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1CAN102201 10 CFR 50.90

October 13, 2022

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Supplement to Proposed Technical Specifications 3.4.12 and 3.4.13 Revised Dose Calculations

Arkansas Nuclear One, Unit 1
NRC Docket No. 50-313
Renewed Facility Operating License No. DPR-51

References:

1) Entergy letter to NRC, "License Amendment Request Proposed Technical Specifications 3.4.12 and 3.4.13 Revised Dose Calculations," (1CAN092101), (ML21274A874), dated September 30, 2021

2) Entergy letter to NRC, "Response to the Request for Additional Information Proposed Technical Specifications 3.4.12 and 3.4.13 Revised Dose Calculations," (1CAN062201) (ML22153A464, ML22153A465), dated June 2, 2022

Entergy Operations, Inc. (Entergy) requested the U.S. Nuclear Regulatory Commission (NRC) to approve a license amendment for Arkansas Nuclear One, Unit 1 (ANO-1) via Reference 1. The requested amendment would revise the Dose Equivalent Iodine (I)-131 and the reactor coolant system (RCS) primary activity limits required by Technical Specification (TS) 3.4.12, "RCS Specific Activity". In addition, a new primary-to-secondary leak rate limit, provided in TS 3.4.13, "RCS Leakage," is being proposed. These proposed changes are due to non-conservative inputs used in the Steam Generator Tube Rupture (SGTR) accident, the Main Steam Line Break (MSLB) accident, and the Control Rod Ejection accident dose calculations.

During the review of the material submitted in Reference 1, the NRC and Entergy determined that some of the information provided should have been marked as proprietary. The purpose of

This letter contains ~~proprietary information.~~
~~Withhold Enclosure 1 from public disclosure in accordance with 10 CFR 2.390.~~

this letter is to update the information provided in Reference 1 Enclosure to contain the necessary proprietary markings. No changes were required to the Technical Specification markup or retyped pages provided in Reference 1 and as such are not included as part of this letter.

It should be noted that the text, including the No Significant Hazards Consideration Analysis, has not been revised, modified, or deleted from the reference submittal.

Please use this supplemental information in your review of the request.

This has been discussed with the NRR Project Manager for ANO-1. It has also been entered into Entergy's Corrective Action Program.

Some information provided in Enclosure 1 is considered proprietary to MPR Associates, Inc. (320 King Street, Alexandria, VA 22314), who requests it to be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. The proprietary information is identified by text enclosed within double brackets **[[**Example**]]**. The non-proprietary version is provided in Enclosure 2.

This information is supported by an affidavit, signed by John W. Simons, Vice President of Power Services at MPR Associates, Inc., the owner of the information. The affidavit sets forth the basis by which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (a)(4) of 10 CFR 2.390 of the Commission's regulations. The affidavit is included in Enclosure 3.

There are no new regulatory commitments established in this submittal.

If there are any questions or if additional information is needed, please contact Riley Keele, Manager, Regulatory Assurance, Arkansas Nuclear One, at 479-858-7826.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on October 13, 2022.

Respectfully,

Philip Couture

Digitally signed by Philip Couture
Date: 2022.10.13 07:53:38 -05'00'

Phil Couture

PC/rwc

Enclosure: 1. Evaluation of Proposed Change (PROPRIETARY)

2. Evaluation of Proposed Change (NON-PROPRIETARY)

3. Affidavit from MPR Associates, Inc.

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector – Arkansas Nuclear One
NRC Project Manager – Arkansas Nuclear One
Designated Arkansas State Official

ENCLOSURE 2

1CAN102201

EVALUATION OF THE PROPOSED CHANGE

(NON-PROPRIETARY)

EVALUATION OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

The proposed amendment to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS) would revise the DOSE EQUIVALENT I-131 and the Reactor Coolant System (RCS) primary activity limits required by TS 3.4.12, "RCS Specific Activity". In addition, the primary-to-secondary leak rate limit provided in TS 3.4.13, "RCS Leakage," is being revised. The proposed changes would be more restrictive and reduce the allowed RCS specific activities and the allowed primary-to-secondary leak rate. The requested changes do not involve a significant hazards consideration.

The proposed changes are due to revising the non-conservative licensing basis dose consequences for some design basis accidents. Specifically, the Steam Generator Tube Rupture (SGTR) accident; the Main Steam Line Break (MSLB) accident; and the Control Rod Ejection (CRE) accident. All other design basis dose consequences are not impacted by the proposed changes. The current bases for the previously listed licensing basis accidents use the alternative source term (AST) methodology as allowed by 10 CFR 50.67, "Accident source term," and described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" to determine dose consequences.

The proprietary information is identified by text enclosed within double brackets [[Example]].

2.0 DETAILED DESCRIPTION

2.1 <u>Current Licensing Basis and Technical Specification Requirements</u>

2.1.1 <u>Current Licensing Basis</u>

Entergy Operations, Inc., (Entergy) has implemented the Alternative Source Term (AST) methodology into the current ANO-1 licensing bases (Reference 1).

The proposed changes to the ANO-1 TS are due to non-conservative inputs used in the dose consequence determination for the following design basis accidents:

- SGTR
- MSLB
- CRE

The proposed changes are more restrictive and reduce the allowed RCS specific activities. These changes will also lower the allowed primary-to-secondary leak rate.

