staff finds the proposed SRs to be adequate to ensure that the necessary quality of systems and components are maintained, that facility operation will be within safety limits, and that the LCO will be met.

On January 31, 2011 (Reference 22), North Anna was approved to use a Risk-Informed Surveillance Frequency Control Program (SFCP). The licensee proposed to have the frequency included in, and controlled by, this program. As discussed in the Safety Evaluation and associated documentation for this approval, SRs are able to be included in the program except:

- Frequencies that reference other approved programs for the specific interval;

- Frequencies that are purely event-driven;

- Frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs; and

- Frequencies that are related to specific conditions or conditions for the performance of a surveillance requirement.

The NRC staff confirmed that the proposed SRs do not meet any of these criteria and are, therefore, appropriate for inclusion in, and control by the SFCP.

Further, the NRC staff finds that the existing LCO does not specify the method of buffering sump pH (i.e., NaOH versus NaTB) nor would the change in method of buffering necessitate different actions or completion times for an inoperable system. Therefore, the existing LCO and ACTIONS remain appropriate. Based on the above, the NRC staff finds the proposed changes would continue to meet 10 CFR 50.36 with respect to incorporating the use of NaTB into the Chemical Addition System TSs.

3.6 Containment Response

3.6.1 Containment Response, Description

In its LAR, Attachment 1, Section 4.1, "Applicable Regulatory Requirements/Criteria," the licensee stated that the ability of the QS and RS systems to cool the reactor core and return the containment to subatmospheric pressure and maintain it at subatmospheric pressure is not affected by the proposed change, and, therefore, the proposed change will not impact the ability of North Anna, Units 1 and 2, to comply with the requirements of GDC 38.

In its supplement dated April 13, 2023, the licensee stated that the analysis of record (AOR) models for the containment response and available NPSH for QS and RS pumps use volumetric flowrates. The change in density of the QS liquid with the removal of the NaOH is negligible because the volumetric percent of NaOH solution in the QS flow is approximately 3.2 percent which results in approximately 0.43 percent increase in its density compared to the density of the borated solution in the refueling water storage tank (RWST). The percent of NaOH present in the minimum containment sump volume at the start of recirculation is conservatively approximated to be 3.33 percent which results in an approximate 0.45 percent increase in density of the containment sump liquid compared to the containment sump liquid without the addition of NaOH from the CAT.

3.6.2 Containment Response, NRC Staff Evaluation

The NRC staff noted that the combined effect of the items described below could affect the following: (a) AOR for LOCA containment pressure and temperature response, and (b) AOR for the available net positive suction head (NPSH) of pumps that draw water from the sump during LOCA recirculation phase.

- The reduction of containment free volume due to the addition of NaTB baskets in the lower level of the containment basement.

- Removal of NaOH from the RWST water in the AOR would affect the QS water density, and, therefore, QS mass flowrate and the QS pump performance.

- Addition of NaTB and removal of NaOH from the containment sump water in the AOR would affect its density and, therefore, its RS mass flowrate and RS pump performance during the LOCA recirculation phase.

- The change in RS mass flowrate may affect the RS cooler overall heat transfer coefficient and therefore may change its performance.

- The change in the spray water composition may affect the heat transfer characteristics of the QS and RS droplets.

- The available NPSH AOR of the RS pump may be affected due to change in density of the sump liquid. Specifically, SRP Section 6.2.2 (Reference 10) states, in Item 2 under the heading "SRP Acceptance Criteria," that the analysis should demonstrate that the

RS pumps available NPSH should be greater than or equal to their required NPSH (to avoid pump cavitation) to satisfy GDC 38 as it relates to the capability of the containment heat removal system to accomplish its safety function.

3.6.2.1 Containment Volume Reduction

The licensee stated in its supplement that the decrease in the AOR containment free volume due to addition of the NaTB baskets is 420 ft^3 which is approximately 0.023 percent and is considered negligible.

Based on its review, the NRC staff finds it acceptable that the decrease in the containment free volume by 0.023 percent is negligible because this small volume would have an insignificant effect on the LOCA containment pressure and temperature response.

