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

December 9, 2022

ANO Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
N-TSB-58
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - ISSUANCE OF AMENDMENT NO. 278 RE: REVISION TO TECHNICAL SPECIFICATIONS 3.4.12 and 3.4.13 BASED ON REVISED DOSE CALCULATIONS (EPID L-2021-LLA-0181)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 278 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 30, 2021, as supplemented by letters dated December 2, 2021, June 2, 2022, and October 13, 2022.

The amendment revises the Dose Equivalent Iodine I-131 and the reactor coolant system (RCS) primary activity limits required by ANO-1 TS 3.4.12, "RCS Specific Activity." In addition, the primary-to-secondary leak rate limit provided in ANO-1 TS 3.4.13, "RCS Operational LEAKAGE," is revised. These changes were proposed by the licensee due to non-conservative inputs used in the steam generator tube rupture accident, the main steam line break accident, and the control rod ejection accident dose calculations.

The NRC staff has determined that the related safety evaluation contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390, "Public inspections, exemptions, request for withholding." The proprietary information is indicated by bold text enclosed with **[[double brackets]]**. The proprietary version of the safety evaluation is provided as enclosure 2. Accordingly, the NRC staff has also prepared a nonproprietary version of the safety evaluation, which is provided as enclosure 3.

Enclosure 2 to this letter contains proprietary information. When separated from enclosure 2, this document is DECONTROLLED.

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

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 278 to DPR-51
2. Proprietary Safety Evaluation
3. Nonproprietary Safety Evaluation

cc without Enclosure 2: Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 278
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (EOI), dated September 30, 2021, as supplemented by letters dated December 2, 2021, June 2, 2022, and October 13, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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 the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Renewed Facility Operating License No. DPR-51 is hereby amended to read as follows:

    (2) Technical Specifications

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

3. This amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-51
and the Technical Specifications

Date of Issuance: December 9, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 278

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

Replace the following pages of Renewed Facility Operating License No. DPR-51 and the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.4.12-1	3.4.12-1
3.4.12-2	3.4.12-2
3.4.13-1	3.4.13-1
3.4.13-2	3.4.13-2

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, 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 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 components;

(6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, 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 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

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

(2) Technical Specifications

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

(3) Safety Analysis Report

The licensee's SAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 14, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than May 20, 2014.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security Plan, Training and Qualifications Plan, and Safeguards Contingency Plan," as submitted on May 4, 2006.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 RCS Specific Activity

LCO 3.4.12 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	-----NOTE----- LCO 3.0.4.c is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 6 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	-----NOTE----- LCO 3.0.4.c is applicable. ----------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> DOSE EQUIVALENT I-131 > 6 μCi/gm.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 5	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	-------------------------------NOTE--------------------------- Only required to be performed in MODE 1 and 2, MODE 3 with RCS average temperature ≥ 500 °F. -- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 2200 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.25 μCi/gm.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 39 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists.	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
B. RCS unidentified or identified LEAKAGE not within limits, except for primary to secondary LEAKAGE.	B.1 Reduce LEAKAGE to within limits.	18 hours
C. Required Action and associated Completion Time not met. <u>OR</u> Primary to secondary LEAKAGE not within limit.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------------------NOTES------------------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation at or near operating pressure. 2. Not applicable to primary to secondary LEAKAGE. -- Verify RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.	In accordance with the Surveillance Frequency Control Program
SR 3.4.13.2	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after establishment of steady state operation. -- Verify primary to secondary LEAKAGE is $\leq$ 39 gallons per day through any one SG.	In accordance with the Surveillance Frequency Control Program

ENCLOSURE 3

(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 278 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within [[double brackets]].

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 278 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated September 30, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21274A874), as supplemented by letters dated December 2, 2021 (ML21337A245), June 2, 2022 (ML22153A464), and October 13, 2022 (ML22286A249), Entergy Operations, Inc. (the licensee) requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1).

The proposed changes would revise the Dose Equivalent Iodine (I)-131 (DEI) and the reactor coolant system (RCS) primary activity limits required by ANO-1 TS 3.4.12, "RCS Specific Activity." In addition, the primary-to-secondary (PS) leak rate limit provided in ANO-1 TS 3.4.13, "RCS Operational Leakage," would be revised. These changes are proposed by the licensee due to non-conservative inputs used in the dose calculations for the steam generator tube rupture (SGTR) accident, the main steam line break (MSLB) accident, and the control rod ejection accident (CREA).