2.1.2 Current Technical Specification Requirements

The limiting condition for operation (LCO) for TS 3.4.12 is that RCS DOSE EQUIVALENT I-131 specific activity shall be within limits during Modes 1, 2, 3 and 4. Action A requires that the DOSE EQUIVALENT I-131 be verified to be less than or equal to 60 micro curies per gram (μCi/gm) every four hours and restore DOSE EQUIVALENT I-131 to within limit in 48 hours. Action C is if Required Action and associated Completion Time of Condition A or B not met OR DOSE EQUIVALENT I-131 greater than 60 μCi/gm, be in Mode 3 in six hours and Mode 5 in 36 hours.

Surveillance Requirement (SR) 3.4.12.2 verifies that the reactor coolant DOSE EQUIVALENT I-131 specific activity is less than or equal to 1.0 μCi/gm on a frequency in accordance with the Surveillance Frequency Control Program.

The LCO for TS 3.4.13 limits the RCS operational leakage to 150 gallons per day (gpd) primary to secondary leakage through any one Steam Generator (SG) in Modes 1, 2, 3 or 4. If the primary to secondary leakage is not within limits, Action B requires the unit to be in Mode 3 in six hours and in Mode 5 within 36 hours.

SR 3.4.13.2 verifies the primary-to-secondary leakage is less than or equal to 150 gpd through any one SG on a frequency in accordance with the Surveillance Frequency Control Program.

2.2 Reason for the Proposed Changes

In 2007, the technical information to support a transition in the model for the ANO-1 SGTR radiological analysis to an Alternate Source Term (AST) methodology was developed. The NRC issued Amendment 238 to the ANO-1 Operating License via Reference 1. The amendment modified requirements of TS 3.4.12, "RCS [reactor coolant system] Specific Activity," as related to the use of an alternate source term (AST) associated with accident offsite and control room dose consequences.

In October 2018, it was identified the ANO-1 SGTR off-site dose evaluation was non-conservative because it failed to consider post-reactor trip High Pressure Injection (HPI) flows during a SGTR event. HPI flows would cause RCS pressure to remain at a higher value, which in turn would result in a higher primary to secondary leak rate value until the assumed operator action occurs to reduce the HPI flow. Due to the higher RCS leakage into the secondary side of the SG, a potential for higher release and subsequent dose to the public became a concern.

As a result of this issue, Entergy determined the ANO-1 SGs were Operable with compensatory measures for this condition and initiated administrative controls to limit potential radioactive releases that could impact offsite dose. Controls employed were administrative limits to reduce the primary and secondary activity levels to 50% of the TS values while the subsequent analyses were completed.

As the thermal-hydraulic analysis to address the above condition was revised, Entergy became aware in July 2020 that certain assumptions made for the SGTR radiological analysis did not appear to have sufficient justification. Specifically, the "flashing fraction," or the amount of RCS leakage that flashes to vapor when it enters the secondary side of the steam generator. The original analyses assumed a flashing fraction of 15% which is potentially non-conservative for Once-Through Steam Generators (OTSGs) used in Babcock and Wilcox (B&W) units like ANO-1.

It was subsequently identified that the potentially non-conservative flashing fraction was used in the ANO-1 SGTR, Main Steam Line Break (MSLB), and Control Rod Ejection (CRE) safety analyses. The safety analyses for these events are described in Chapter 14 of the ANO-1 Safety Analysis Report.

Based on the above information, Entergy further reduced the ANO-1 administrative controls implemented in 2018 to 10% of the allowed TS values and instituted a reduction in the allowed steam generator leakage from 150 gallons per day (gpd) to 30 gpd. These revised administrative limits were based on the revised thermal-hydraulic analysis and a conservative limiting flashing fraction of 100%.

Entergy believed that the 100% flashing fraction was too conservative. The analyses needed to resolve the flashing fraction were performed and a summary of these analyses is provided below.

3.0. TECHNICAL EVALUATION

The dose consequence analyses for the SGTR, MSLB, and CRE events were performed. RADTRAD (References 2, 3, and 4) was used in the dose consequence analyses. The analyses maintained the previous release points and the offsite and Control Room atmospheric dispersion factors and other inputs approved in Reference 1. The loss-of-offsite-power (LOOP) assumption was maintained.

The source terms, SGTR thermal-hydraulic model, and flashing fraction analyses were revised. Details of these changes are provided below.

3.1 Source Term

Many radiological events do not involve the failure of the fuel cladding such that the only released source terms are those associated with the reactor coolant inventory and secondary side in Pressurized Water Reactors (PWRs). Based on RG 1.183 (Reference 5), these events include:

- PWR Main Steam Line Break
- PWR Steam Generator Tube Rupture

The nuclide groups in the reactor coolant are expected to be those in the gap of the fuel rods since fuel leakage is generally the primary source for reactor coolant activity. Based on the gap release fraction in RG 1.183, these isotope groups would be the noble gases (e.g., Krypton and Xenon), halogens (e.g., Iodine), and alkali metals (e.g., Cesium). Regulatory Issue Summary 2006-04 (Reference 6) Item 9 notes that these initial activities should include noble gases and Cesium isotopes for AST analyses. Activated corrosion products (e.g., Cobalt-60) or coolant isotopes (e.g., Nitrogen-16) are generally not considered in the design basis radiological analyses since they would be negligible contributors to the ultimate doses.