3.6.2.2 Density of QS and RS Liquids and Flowrates

The licensee stated in its supplement dated April 13, 2023, that during the start of the LOCA recirculation phase, with the proposed NaTB baskets in containment, a maximum dissolution of NaTB would result in a small increase of approximately 1.57 percent in the containment sump liquid density which would decrease as the volume of the sump liquid increases due to the break. The small increase in the sump liquid density would be expected to only slightly increase the Reynolds number resulting in small or no change in piping friction factor with a corresponding negligible impact to system and associated RS and QS pump flow rates. Therefore, at the start of recirculation and beyond, the change in density of the containment sump liquid due to the removal of NaOH from the CAT and the addition of NaTB from the baskets is negligible.

Based on its review, the NRC staff finds that the QS and RS mass flowrates would not be significantly impacted at the start of recirculation and beyond because the change in density of the containment sump liquid due to the removal of NaOH from the CAT and the addition of NaTB from the baskets is negligible.

3.6.2.3 RS Cooler Performance

The parameters related to the proposed change on which the RS cooler performance depends are the RS flowrate and the overall heat transfer coefficient of the RS cooler. As discussed above, the RS volumetric flow rate in the AOR is unaffected. The heat transfer coefficient of the RS cooler in the AOR which is based on the NRC-approved GOTHIC [Generation of Thermal-Hydraulic Information for Containments] methodology documented in Topical Report (TR) DOM-NAF-3-NP-A (Reference 23) does not consider minor density effects due to the currently used NaOH buffer in the CAT. The GOTHIC version 7.2a technical manual (used in the AOR) states that the heat transfer coefficient for coolers is calculated *using pure water phase properties based* on the American Society of Mechanical Engineers (ASME) steam tables.

Based on its review, the NRC staff finds that the AOR for the RS cooler performance would not be affected by replacing the dissolved NaOH with NaTB in the RS liquid because of the replacement would result in only a minor change in liquid density and the heat transfer coefficient is the same as in the AOR.

3.6.2.4 Properties of QS and RS Droplets

The licensee stated that the droplet sizes modeled in the AOR are not based on the best-estimate droplet sizes which may potentially be affected by minor changes in the density of the liquid. Instead, the spray nozzle vendor determined Sauter mean droplet diameter at multiple differential pressures across the nozzle are input in the AOR GOTHIC containment model. The Sauter mean diameter of the droplet is same as the ratio of volume to surface area as the entire ensemble. The licensee conservatively adjusted the Sauter mean diameter in the AOR to generate the most limiting results. Section 3.3 of the NRC SE (included in the TR) for TR DOM-NAF-3-NP-A methodology describes the droplet modeling using pure water properties and adjustment of the Sauter mean diameter as conservative.

Based on its review, the NRC staff finds it acceptable that the QS and RS droplet sizes would not be affected by the minor change in the liquid density because the bounding droplet size used in the AOR is independent of minor density effects.

3.6.2.5 Available NPSH

Section 3.3 of the NRC staff SE for the TR DOM-NAF-3-NP-A, on which the available NPSH AOR is based, mentions the following assumptions for the reactor coolant system containment model to ensure a conservative calculation of the available NPSH for pumps that draw water from the containment sump during the LOCA recirculation phase:

- In the GOTHIC model, the heat transfer coefficient for the containment heat sinks was increased by applying a multiplier of 1.2 to compensate for any non-conservative values generated by the Direct Diffusion Layer Model (DLM). In the previous GOTHIC 7.2 version, the built-in Mist DLM (MDLM) heat and mass transfer model option was replaced with the Direct DLM model option in the GOTHIC 7.2a version used in the AOR. This replacement was done by the licensee to address the NRC staff concerns identified in then SE (Reference 24) for a Kewaunee license amendment.

- Conservatively, without assuming any loss, the AOR considers the entire spray water injected as droplets into the containment atmosphere for containment cooling during a LOCA.

- A conservative water holdup volume is subtracted from the containment liquid volume to reduce the sump water height.

- Conservatively, the analysis is based on the upper limit of containment free volume and minimum initial containment pressure.