From January 31, 2022, through July 6, 2022, the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff conducted an audit to support its review of the amendment request, as discussed in the staff's audit plan dated January 21, 2022 (ML22019A134), and audit summary dated July 12, 2022 (ML22174A334).

On February 22, 2022, the NRC staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (87 FR 9651) for the proposed amendment. Subsequently, by letters dated June 2, 2022, and October 13, 2022, the licensee provided additional information that clarified the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, the NRC published a second proposed NSHC determination in the *Federal Register* on November 3, 2022 (87 FR 66328), which superseded the original notice in its entirety.

2.0 REGULATORY EVALUATION

2.1 Reason for the Proposed Changes

The licensee stated that in October 2018, it identified that the ANO-1 SGTR offsite dose evaluation was non-conservative because the calculations failed to consider post-reactor trip high pressure injection (HPI) flows during an SGTR accident. HPI flows would cause RCS pressure to remain at a higher value, which in turn would result in a higher PS 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 steam generator (SG), a potential for higher release and subsequent dose to the public became a concern.

As a result of this issue, the licensee took compensatory measures and initiated administrative controls to limit potential radioactive releases that could impact offsite doses in order to keep the ANO-1 SGs operable. The specific controls employed by the licensee were administrative limits to reduce the primary and secondary activity levels to 50 percent of the TS values while the subsequent analyses were completed.

As the thermal-hydraulic analysis was being updated, the licensee became aware that certain assumptions made for the SGTR radiological analysis did not appear to have sufficient justification. Specifically, the licensee determined that the “flashing fraction,” or the amount of RCS leakage that flashes to vapor when entering the secondary side of the SG, was potentially non-conservative. The licensee also determined that the MSLB and control rod ejection (CRE) analyses used potentially non-conservative flashing fractions.

Subsequently, the licensee instituted administrative controls to reduce the PS activity levels to 50 percent of the allowed TS values and reduced the allowable SG leakage from 150 gallons per day (gpd) to 30 gpd.

2.2 Licensee’s Proposed TS Changes

The licensee proposed more restrictive changes that would reduce the allowable RCS specific activities and would also lower the allowed PS leakage rate. The current limiting condition for operation (LCO) for TS 3.4.12 is that “RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE [Xenon]-133 [DEX] specific activity shall be within limits,” during Modes 1, 2, 3 and 4. Action A requires that the DEI be verified to be less than or equal to 60 microcuries per gram (μCi/gm) every 4 hours and to restore DEI to within limit in 48 hours. Action C requires that if the “[r]equired action and the associated completion time of Condition A or B are not met OR DOSE EQUIVALENT I-131 > 60 μCi/gm,” the plant must be in Mode 3 in 6 hours and in Mode 5 in 36 hours. The proposed TS change would revise the value for DEI for Conditions A.1 and C from 60 μCi/gm to 6 μCi/gm.

The current surveillance requirement (SR) 3.4.12.2 verifies that the reactor coolant DEI specific activity is less than or equal to 1.0 μCi/gm on a frequency in accordance with the Surveillance Frequency Control Program. The proposed TS change would revise the value for DEI for SR 3.4.12.2 from 1.0 μCi/gm to 0.25 μCi/gm.

The current LCO for TS 3.4.13 limits the RCS operational leakage to 150 gpd PS leakage through any one SG in Modes 1, 2, 3 or 4. If the PS leakage is not within limits, Action B requires the unit to be in Mode 3 in 6 hours and in Mode 5 within 36 hours. The proposed change would revise the value for the PS leakage limit for LCO 3.4.13.d and SR 3.4.13.2 from 150 gpd to 39 gpd through any one SG.