The applicable regulatory guidance for design basis accidents does not provide an approved methodology for calculating reactor coolant sources. Although Section 3.1 of RG 1.183 (Reference 5) endorses the ORIGEN methodology for core inventory calculations, it does not endorse or mention any specific methodology associated with the reactor coolant source terms.

The most applicable regulatory reference addressing reactor coolant sources is RG 1.112 Revision 1 (Reference 7) which describes acceptable methods for the calculation of gaseous and liquid radioactive effluents. This document endorses the application of ANSI/ANS-18.1-1999 for the calculation of reactor coolant sources as part of the calculation of plant effluents.

The NRC's inclusion of the ANSI/ANS-18.1 standard in the recent updates to the BWR and PWR GALE codes (References 8 and 9) further demonstrates the NRC recognizes this methodology for the calculation of downstream effects (i.e., plant effluents) from the reactor coolant system (RCS). In addition, Reference 10 addresses acceptable models for developing reactor coolant source terms for effluent releases states:

> In general, reactor coolant and steam source terms used as the design basis for expected releases have been found acceptable if these values are determined using models and parameters that are consistent with NRC and industry guidance. The guidance includes RG 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," NUREG-0016 and NUREG-0017 as the basis of the BWR gaseous and liquid effluent (GALE) and PWR-GALE codes, and industry guidance provided in ANSI/ANS 18.1-1999, as adjusted to reflect specific design features.

The reactor coolant source terms for the ANO-1 primary and secondary sides using the methodology in ANSI/ANS-18.1-2020 were developed. This methodology produces the normal reactor coolant source terms including a standardized set of isotopes and the relative magnitudes among the isotopes. Then, these inventories are calculated at the applicable limits in the ANO-1 TSs for application in the design basis radiological analyses. Table 1 provides the results of the ANSI/ANS-18.1-2020 methodology for ANO-1.

ANO-1 Reactor Coolant Activities (μCi/g)

Isotope	Dose Equivalent I-131		Dose Equivalent Xe-133
	1.0	0.1	2200
	Primary	Secondary	Primary
Kr-85m			3.51E+00
Kr-85			1.45E+02
Kr-87			5.98E+00
Kr-88			7.01E+00
Xe-131m			1.61E+03
Xe-133m			3.14E+00
Xe-133			7.90E+01
Xe-135m			1.50E+01
Xe-135			2.69E+01
Xe-137			7.00E+01
Xe-138			1.81E+01
Br-84	3.85E+00	1.23E+00	
I-131	5.65E-01	5.61E-02	
I-132	2.00E+00	1.99E-01	
I-133	1.74E+00	1.75E-01	
I-134	2.93E+00	3.06E-01	
I-135	2.45E+00	2.51E-01	
Rb-88	1.49E-01	2.63E-02	
Cs-134	2.95E-01	5.33E-02	

Isotope	Dose Equivalent I-131		Dose Equivalent Xe-133
	1.0	0.1	2200
	Primary	Secondary	Primary
Cs-135	4.52E-02	8.23E-03	
Cs-137	1.88E-01	2.29E-02	

The source terms, listed above, are used in the revised dose consequence analyses for the SGTR and MSLB events. The source terms used in the CRE dose consequence analysis is the same as those approved in Reference 1.

3.2 Thermal-Hydraulic Model

While operating at full power, the double-ended rupture of a SG tube occurs with unrestricted discharge from each end. The initial leak rate exceeds the normal makeup (MU) to the RCS and system pressure decreases. No initial operator action is assumed, and the primary system pressure decreases until a reactor trip occurs on low reactor coolant pressure. The main turbine trips because of the reactor trip.

For the LOOP scenario, offsite power is assumed to be lost coincident with reactor trip. Consequently, the reactor coolant pumps (RCPs) trip, the main feedwater (MFW) pumps trip, and the condenser becomes unavailable. Emergency feedwater (EFW) actuates automatically to raise SG liquid levels to 50 percent on the operate range (natural circulation setpoint). It is conservatively assumed that the steam supply to the turbine driven EFW (TDEFW) pump originates from the SG with the tube rupture and exhausts directly to the atmosphere. Closure of the turbine stop valves (TSVs) causes the main steam line pressure to increase and open the atmospheric dump valves (ADVs) and main steam safety valves (MSSVs).

The fission products escaping from the ADVs and/or MSSVs are released directly to the atmosphere. The operator performs normal post-trip verifications and then begins a cooldown of the RCS using the ADV(s) in manual mode. After the RCS temperature has decreased to a value that corresponds to a saturation pressure which is below the lowest MSSV setpoint, the affected SG can be isolated. As a result of this analysis, the time for this action has been revised from 34 minutes after the initiation of the event to 70 minutes. The RCS cooldown continues by steaming the unaffected SG. The RCS pressure is maintained near the minimum adequate subcooling margin (SCM) by throttling high pressure safety injection (which started automatically on a low reactor coolant pressure signal or was started manually by the operator) and cycling the electromatic relief valve (ERV).

Prior to reactor trip, all primary-to-secondary leakage, as well as the ruptured tube flow, is directed to the condenser, where it is partitioned prior to release. Primary to secondary SG leakage to the unaffected SG is assumed to be at the Technical Specification (TS) maximum of 150 gallons per day (gpd) throughout the event. The LOOP results in a loss of the condenser causing the MSSVs to open and provide steam relief. Steam release via the MSSVs and ADVs (during operator-controlled RCS cooldown) continues until the affected SG is isolated. At that time, the release point then becomes the ADV for the unaffected SG. The release continues until decay heat removal is initiated and the unaffected SG is isolated, thus terminating the steam release.