Based on its review, the NRC staff finds that the NPSH AOR is sufficiently conservative that the containment temperature and the available NPSH for the RS pumps that draw water from the sump during LOCA recirculation phase is not affected by replacing the dissolved NaOH with NaTB in the RS liquid.

3.6.2.6 Containment Response, Conclusion

The NRC staff concludes that, from the LOCA containment response standpoint, the proposed TS Section 3.6.8 change of replacing NaOH solution in the CAT with granular NaTB in baskets located in the lower level of the containment basement is acceptable based on the following:

- Change in containment volume due to the addition of the NaTB baskets in the lower level of the containment basement is negligible and does not affect the AOR LOCA containment pressure and temperature response;

- The AOR QS and RS mass flowrates that cool the containment during LOCA are not significantly affected;

- The AOR RS cooler performance is not significantly affected by the addition of NaTB buffer instead of NaOH buffer to the RS liquid;

- The QS and RS droplet sizes are not significantly affected by the minor change in the liquid density because the bounding droplet size used in the AOR is independent of minor density effects;

- NPSH AOR is conservative enough that the containment temperature and the available NPSH for the RS pumps that draw water from the sump during LOCA recirculation phase are not affected by the removal of the NaOH buffer in the CAT or the addition of the NaTB baskets in containment;

- The AOR GOTHIC models use conservative inputs to maximize containment pressure for depressurization cases and minimize containment pressure for available NPSH analyses as described in TR DOM-NAF-3-NP-A; and

- Since the GOTHIC model is based on ASME steam tables for the properties of pure water, the AOR is not affected by the CAT removal and the addition of NaTB baskets in containment as they are not explicitly modeled.

Based on the above, the NRC staff finds that:

- GDC 16 remains satisfied because the containment and associated systems will continue to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require;

- GDC 38 remains satisfied because the containment heat removal system in conjunction with the functioning of the associated safety-related systems will perform its safety function to reduce rapidly the containment pressure and temperature following any LOCA and maintain them at acceptably low levels; and

- GDC 50 remains satisfied because the containment structure and its internal compartments, including access openings, penetrations, and the containment heat removal system will accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Based on the above, the NRC staff finds that the containment response will remain appropriate with the proposed change.

3.7 Structural Evaluation

3.7.1 Structural Evaluation, Description

Eight NaTB baskets, constructed from stainless-steel (Type 304 SS) material, will be installed (anchored) on the containment floor elevation at 216 feet 11 inches at North Anna, Units 1 and 2. An empty NaTB basket weighs approximately 1,465 pounds, and a basket loaded with granular form of NaTB weighs approximately 4,450 pounds. Each NaTB basket has nominal dimensions of 5 feet (60 inches) by 6 feet (72 inches) in horizontal directions and by 1.83 feet (22 inches) in vertical direction. The licensee also designed four (4) caster wheels (stainless steel, Type 304 and 2205 Duplex) to facilitate movement during outages, if required. The licensee states that the NaTB basket locations were selected such that they are either sufficiently protected from the effects of High Energy Line Breaks (HELBs) using barriers, restraints, and distance, or the piping systems which are adjacent to the NaTB baskets are not susceptible to a postulated HELB.

The NaTB baskets are classified as Non-safety Quality (NSQ) based on not being functionally safety-related; however, they are required to be seismically anchored to prevent damage to nearby safety-related equipment. The baskets are also required to remain functional during and/or after a design-basis earthquake (DBE) events. Therefore, the baskets are designed to meet Seismic II/I requirements and maintain their structural integrity during DBE events.

The NaTB basket structural components are designed to meet the requirements of the 9th Edition of AISC. The licensee provided design calculations pertaining to the NaTB basket in report: CEM-0226, Revision 0, in Attachment 2 in a letter dated April 13, 2023, (herein designated as "the report"), and provided in responses to the Requests for Additional Information (RAIs) (Reference 2). The licensee also provided the fabrication drawings, No's. 1901155-11715-FM-1D, Revision 0 and 1901156-120505-FM-1D, Revision 0, in Attachment F of the report.