Following is a mark-up showing the licensee's proposed TS changes (additions in **<u>bold underline</u>**, deletions in ~~strikeout~~):

3.4.12 RCS Specific Activity

LCO 3.4.12 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	--------------------NOTE------------------ LCO 3.0.4.c is applicable ---	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ **<u>6</u>**~~60~~ µCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	--------------------NOTE------------------ LCO 3.0.4.c is applicable ---	
	B.1 Restore DOSE EQUIVALENT XE-131 to within limit.	48 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> DOSE EQUIVALENT I-131 > 6~~0~~ µCi/gm	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ **0.25**~~1.0~~ µCi/gm.	In accordance with the Surveillance Frequency Control Program

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:
a. No pressure boundary LEAKAGE;
b. 1 gpm unidentified LEAKAGE;
c. 10 gpm identified LEAKAGE; and
d. **39**~~150~~ gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG).

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.2	------------------------------NOTE-------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ **39**~~150~~ gallons per day through any one SG.	In accordance with the Surveillance Frequency Control Program

2.3 Regulatory Requirements and Guidance

The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

- In Section 50.36, “Technical specifications,” of Title 10 of the *Code of Federal Regulations* (10 CFR), the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls.

  Licensees may propose revisions to the TSs. The NRC staff reviews proposed changes and will generally issue changes provided that the plant-specific review supports a finding of continued reasonable assurance of adequate protection of public health and safety because: (1) the change is editorial, administrative, or provides clarification (i.e. no requirements are materially altered), (2) the change is more restrictive than the licensee’s current requirement, or (3) the change is less restrictive than the licensee’s current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards.

- The regulation in 10 CFR 50.67, "Accident source term," paragraph (b)(1), "Applicability," states:

  > A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

  The NRC staff evaluated the radiological consequences of affected design-basis accidents (DBAs) for ANO-1, as proposed by the licensee, against the dose criteria specified in 10 CFR 50.67(b)(2).

- Regulatory Guide (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 4, dated September 2012 (ML12159A013).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000 (ML003716792), provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms (ASTs).

- NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (ML003734190), assigns responsibilities for NRC staff review of applications related to the implementation of the AST. This section includes guidance for the review of emergency operating procedures and human factors engineering design, and states, in part, that "[a]n acceptable implementation of an AST should demonstrate compliance with plant-specific licensing commitments made in response to the NUREG-0737 ["Clarification of TMI [Three Mile Island] Action Plan Requirements," dated November 1980 (ML051400209)"]. Specific provisions of interest within the context of this review plan section include III.D.3.4, "Control Room Habitability," as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

3.0 TECHNICAL EVALUATION

In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the regulations, guidance, and licensing and design basis information discussed in section 2.3 of this safety evaluation (SE). The NRC staff reviewed the licensee's statements in the license amendment request (LAR) dated September 30, 2021, as supplemented, the relevant sections of the ANO-1 safety analysis report, the licensee's calculations, and the ANO-1 TS Bases to determine if the proposed changes are acceptable.

3.1 Radiological Consequences of DBAs

To support the necessary changes to the TSs, the licensee revised the dose consequence analyses of record for the SGTR, MSLB and CRE DBAs. The revised analyses are evaluated below. Tables 1 through 6 at the end of the SE provide information and assumptions supporting the licensee's proposed changes, based on information in the licensee's LAR, as supplemented, and ANO-1 License Amendment No. 238 (ML092740035) regarding the adoption of the AST.

3.1.1 SGTR Accident

The SGTR accident analysis assumes the instantaneous rupture of a SG tube with a resultant release of reactor coolant into the lower pressure secondary system. Based on an assumption of a loss of offsite power (LOOP) occurring simultaneously with the reactor trip, the condenser is assumed to be unavailable. Environmental steam releases via the main steam safety valves (MSSVs) and atmospheric dump valves (ADVs) of the intact SG are used to cool down the reactor until initiation of shutdown cooling via the residual heat removal (RHR) system. A portion of the reactor coolant break flow into the ruptured SG flashes and is released to the condenser for a short duration prior to the reactor trip and thereafter, directly to the environment via the MSSVs and/or ADVs (MSSVs/ADVs). The remaining break flow mixes with the secondary side liquid and is released to the environment via steam releases through MSSVs/ADVs. The activity in the RCS also leaks into the intact SG via SG tube leakage and is released to the environment from the MSSVs/ADVs.