The analysis credits the non-safety main steam line ADVs to cool down the plant. The worst single failure that was determined for this event is the ADV on the unaffected SG failing closed following the reactor trip. This results in only the affected SG being available for cooling down the plant until operator action is taken to manually open the ADV. The following operator actions are credited during this event:

- Operator uses ERV cycling to control core exit SCM.
- Operator uses high pressure injection (HPI) throttling to control pressurizer level.
- Operator attempts to take remote manual control of the ADVs on both SGs to initiate RCS cooldown.
- Operator identifies the failed-closed position of the ADV on the unaffected SG.
- Operator performs local manual opening of the ADV on the unaffected SG.
- Operator continues RCS cooldown using both SGs.
- Operator isolates the affected SG when conditions allow (ADV closed, EFW secured).

The RELAP5/MOD2-B&W computer code (Reference 11) is used to perform the SGTR thermal-hydraulics (T-H) transient analysis. RELAP5/MOD2-B&W is approved by the NRC for use in both loss of coolant accident (LOCA) and non-LOCA safety analyses on B&W-designed PWRs. The code allows the modeling of both the primary and secondary systems as well as fill systems, such as high-pressure injection and emergency feedwater. No code interfaces are required for the SGTR T-H transient analysis.

The SGTR T-H transient analyses are performed in accordance with the NRC approved non-LOCA methodology in Reference 11. Where possible, this methodology utilizes the plant design bases to establish acceptance criteria and input boundary conditions. The approved methodology includes the manner for determining the responses to postulated accidents for the primary system, secondary systems, and the core. In addition, the methodology requires the use of conservative setpoints, valve and pump capacities, and reactivity coefficients to demonstrate adequate margin to the applicable limits. The double-ended rupture of a single SG tube, at the lower face of the upper tubesheet, is modeled to occur at full power conditions.

The leak rate is calculated using the momentum equation with friction and form losses as appropriate. Credit is taken for critical flow, when predicted, using critical flow models approved for use in the small break evaluation model.

The SGTR LOOP scenario includes a limiting single failure that maximizes the activity releases. The single failure to be assumed is determined by reviewing the safety actions that are required by various systems for the mitigation of the SGTR event and then postulating the loss of the safety action that has the potential to have the greatest adverse impact on event progression.

Several safety-grade and non-safety-grade components and systems are required to mitigate a SGTR with a coincident LOOP. A failure of one component required to mitigate the event is assumed.

The limiting single failure with respect to activity releases during SGTR with coincident LOOP is determined to be the failure of the ADV on the unaffected SG to open following reactor trip and LOOP. This would require dispatch of an operator to manually open the valve. Meanwhile, the majority of decay heat and RCS sensible heat is removed by steaming the affected SG to the atmosphere. This failure provides the latest isolation of the affected SG as compared with the other ADV failure scenarios. Furthermore, this failure provides for extensive steam release from the affected SG. These combined effects make this the limiting single failure for SGTR with LOOP.

The MSLB and CREA events are not impacted by the revised thermal-hydraulic model changes.

3.3 Flashing Fraction Determination

During the postulated SGTR event, high energy fluid from the primary side (i.e., inside the tube) discharges into the OTSG secondary side. Steam is formed due to phase separation during initial depressurization as well as heat transfer to the remaining liquid portion of the rupture flow. The amount of steam generated consists of the following:

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The steam converted from the primary coolant has implications for offsite dose consequences, since it can carry off radioactive materials that are present in the primary coolant. During a SGTR event, steam release through the MSSVs and/or ADVs in the main steam line will result in an offsite release.

Flashing fraction is used to quantify the amount of steam generated from the rupture flow during a SGTR event. The flashing fraction is defined as the ratio between the mass of generated steam and total mass of the rupture flow that escapes from the primary loop into the SG secondary side. The flashing fraction also equals the ratio between the mass rate of steam generation and total break mass flowrate.

A best estimate analysis approach was developed with a conservative bias to determine the flashing fraction during a SGTR event. Appropriate assumptions were used in the model to ensure the conservatism of the calculated flashing fraction.

An iterative numerical solution was developed to simultaneously solve the appropriate equations, using the inputs and property definitions discussed below. The steam flashing fractions were calculated for all times during the postulated SGTR event, consistent with the limiting tube rupture scenario analyzed in the thermal-hydraulics model.

The SGTR was modeled as a double-ended guillotine break at the top of the secondary side of SG just below the lower face of the upper tubesheet. During the postulated SGTR event, high energy fluid from the primary side (i.e., inside the tube) discharges into the SG secondary side. Steam is formed due to phase separation during initial depressurization as well as heat transfer to the remaining liquid portion of the rupture flow.

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This analysis uses the results of the T-H analyses for the SGTR scenario with a coincidental loss-of-offsite-power and delayed reactor trip.

The rupture flow that reaches the water collected near the bottom of the SG is assumed to not have a radiological consequence. The radioactive isotopes will remain in the bulk water and not be carried into the SG steam flow during the SGTR event. Therefore, evaporation of water from rupture flow after it reaches and mixes with the bulk water collected near the bottom of the SG is not modeled.