3.7.2 Structural Evaluation, NRC Staff Evaluation

The license stated that its approach did not determine the natural frequency of the NaTB basket frame. The NRC staff performed an independent confirmatory check to ensure the natural frequency of the NaTB basket frame is within the rigid range as defined having a natural frequency greater than 33 Hz. The NRC staff used the equation for a concentrated center load of both ends simply supported beam from "Roark's Formulas for Stress & Strain," 6th Edition, to determine the natural frequency of the NaTB basket frame. The NRC staff used the elastic and the cross-sectional properties of the frame members (hollow structural sections (HSS)-2½ x 2½ x ¼) and weight/dimensional input information from the report to perform independent calculations. The NRC staff determined that the natural frequency of the NaTB basket frame is well over the 33 Hz. range, such that it can be considered a rigid structure. Therefore, the NRC staff concluded that the zero-period acceleration (ZPA) values (a high-frequency acceleration at non-amplified portion of the response spectrum) can be applied for the design of members of NaTB basket structure. In Section 3.7.2, "Seismic System Analysis," of the North Anna Updated Final Safety Analysis Report (Reference 25) , the licensee defined the DBE corresponds to the safe-shutdown earthquake (SSE), and one-half the safe-shutdown earthquake is analogous to the operational-basis earthquake (OBE). In Section 2.5.2.5.1, "Seismic Event of August 23, 2011," the licensee developed amplified response spectrum (ARS) curves at 5 percent damping from the time histories and tabulated this event's ground accelerations in three orthogonal directions as: 0.264g in horizontal, North-South, direction, 0.109g in horizontal, East-West, direction, and 0.118g in vertical direction. In Section 9.3, “Seismic Properties,” of the report, the licensee provided the DBE horizontal (g_h) / vertical (g_v) acceleration values of 0.613g and 0.4714g, respectively. To account for multi-mode actions, the licensee multiplied those accelerations by a factor of 1.3 to determine the DBE horizontal (a_h) / vertical (a_v) acceleration values of 0.8g and 0.61g, respectively, which were conservatively used in the design of structural members, welds, and anchors of NaTB basket. Based on the comparison of the acceleration values of ARS and DBE, the NRC staff concludes that the analytical analyses performed in the report is conservative because the DBE acceleration values are three times greater the ARS acceleration values.

As shown in Figure 3.8-3, “Typical Detail: Foundation Mat and Base,” the elevation at the top of the containment foundation mat is 214 feet 5 inches and the bottom elevation of a 10-foot-thick containment foundation mat would be 204 feet 5 inches (foundation input response spectra (FIRS) elevation). The NaTB baskets will be installed in the containment floor at 216 feet 11 inches elevation; however, the acceleration values for the analyses are from the 204 feet 5 inches elevation. The staff concluded that the 10-foot-thick containment foundation mat will behave in a cohesive manner throughout the thickness during a seismic event thus the DBE accelerations at the ground level elevation (204 feet 5 inches) will be broadly equivalent at the top of the foundation elevation (214 feet 5 inches). Therefore, based on its engineering judgment, the staff determined that the difference in DBE acceleration values between the top of the foundation elevation (216 feet 11 inches) and where the NaTB baskets will be installed, at elevation 214 feet 5 inches, is negligible since the difference is only 2 feet 6 inches or about 1 percent of the total elevation. Based on the above, the NRC staff concludes that the analyses performed in the report is conservative because the DBE acceleration values used in the design of NaTB baskets are three times greater than the ARS acceleration values.

The licensee used the NQA-1 compliant finite element (FE) code of STAAD.Pro to generate the NaTB basket model and the analysis was performed under the unfactored design loads of dead, pressure (chemical), and seismic loadings per the North Anna UFSAR. The STAAD.Pro program is a comprehensive structural FE analysis and design application code that provides analytical analysis capabilities to perform analysis on any structure exposed to static, dynamic, wind, earthquake, thermal, and moving loads with powerful visualization capabilities with the applications of a wide range of design codes. The licensee provided the mathematical FE model, inputs, and outputs of STAAD.Pro program in Attachment B of the report and the results of displacements, forces and stresses at the model nodes/members were listed in tabular format. The licensee created the load combinations within the STAAD.Pro FE program using seismic acceleration in orthogonal directions. The licensee also described that the results from STAAD.Pro, which uses AISC Manual of Steel Construction, 9th Edition code provisions, are also converted to show results per ASCE-8-90 code provisions.