The NRC staff previously evaluated the radiological consequences for the SGTR accident for ANO-1 in License Amendment No. 238 (ML092740035) regarding the adoption of the AST. Notable changes between the current LAR and the previous SGTR dose consequence analysis include changes to the value for the RCS iodine pre-accident spike limit from 60 to 6 µCi/gr DEI and the iodine equilibrium activity from 1.0 µCi/gr to 0.25 µCi/gr DEI. In addition, changes were made to the fraction of ruptured flow and PS leakage that is vaporized based on revised analyses.

Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR [Pressurized-Water Reactor] Steam Generator Tube Rupture Accident," of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR. The licensee's analysis indicates that no fuel melting or fuel cladding failure is postulated for the SGTR accident. Therefore, in accordance with RG 1.183, the licensee assumed that the activity released is based on the maximum coolant activity allowed by the ANO-1 proposed maximum allowable coolant activity. Following the guidance in RG 1.183, the licensee considered two spiking scenarios, a pre-accident iodine spike and an accident-initiated iodine spike, which resulted in an increase in the iodine appearance rate from the fuel to the RCS by a factor of 335.

The NRC staff notes that the ANO-1 SGTR analysis has a significant added conservatism since the dose consequence analysis maintained the current licensing basis (CLB) value of 150 gpd per SG for PS leakage. As described in the LAR, the TS value for PS leakage will be reduced from 150 to 39 gpd per SG in accordance with the proposed TS change to ensure that the dose consequence from the CREA meets the accident-specific acceptance criteria stated in RG 1.183 and NUREG-0800 Section 15.0.1. Since the SGTR dose consequence analysis meets the acceptance criteria using the higher value for PS leakage, the revised SGTR analysis has a considerable amount of margin that is not reflected in the licensee's calculated dose consequences, as shown on table 1 of this SE.

PS leakage is assumed to be 150 gpd into the bulk water of the ruptured SG and 150 gpd into the bulk water of the unaffected SG. The iodine activity from the flashed portion of the break flow through the ruptured SG is assumed to be directly released to the environment with no iodine partitioning. The radionuclides in the intact SG bulk water are assumed to become vapor at a rate that is a function of the steaming rate for the SGs and the partition coefficient. The licensee assumed that the radionuclide concentration in the SG is partitioned such that 1.0 percent of the radionuclides in the unaffected SG bulk water enter the vapor space and are released to the environment. The steam release from the unaffected SGs continues for approximately 8 hours until the RHR shutdown cooling system can be used to complete the cooldown.

3.1.1.1 SGTR Accident Conclusion

Based on the above, the NRC staff concludes that the licensee's analysis was performed using acceptable models, assumptions, and inputs performed consistent with applicable regulatory guidance identified in section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in tables 3 and 4 of this SE and the licensee's calculated dose results are provided in table 1 of this SE. The SGTR atmospheric dispersion coefficients previously accepted by the NRC are shown in tables 2A and 2E of this SE. The NRC staff reviewed the licensee's evaluation of the radiological consequences resulting from the postulated SGTR and concludes that the radiological consequences at the exclusion area boundary (EAB), low population zone (LPZ), and control room continue to meet the acceptance criteria provided in 10 CFR 50.67 and the accident-specific acceptance criteria specified in NUREG-0800 Section 15.0.1. Therefore, the licensee's SGTR evaluation is acceptable from a dose consequence perspective.

3.1.2 MSLB

This accident consists of a double-ended break of one main steam line (MSL). The licensee's analysis focuses on a MSLB outside the containment, since a MSLB inside containment results in a lesser dose to a control room operator or to the public at offsite locations due to holdup of activity in the containment. Following a MSLB outside containment, the faulted SG rapidly depressurizes and releases its initial secondary side liquid contents to the environment via the break. Based on an assumption of a simultaneous LOOP, the condenser is assumed to be unavailable, and environmental steam releases via the MSSVs/ADVs of the intact SGs are used to cool down the reactor until initiation of shutdown cooling via the RHR system. The NRC staff previously evaluated the radiological consequences for the MSLB accident for ANO-1 in License Amendment No. 238 regarding the adoption of the AST. Notable changes from the CLB MSLB dose consequence analysis include changes to the value for the fraction of PS leakage that is assumed to vaporize in the unaffected SG, as well as changes in the values used for the RCS and secondary system liquid mass.