3.3.1 Inputs

The inputs to this evaluation include:

- Basic inputs for SG (i.e., number of tubes, tube material, total tube length, tube inner and outer diameters, number of plugged tubes)
- Thermal conductivity of tube material (Alloy 690)
- Properties of water and steam (i.e., saturation temperature, enthalpy of saturated water, enthalpy of evaporation, density of saturated water, density of saturated steam, specific heat of saturated water, specific heat of saturated steam, thermal conductivity of saturated water, dynamic viscosity of saturated water, Prandtl number of saturated water, and surface tension of saturated water)

3.3.2 SG conditions from T-H analysis

A computational fluid mechanics model was developed to determine the number of wetted tubes upon which the liquid portion of the break flow could be splashed after initial phase separation. The wetted tubes were found to consist of the following:

- 650 wetted tubes above the top tube support plate, with a span of 46.32 inches,
- 180 wetted tubes below the top tube support plate, with a span that extends until the collapsed water level near the bottom of the SG.

Details of this evaluation are provided in Section 3.3.6 of this enclosure.

3.3.3 Break flow conditions

The break flow conditions for a double-ended guillotine break were determined in the T-H model discussed previously, consisting of the primary fluid coming out of the rupture location from both the hot leg and cold leg sides, i.e., break flows from top and bottom sides of the ruptured tube, respectively.

The total break flow rate is determined by combining the top and bottom flows obtained from the T-H model discussed above. The temperature of the break flow is calculated using the weighted average of the hot leg and cold leg temperatures along with the corresponding break flows from the top and bottom sides of a ruptured tube.

3.3.4. Phase separation due to depressurization

Flashing occurs as the break flow exiting the ruptured tube enters the SG secondary side, resulting in phase separation caused by depressurization in the SG secondary side. The break flow separates into steam and liquid phases, both at the saturation temperature corresponding to the SG secondary pressure. The initial flashing fraction of the rupture flow and the steam generation rate due to initial phase separation are calculated based on the conservation of mass and energy (enthalpy).

3.3.5 Evaporation due to heat transfer from SG tubes

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Comparison between Total Break Flow and Steam Generation Rates

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The flashing fraction as a function of time is shown in the next figure.

Flashing Fraction during the SGTR Event

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3.3.7 Determination of wetted tubes

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3.4 DOSE CALCULATIONS

Using the appropriate revised inputs discussed above and retaining all the other inputs from the previous dose calculations, new dose results and proposed values for the revised TSs were performed. Each of the revised calculations is summarized below.

3.4.1 SGTR

In addition to the pre-scram condenser release, the SGTR release path is to the atmosphere outside of the containment through either the MSSVs or the ADVs. The SGTR dose analysis considers both the possibility of a Pre-Existing (PE) spike in the reactor coolant Iodine concentration or of an Accident-Initiated (AI) Iodine spike.

Total Effective Dose Equivalent (TEDE) doses are determined for offsite individuals and for the operators in the Control Room using the guidance of RG 1.183. The methods and guidance described in RG 1.183 have been applied to the dose analysis of the SGTR accident.

3.4.1.1 Sequence of events

The SGTR scenario considers a single active failure and a coincident loss-of-offsite power. The accident chronology applied in this analysis is below.

Time	Event
0 seconds	SGTR occurs
9.6 minutes	Reactor trip on low RCS pressure with consequential loss-of-offsite-power. ADV block valve on intact SG fails to open. All releases are through MSSVs.
9.6 minutes + 10 seconds	Control Room isolated due to High Radiation in the air intakes.
30 minutes	Operators begin plant cooldown initially using the intact SG. Operators note the ADV on intact SG fails to open. Operator dispatched to take manual control of ADV. Cooldown continues using the ruptured SG.
60 minutes	Failed ADV is manually opened.

70 minutes	Ruptured SG isolated. Leakage from the ruptured tube and the primary-to-secondary leak path of the ruptured SG are no longer released to the atmosphere. RCS temperature decreased to a value that corresponds to a saturation pressure which is below the lowest setpoint for the MSSVs.
237.8 hours	Decay heat removal system initiation conditions are reached. The primary-to-secondary leak path from the intact SG is terminated.

This analysis uses the new inputs discussed above. It should be noted that the CFD analysis indicates that the vaporization fraction decreases quickly after the scram. This analysis applies a 40% vaporization fraction at 4.4 minutes after the reactor trip, which is 14 minutes after the SGTR occurs.

3.4.1.2 <u>Results</u>

The radiological analysis is based on six different cases representing different source term pathways. An additional case for each pathway is also prepared to address the impact of cloud shine on the control room dose. These cases are summarized below.

1. Secondary coolant iodine release from both steam generators
2. Secondary coolant alkali metal release from both steam generators
3. Primary coolant iodine release due to P/S and tube leakage
4. Primary coolant alkali metal release due to P/S and tube leakage
5. Primary coolant noble gas release due to P/S and tube leakage
6. Primary coolant iodine release from accident induced spike due to P/S and tube leakage

The radiological doses from each of these cases are analyzed with the RADTRAD 3.03 methodology. The results from all applicable cases are then summed for a total SGTR dose.