The licensee considered the elevated temperatures of 280°F due to the post LOCA and determined the allowable stress of 15.34 ksi by linear interpretation using the allowable stress values from Table A-3 of ASME B31.1 for the material of ASTM A213, Grade TP304 for the temperatures between 200°F - 300°F. Therefore, the licensee used this reduction in allowable stress in comparing to the calculated stresses on the NaTB basket members.

The licensee described that the basket is anchored to the containment floor with structural steel angles that have slotted holes in the horizontal direction which provides a nonrestrictive thermal expansion of the NaTB baskets during a LOCA event, thus the thermal stresses in the basket members will be insignificant. The licensee also described that weepholes in the NaTB basket provides venting for closed structural sections to ensure that the basket members will not be subject to exterior containment pressure and water buoyancy force during a LOCA event. The licensee considered the anchorage configuration tolerance as "2 inches by 2 inches." The anchorage will be installed in accordance with plant processes which provide guidance and instruction for the installation of anchors that includes rebar scanning of the reinforced concrete floors in containment at installation locations. The best anchorage configuration dependent on the rebar scan results will be approved by engineering personnel supporting North Anna prior to the installation of the NaTB baskets. The licensee performed the analytical design calculations of each anchorage under the governing maximum anchorage reactions in the report per the current fabrication drawings.

The licensee provided the weld stresses with respective weld geometries due to the end-forces tabulated in the spreadsheets in Attachment D of the provided calculation with the appropriate formulas. The welded joints across the entire NaTB basket structure are conservatively designed to resist the enveloping beam end-forces as provided from the STAAD.Pro output. The licensee referenced a report which evaluated the behavior of welded joints in stainless and alloy steels at elevated temperatures generated by Oak Ridge National Laboratory for the U.S. Atomic Energy Commission in 1972 (ORNL-4781) conservatively using the tensile strength value of 72.58 ksi at 900°F. The licensee used the allowable strength design safety factor per ASCE 8-90 (Reference 16) to calculate the allowable strength of weld as 28.96 ksi. The licensee identified the maximum weld stress as 23.3 ksi and calculated the interaction ratio of 0.8 by taking the ratio of the maximum weld stress and the allowable strength of weld. The NRC staff confirmed that the interaction ratio of 0.8 is for the worst weld configuration. Based on the above, the NRC staff confirmed that the calculated weld stresses are based on the conservative loading combinations and are well below the allowable stress level of 28.96 ksi, and are, therefore, acceptable.

Based on the above, the NRC staff has determined that the licensee has sufficiently considered the load combinations of the weight of the basket, the NaTB chemical weight and pressure loading on basket walls, and seismic loading as input to the STAAD.Pro computer code to obtain displacements, forces and stresses in the members of the NaTB basket and design the members and weld joints to sustain the DBE acceleration in accordance with the AISC Manual of Steel Construction, 9th Edition. Therefore, the NRC staff finds the licensee's NaTB basket design will continue to meet GDC 1 and is, therefore acceptable with respect to its structural design.

3.8 Quench Spray Header Piping Stresses

3.8.1 Quench Spray Header Piping Stresses, Description

In its LAR, the licensee proposes to modify the caustic addition piping outside containment by cutting and capping it at the connection to the RWST. The QS system pump suction piping design pressure and temperature do not meet the criteria to be classified as high energy line piping because those parameters are less than the classification of high energy piping outside of containment per BTP 3-3. This classification is also reflected in Section 3C.2.2.1 of the North Anna UFSAR, which similarly defines high energy as piping with a maximum operating pressure exceeding 275 pounds per square inch gage (psig) or the maximum operating temperature exceeding 200°F.