Dose Results (Rem TEDE)

Event	Initial DEI	Exclusion Area Boundary (EAB)	Low Population Zone (LPZ)	Control Room (CR)
Pre-existing spike	6	1.81	0.29	**4.92**
RG 1.183 Acceptance Criteria		25	25	5
Accident-induced spike	0.25	**2.45**	0.41	2.76
RG 1.183 Acceptance Criteria		2.5	2.5	5

3.4.2 MSLB

The MSLB can occur either inside the containment or outside the containment. The accident analysis models the event occurring outside the containment since a MSLB occurring inside the containment would have releases limited by containment isolation. The MSLB dose analysis also needs to consider the possibility of a Pre-Existing (PE) spike in the reactor coolant iodine concentration or an Accident Initiated (AI) iodine spike.

TEDE doses are determined for individuals at offsite locations and for the operators in the Control Room using the guidance of RG 1.183. The Control Room operator dose calculation includes both the dose from the activity entering the Control Room and the dose from the cloud of activity outside the Control Room.

3.4.2.1 Sequence of events

The MSLB scenario considers a single active failure and a coincident loss-of-offsite power. The accident chronology applied in this analysis is below.

Time	Event
0 seconds	MSLB occurs outside containment.
~0 seconds	Reactor trip with consequential loss-of-offsite-power. ADV block valve on intact loop fails to open. All releases are through MSSVs.
10 seconds	Control Room isolated.

1 minute	Faulted SG secondary side completely released. Main Steam Isolation Valves (MSIVs) are closed. Operators note the ADV on intact SG fails to open. Operator dispatched to take manual control of ADV.
30 minutes	Failed ADV is manually opened. Operators begin plant cooldown using the intact loop.
237.8 hours	Decay heat removal system initiation conditions are reached. The primary-to-secondary leak path from the intact loop is terminated.
251.8 hours	RCS temperature is below 212 ºF. The primary-to-secondary leak path from the faulted loop is terminated.

3.4.2.2 Results

The radiological analysis is based on 9 different cases representing different source term pathways. An additional case for each pathway is also prepared to address the impact of cloud shine on the control room dose. These cases are summarized below.

1. Secondary coolant iodine release from faulted SG
2. Secondary coolant iodine release from intact SG
3. Secondary coolant alkali metal release from faulted SG
4. Secondary coolant alkali metal release from intact SG
5. Primary coolant iodine release due to primary-to-secondary leakage
6. Primary coolant alkali metal release due to primary-to-secondary leakage
7. Primary coolant noble gas release due to primary-to-secondary leakage
8. Primary coolant iodine release from pre-existing spike due to primary-to-secondary leakage from faulted SG
9. Primary coolant iodine release from accident-induced spike due to primary-to-secondary leakage from faulted SG

The radiological doses from each of these cases are analyzed with the RADTRAD 3.03 methodology. The results from all applicable cases are then summed for a total MSLB dose.

Total Dose Results (Rem TEDE)

Event	Initial DEI	EAB	LPZ	Control Room
Pre-existing spike	60	0.32	0.18	1.08
RG 1.183 Acceptance Criteria		25	25	5
Accident-induced spike	1	1.26	0.72	2.55
RG 1.183 Acceptance Criteria		2.5	2.5	5

3.4.3 CRE

A CRE is the rapid ejection of a single rod from the core region during operation. For the postulated CRE, a mechanical failure of a control rod drive mechanism (CRDM) housing or the CRDM nozzle is assumed such that the pressure differential, RCS pressure to reactor building atmospheric pressure, acting on the control rod would eject the control rod and drive shaft to the fully withdrawn position.

The ejection of the control rod causes positive reactivity to be inserted into the core, equivalent to the worth of the ejected rod, increasing the local power and fuel temperatures. The increasing fuel temperatures cause negative reactivity to be inserted into the core due to the mitigating effects of the negative Doppler fuel temperature coefficient. As the power increases, the high neutron flux (or high pressure for some of the hot zero power (HZP) cases) setpoint is reached and initiates a reactor trip, which drops the remaining control rods. The additional negative reactivity inserted by the control rods decreases the power further and the power transient is terminated.

This analysis calculates the CRE doses from the two pathways described in RG 1.183:

- Containment leakage
- Primary-to-secondary leakage

Specific assumptions for the containment leak path:

1. The source terms are released from the core and homogenously mixed into the containment in 10 seconds.

2. The containment volume is assumed to be 1.81E6 ft^3, consistent with the LOCA analysis. The only credited containment removal mechanism is aerosol deposition. Containment spray or filtration via the Penetration Room Ventilation System (PRVS) is conservatively ignored. A bounding removal rate of 0.1 hr^{-1} is applied to the aerosol activity for the first 69 hours for a DF of 1000. This is consistent with the approved ANO-1 AST Safety Evaluation.

3. Containment leakage rate is identical to that applied in the LOCA analysis and consistent with ANO-1 TS 5.5.16. A leakage rate of 0.2%/day is assumed for the first 24 hours and 0.1%/day for the following 29 days consistent with the guidance in Section 6.2 to Appendix H to RG 1.183.

Assumptions used for the primary-to-secondary leakage pathway include:

1. The source terms are released from the core and homogenously mixed into the RCS in 10 seconds. Since the fuel failures are associated with Departure from Nucleate Boiling only, this time is expected to be much shorter than that expected for the assumed failure of the cladding, release of the gap activity, and transport throughout the RCS with the RCPs tripped due to the LOOP.

2. The RCS is cooled down with both loops and shutdown cooling is initiated at 38.25 hours.

3.4.3.1 Sequence of events

The CRE radiological analysis considers a single active failure and a coincident LOOP. The accident chronology applied in this analysis is reported below.