The licensee also stated that the proposed change will not alter the seismic classification of the QS system pump suction piping. Specifically, the CAT supplies water directly to the RWST near the suction piping for the QS pump. Therefore, no new supports or revisions to existing supports on the QS suction piping are needed and the stresses remain within allowable stress limits for the modified configuration of the piping.

3.8.1 Quench Spray Header Piping Stresses, NRC Staff Evaluation

The NRC staff confirmed that Table 6.2-39 of the North Anna UFSAR lists the discharge of the QS Pump to be 150 psig and that the operating temperature of the RWST is 40-50°F (design temp 150°F). As these values are within both the BTP 3-3 and UFSAR limits, the NRC staff finds that no further HELB consideration is necessary for the QS pump suction piping.

Given that the CAT tank piping does not directly connect to the QS suction piping, the NRC staff concludes that the proposed configuration of the CAT tank piping associated with this LAR does not affect the seismic qualification of the QS system suction piping. Therefore, the NRC finds that this the QS system would continue to meet the requirements of GDC 4.

3.9 Environmental Qualification for Components in Containment

3.9.1 Environmental Qualification for Components in Containment, Description

Section 3.11.2.9 of the North Anna UFSAR discusses environmental qualification of electrical equipment. This section states, in part, that as identified in NRC IEB 79-01B, Supplement 2, all reactors with operating licenses as of May 23, 1980, will be evaluated against the guidelines included with IEB 79-01B. For those plants with a construction permit granted after July 1, 1974, and operating license granted after May 23, 1980, the equipment will be qualified to the requirements of NUREG-0588, Category II. Therefore, the equipment qualification regulatory basis is IEB 79-01B for NAPS Unit 1 and NUREG-0588, Category II for NAPS Unit 2.

Section 3.1.6, "Environmental Qualification (EQ) of Equipment," of the LAR provides the licensee's evaluation on the impact of the proposed changes on the electrical equipment subject to 10 CFR 50.49. The proposed change of eliminating the CAT to allow the use of NaTB to replace NaOH as a chemical additive for containment sump pH control following a LOCA would result in a change in the chemical environmental parameters of the electrical equipment subject to 10 CFR 50.49. Currently, the containment spray solution is alkaline due to the direct addition of NaOH to the borated solution from the refueling water storage tank (RWST). According to the licensee, equipment in the NAPS EQ Program was qualified using a chemical spray with a pH range of 8.5 to 10.5 for the first four hours and a pH range of 7.0 to 8.5 from four hours to 120 days. The changes proposed in this LAR would result in the containment spray solution during the injection mode being acidic consisting of the borated solution from the RWST only.

3.9.2 Environmental Qualification for Components in Containment, NRC Staff Evaluation

The licensee's evaluations relied upon available industry and technical/research data regarding the chemical resistance of materials for acidic and alkaline sprays as well as the corrosion rate from the spray composition for the enclosures that house part of the equipment. The licensee considered the chemical resistance of organic materials, the corrosive effects of metallic materials exposed to the spray, and the duration of the initial acidic spray followed by the longer-term alkaline spray. The licensee also evaluated the physical installation to determine which parts of the component would be subjected to direct spray. The licensee credited housing and conduit for protection against chemical spray. Given that the licensee's evaluation utilized appropriate available data regarding chemical resistance and that its evaluations confirmed that

EQ equipment located in the containment remains qualified for the altered containment and recirculation sprays without the need for additional protection from spray, the NRC staff finds that the licensee has adequately addressed chemical resistance.

The NRC staff also evaluated the proposed changes to determine if the licensee evaluated other environmental parameters such as temperature, pressure, and radiation as required under 10 CFR 50.49(e).

In its supplement, the licensee confirmed that there are no changes to the containment temperature or pressure profiles resulting from the proposed modifications since the ability of the quench spray and recirculation spray subsystems to cool the reactor core and return the containment to subatmospheric pressure and maintain it at subatmospheric pressure is not affected due to the quench spray and recirculation spray flow rates and recirculation spray cooler performance not being affected by the proposed changes. Therefore, the NRC staff finds that environmental qualification of electrical equipment important to safety remains unchanged with respect to temperature and pressure.