Containment Leakage Pathway

Time	Event
0 seconds	CRE and LOOP occurs. Reactor trip on high neutron flux. Source term release begins.
10 seconds	Control Room isolated on high radiation in intakes. Source term release into containment ends.
30 days	Releases are terminated.

Primary-to-secondary Leakage Pathway

Time	Event
0 seconds	CRE and LOOP occurs. Reactor trip on high neutron flux. Source term release begins. ADV block valve on one SG fails to open. All releases are through MSSVs for the affected loop.
10 seconds	Control Room isolated on high radiation in intakes. Source term release into coolant ends.
30 minutes	Failed ADV is manually opened. Releases from the affected loop via ADV.
38.25 hours	Decay heat removal system initiated. Releases terminated.

3.4.3.2 Results

The radiological analysis will be based on 2 different cases representing the different source term pathways. An additional case for each pathway is also prepared to address the impact of cloud shine on the control room dose. These cases are summarized below.

1. Containment leakage pathway
2. Primary-to-secondary leakage pathway

The radiological doses from each of these cases are analyzed with the RADTRAD 3.03 methodology. The results from all applicable cases are then summed for a total MSLB dose.

Dose Results (Rem TEDE)

Event	Primary-to-secondary leak rate (gpd/SG)	EAB	LPZ	Control Room
Containment pathway	N/A	4.72	2.27	3.12
Primary-to-secondary pathway	39	3.29	2.20	4.96
RG 1.183 Acceptance Criteria		6.3	6.3	5

3.5 Conclusions

Based upon the results of the analyses described above, the proposed new TS limits are:

- Dose Equivalent I-131 — 6 μCi/gm (TS 3.4.12, Conditions A.1 and C)
- Dose Equivalent I-131 — 0.25 μCi/gm (SR 3.4.12.2)
- Primary-to-secondary leak rate — 39 gpd/SG (TS LCO 3.4.13.d; SR 3.4.13.2)

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Entergy proposes to revise the dose equivalent I-131 and primary-to-secondary leak rate limits based on revising the previously approved dose calculations using the requirements of 10 CFR 50.67 and the guidance in RG 1.183. These calculations were revised due to non-conservative inputs. The revised limits provide results that meet the acceptance criteria listed in RG 1.183.

The regulatory requirements are the reference values in 10 CFR 50.67, "Accident source term," and the accident-specific guideline values in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ADAMS Accession No. ML003716792), and Table 1, "Accident Dose Criteria," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 15.0.1, Radiological Consequence Analyses Using Alternative Source Terms" (ADAMS Accession No. ML003734190). There is no proposed significant deviation or departure from the guidance provided in RG 1.183 in the current evaluations. The evaluation is based upon the following regulations, regulatory guides, and standards:

10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires that the safety-related electrical equipment which are relied upon to remain functional during and following design basis events be qualified for accident

(harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires that preventative maintenance activities must not reduce the overall availability of the systems, structures, or components.

10 CFR Section 50.67, "Accident source term," as described above.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

10 CFR Part 50, Appendix A, GDC 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

10 CFR Part 50, Appendix A, GDC 19, "Control room," requires that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and to maintain the reactor in a safe condition under accident conditions, including a LOCA. With regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE), as defined in 10 CFR 50.2 for the duration of the accident. 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

NRC RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 3, June 2001 (ADAMS Accession No. ML011710176).

NRC RG 1.75, "Criteria for Independence of Electrical Safety Systems," Revision 3, dated February 2005 (ADAMS Accession No. ML043630448) describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000 (ADAMS Accession No. ML003716792) provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. This RG states that licensees may use either the AST or the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962 (ADAMS Legacy Library Accession No. 8202010067), assumptions for performing the required environmental qualification analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID-14844) on environmental qualification doses. RG 1.183 also states that maintaining pH basic will minimize re-evolution of iodine from the suppression pool water.

NRC RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0, June 2003 (ADAMS Accession No. ML031530505)

NRC RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 0, May 2003 (ADAMS Accession No. ML031490611)

Industry standards and NRC guidance documents were also considered with respect to the proposed change including:

NRC NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985

NRC NUREG-0800, Revision 2, "Standard Review Plan," Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Revision 3, March 2007 (ADAMS Accession No. ML070730398)

NRC NUREG-0800, Revision 2, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Revision 3, March 2007 (ADAMS Accession No. ML070550069)

NRC NUREG-0800, Revision 2, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007 (ADAMS Accession No. ML070190178)

NRC NUREG-0800, Revision 2, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190)

NRC NUREG-0800, Revision 2, "Standard Review Plan," Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, July 1981 (ADAMS Accession No. ML052350147)

NRC NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995 (ADAMS Accession No. ML0410400063)

NRC NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992 (ADAMS Accession No. ML063460464)

NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

NRC NUREG/CR 5732, "Iodine Chemical Forms in LWR Severe Accidents," contains information with respect to the benefit of RB sump pH control during accident conditions

NRC NUREG/CR-5950, "Iodine Evolution and pH Control," also contains information with respect to the benefit of RB sump pH control during accident conditions

NRC IE Bulletin No. 79-01B, "Environmental Qualification of Class 1E Equipment," contains information intended to support equipment functionality under postulated accident conditions.

Note that ANO-1 was not licensed to the 10 CFR 50, Appendix A, GDC. ANO-1 was originally designed to comply with the 70 "Proposed General Design Criteria for Nuclear Power Plant Construction Permits," published in July 1967. However, the ANO-1 SAR provides a comparison with the AEC GDC published as Appendix A to 10 CFR 50 in 1971.