With regards to effects of the proposed change on the radiation environment, the licensee stated in the LAR that the revised pH is sufficient to achieve long-term retention of iodine by the containment sump fluid for the purpose of reducing accident-related radiation dose following a LOCA. In addition, in response to the NRC staff's RAI, the licensee stated that the dose rates inside containment using NaTB as a buffer are consistent with the dose rates under the existing configuration using NaOH and the EQ equipment remains qualified for radiation. Furthermore, since the amended TS would achieve a sump pH of 7.0 or greater using NaTB, dose related safety margins would not be significantly reduced. Other parameters such as humidity, aging, synergistic effects, and submergence were evaluated by the licensee and were found to be bounded by the current analysis. The licensee also noted that no changes to the containment analysis was needed as a result of the proposed changes.

Based on its review of the information in the LAR, the supplemental letter, and the NAPS UFSAR for Units 1 and 2, the NRC staff finds that the licensee has sufficiently evaluated the impact of the proposed changes on the EQ of electrical equipment important to safety. Specifically, the licensee's analysis considered the chemical resistance of materials when determining susceptibility to acidic and alkaline sprays, including the corrosion rate for enclosures housing parts of equipment, for the required durations following a LOCA. The licensee also confirmed that other EQ parameters such as temperature, pressure, humidity, radiation, aging, submergence, synergistic effects, and margin will not be affected because of the proposed changes. Therefore, the NRC staff finds that the proposed changes will have no adverse impact on the NAPS, Units 1 and 2, EQ Program (in accordance with NUREG-0588, Revision 1 or IEB 79-01B, as applicable) or its ability to continue to meet the requirements of 10 CFR 50.49.

3.10 NaTB Basket Protection from High Energy Line Breaks

3.10.1 NaTB Basket Protection from High Energy Line Breaks, Description

The licensee proposes the addition of eight NaTB baskets inside containment and cutting and capping the caustic chemical addition piping at the connection to the containment spray pump suction piping located outside of containment. The licensee states that the NaTB baskets are procured as non-safety related and classified as non-safety related with quality requirements and are designed to meet seismic II/I and structural integrity requirements.

Further, the licensee stated in its application:

> The basket locations have been selected such that they are not adversely affected by or adversely affect the Containment sump strainers due to the effects of High Energy Line Break (HELB). Protection against the effects of blowdown jet forces and pipe whip resulting from a postulated pipe rupture of the Reactor Coolant, Pressurizer, Main Steam, or Feedwater System piping is provided by a combination of distance, restraints, and barriers. Specifically, high energy piping is protected/isolated by missile barriers and restrained to limit pipe whip. The baskets located in the containment annulus area are protected by the crane wall. Baskets that are not protected by the crane wall are located so that the impingement pressure from a HELB would not affect the baskets, except for three (3) baskets, such that the ability of the NaTB buffer to perform its design function would not be impeded based on the zone of influence (ZOI) radius. Three (3) baskets located in the Unit 2 Containment are in close proximity to pressurizer spray lines. The portions of these lines do not contain postulated breaks based on the break location criteria outlined in the North Anna Units 1 and 2 UFSAR. Therefore, the baskets are either sufficiently protected from the effects of HELBs using barriers, restraints, and distance, or the lines which are located in close proximity to the baskets are not susceptible to a postulated break.

3.10.2 NaTB Basket Protection from High Energy Line Breaks, NRC Staff Evaluation

The NRC staff reviewed the licensee's evaluation of protection of the NaTB baskets from the effects of HELBs to ensure the baskets would perform their function following a LOCA.

The licensee's break location criteria are discussed in the UFSAR Section 3A.32.2.2, "Present Break Location Criteria." The NRC staff determined that the UFSAR criteria is consistent with the criteria in the NRC's Branch Technical Position (BTP) MEB 3-1, Revision 2, of Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Revision 2 of BTP MEB 3-1 was issued in conjunction with Generic Letter (GL) 87-11, "Relaxation of Arbitrary Intermediate Pipe Rupture Requirements." (Reference 26). GL 87-11 sought to provide relaxation of the requirements for evaluating the dynamic effects (including missile generation, jet impingement forces, and pressures and temperatures) from intermediate pipe locations. The pressurizer spray lines in the vicinity of three (3) baskets are relatively small bore and meet the exclusion criteria of BTP MEB 3-1for considering break dynamic effects.