4.2 Precedent

ANO-1 is one of several plants to be approved to use AST (Reference 1). A license amendment request (LAR) to revise the primary and secondary coolant activity Technical Specifications for Beaver Valley was approved by the NRC (Reference 13). These revisions were due to non-conservative inputs to the dose calculations using AST.

4.3 No Significant Hazards Consideration Analysis

Entergy Operations, Inc. (Entergy) requests a revision to Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specification (TS) and dose consequences for the SGTR, MSLB, and CRE design basis accidents licensing bases. The proposed changes would be more restrictive and reduce the primary and secondary specific activity and the primary-to-secondary leak rate for each SG.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed amendment re-analyzes EAB, LPZ, and CR doses for three design basis accidents to address non-conservative inputs previously used. There are no plant modifications or operating procedure changes that would increase the probability of an accident previously evaluated. While the revised doses generally increase, they remain below the allowable regulatory limits.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   The proposed amendment changes accident analysis inputs for calculating dose consequences at the EAB, LPZ, and CR. There are no plant modifications or operating procedure changes.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The proposed amendment re-analyzes EAB, LPZ, and CR doses for three design basis accidents to address non-conservative inputs used previously. While the revised doses generally increase, they are below the allowable regulatory limits. The margin of safety for the radiological consequences of these accidents is provided by meeting the applicable regulatory limits. An acceptable margin of safety is inherent in these limits.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

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

5.0 ENVIRONMENTAL EVALUATION

The proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. NRC letter to Entergy Operations, Inc. (Entergy), "Arkansas Nuclear One, Unit No. 1 – Issuance of Amendment RE: Use of Alternate Source Term (TAC No. MD7178)," (1CNA100901) (ML092740035), dated October 21, 2009.

2. NUREG/CR-6604, "RADTRAD – A Simplified Model for RADionuclide Transport and Removal And Dose Estimation".

3. NUREG/CR-6604, Supplement 1, "RADTRAD – A Simplified Model for RADionuclide Transport and Removal And Dose Estimation".

4. NUREG/CR-6604, Supplement 2, "RADTRAD – A Simplified Model for RADionuclide Transport and Removal And Dose Estimation".

5. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (ML063000023), dated July 2000.

6. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," (ML053460347), dated March 7, 2006.

7. NRC Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors," Revision 1, (ML070040189), dated March 2007.

8. NRC NUREG-0016, "Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors, GALE-BWR 3.2 Code," (ML20213C728), dated July 2020.

9. NRC NUREG-0017, "Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, GALE-PWR 3.2 Code," Revision 3, (ML20213C729), dated July 2020.

10. NRC NUREG-0800, "Standard Review Plan for the Review of Safety Reports for Nuclear Power Plants: LWR Edition," Section 11.1, "Coolant Source Terms," Revision 3 (ML070790010).

11. Framatome Topical Report BAW-10164-A, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," Revision 6.

12. Sham Rane and Li He, "CFD analysis of flashing flow in two-phase geothermal turbine design", Journal of Computational Design and Engineering, 2020, 7(2), 238 - 250.

13. NRC letter to Energy Harbor Nuclear Corp, "Beaver Valley Power Station, Units 1 and 2 – Issuance of Amendment Nos. 308 and 195 to Modify Primary and Secondary Coolant Activity Technical Specifications (EPID-2019-LLA-0223)," (ML20213A731), dated September 23, 2020.

ENCLOSURE 3

1CAN102201

AFFIDAVIT FROM MPR ASSOCIATES, INC.

(2 Pages)

MPR

September 29, 2022

AFFIDAVIT

I, John W. Simons, state and affirm as follows:

(1) I am the Vice President of Power Services at MPR Associates, Inc. (MPR) and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld from public disclosure and have been authorized to apply for its withholding.

(2) The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information sought to be withheld is contained in Enclosure to Entergy Letter 1CAN102201 to be issued on October 13, 2022.

The document is to be treated as MPR proprietary information because it contains information which has a commercial value to MPR.

- The evaluation includes details of an analysis methodology that MPR developed. The details of the analysis methodology were developed based on extensive research and deliberations among MPR personnel and reflect the combined technical expertise and insights of MPR. Public release of the information would concede intellectual property and a commercial advantage to others pursuing similar analysis methodology and/or technical services.

- The evaluation includes excerpts from the analysis that MPR performed using this methodology. Public release of the information would concede intellectual property and a commercial advantage to others engaged in evaluation of the flashing fraction calculation during a steam generator tube rupture event.

(3) The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

(4) The material for which exemption from disclosure is herein sought is considered proprietary for the following reasons:

a) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

b) The information is vital to a competitive advantage, would be helpful to competitors and would likely cause substantial harm to the competitive position.

(5) Public disclosure of the information sought to be withheld is likely to cause substantial harm to MPR.

(6) The foregoing statements are true and correct to the best of my knowledge, information, and belief.

John W. Simons
Vice President

SUBSCRIBED before me this 29th day of September 2022

Nancy Gail Svites [NOTARY]

Nancy Gail Svites
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MY COMMISSION EXPIRES FEBRUARY 28, 2025

NANCY GAIL SVITES
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MY COMMISSION EXPIRES 02/28/2025
COMMONWEALTH OF VIRGINIA