Based on the above, the NRC staff finds that the NaTB baskets will be sufficiently protected from HELB effects either due to location (distance and shielding) or because piping in the vicinity of the baskets meets the exclusion criteria of BTP MEB 3-1 for considering dynamic effects. Therefore, the NRC staff concludes that the NaTB baskets are adequately protected from the effects of a HELB and therefore satisfy the applicable requirements of GDC 4.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Commonwealth of Virginia official was notified of the proposed issuance of the amendments on July 7, 2023. On July 7, 2023, the state official confirmed that the Commonwealth had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on January 24, 2023 (88 FR 4219). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 REFERENCES

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2 Virginia Electric Electric Power Company (Dominion Energy Virginia), North Anna Power Station, Units 1 and 2 - Response to Request for Additional Information Regarding Proposed License Amendment Request for Removal of Refueling Water Chemical Addition Tank and Replacement of Containment Sump Buffer., April 13, 2023 (ML23103A275).

3 U.S. Nuclear Regulatory Commission, SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program., September 18, 1992 (ML003763736).

4 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ML003716792).

5 U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," Revision 3., March 2007 (ML063600397).

6 U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan, Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Revision 3, December 2016 (ML16088A041).

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12 U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan, Branch Technical Position (BTP) 6-1, "pH for Emergency Coolant Water for Pressurized Water Reactors," March 2007 (ML063190011).

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16 American Society of Civil Engineers, ANSI/ASCE 8-90, "Specification for the Design of Cold-Formed Stainless Steel Structural Members," 1996.

17 Oak Ridge National Laboratory, *The Behavior of Welded Joints in Stainless and Alloy Steels at Elevated Temperatures (ORNL-4781)*., August 1972.

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19 Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission, NUREG/CR-5950 (ORNL/TM-12242), "Iodine Evolution and pH Control," December 1992 (ML083360651).

20 Westinghouse Electric Company, LLC, WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008 (ML081150379).

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23 Virginia Electric and Power Company (Dominion), Dominion Nuclear Connecticut, Inc (DNC) Dominion Energy Kewaunee, Inc (DEK), North Anna and Surry Power Stations, Units 1 and 2, Millstone Power Station, Units 2 and 3, Kewaunee Power Station - Approved Topical Report DOM-NAF-3 NP-A GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment, November 6, 2006 (ML063190467).

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25 Virginia Electric and Power Company, letter to U.S. Nuclear Regulatory Commission (NRC), Virginia Electric and Power Company (Dominion Energy Virginia) North Anna Power Station Units 1 and 2 - Updated Safety Analysis Report - Revision 58, September 29, 2022 (ML22283A023).

26 U.S. Nuclear Regulatory Commission, Generic Letter (GL) 87-11, "Relaxation of Arbitrary Intermediate Pipe Rupture Requirements," June 19, 1987 (ML031150493).

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6.0 CONCLUSION

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

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Date: August 22, 2023

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 295 AND 278 RE: LICENSE AMENDMENT REQUEST TO REMOVE THE REFUELING WATER CHEMICAL ADDITION TANK AND CHANGE THE CONTAINMENT SUMP PH BUFFER (EPID L-2022-LLA-0164) DATED AUGUST 22, 2023

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ADAMS Accession No.: ML23181A135 ***Via SE Input**

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DATE	6/30/2023	7/07/2023	5/30/2023	7/11/2023
OFFICE	NRR/DEX/ESEB/BC	NRR/DEX/ELTB/BC	NRR.DEX/EMIB/BC	NRR/DNRL/NCSG/BC
NAME	ITseng*	JPaige*	SBailey*	SBloom*
DATE	6/13/2023	7/06/2023	6/20/2023	6/13/2023
OFFICE	NRR/DRA/ARCB/BC	NRR/DSS/STSB/BC	OGC - NLO	NRR/DORL/LPL2-1/BC
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