The licensee's CLB analysis indicates that no fuel melting or fuel cladding failure is postulated for the ANO-1 MSLB event. Therefore, in accordance with RG 1.183, the licensee assumed that the activity released is based on the maximum coolant activity allowed by the ANO-1 TSs. Following the guidance in RG 1.183, the licensee considered two spiking scenarios: (1) a pre-accident iodine spike that reflects the maximum allowable TS iodine spike activity level, and (2) an accident-initiated iodine spike that results in an increase in the iodine appearance rate from the fuel to the RCS by a factor of 500.

The NRC staff notes that the ANO-1 MSLB analysis has significant added conservatisms, since the revised dose consequence analysis maintained the CLB values for both PS leakage and the RCS DEI values, which would be significantly reduced by the implementation of the proposed license amendment. As described in the LAR, the TS allowable value for PS leakage would be reduced in accordance with the proposed TS changes.

3.1.2.1 MSLB Accident Conclusion

The NRC staff concludes that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in tables 3 and 5 this SE and the licensee’s calculated dose results are provided in table 1 of this SE. The MSLB atmospheric dispersion coefficients previously accepted by the NRC are shown in tables 2B and 2E of this SE. The NRC staff reviewed the licensee’s evaluation of the radiological consequences resulting from the postulated MSLB and concludes that the radiological consequences at the EAB, LPZ, and control room continue to meet the acceptance criteria provided in 10 CFR 50.67 and the accident-specific acceptance criteria specified in NUREG-0800 Section 15.0.1. Therefore, the licensee’s MSLB evaluation is acceptable from a dose consequence perspective.

3.1.3 CREA

This accident consists of an uncontrolled withdrawal of a control rod from the reactor core. The CREA results in reactivity insertion that leads to a core power level increase resulting in fuel damage, and a subsequent reactor trip. Following the reactor trip and based on an assumption of a LOOP coincident with the reactor trip, the condenser is assumed to be unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs/ADVs.

The NRC staff previously evaluated the radiological consequences for the CREA for ANO-1 in License Amendment No. 238 regarding the adoption of the AST. Notable changes from the previous CREA dose consequence analysis include changes in the values used for the RCS and secondary system liquid mass, the fraction of PS leakage that is assumed to be vaporized, and a significant reduction in the allowable PS leakage rate as governed by the proposed TSs.

Per RG 1.183, the CREA evaluation addresses two independent release scenarios. The first release path scenario evaluated is the containment leakage pathway, which assumes that the released fission products breach the RCS and are released into the containment. The second release pathway scenario assumes that the RCS remains intact and that secondary side releases occur from the assumed PS leakage. With the assumption of a LOOP concurrent with the reactor trip, the condenser is unavailable for decay heat rejection. Therefore, the ADVs and MSSVs are used to cool down the plant.

The NRC staff notes that the doses resulting from a postulated CREA could be a composite of the releases from the containment leakage pathway and the secondary system release pathway. Therefore, if the evaluation indicates that the acceptance criteria are met for each of the independent scenarios, then the dose consequence of a scenario that is a combination of the two will be encompassed by the more restrictive of the two analyzed scenarios.

3.1.3.1 CREA Conclusion

The NRC staff concludes that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in tables 3 and 6 of this SE and the licensee's calculated dose results are provided in table 1 of this SE. The CREA atmospheric dispersion coefficients previously accepted by the NRC are shown in tables 2C, 2D and 2E of this SE. The NRC staff reviewed the licensee's evaluation of the radiological consequences resulting from the postulated CREA and concludes that the radiological consequences at the EAB, LPZ, and control room continue to meet the acceptance criteria provided in 10 CFR 50.67 and the accident-specific acceptance criteria specified in NUREG-0800 Section 15.0.1. Therefore, the licensee's CREA evaluation is acceptable from a dose consequence perspective.

3.1.4 Control Room Habitability

The NRC staff previously evaluated the ANO-1 control room radiological consequences in License Amendment No. 238 regarding the adoption of the AST. The licensee maintained all of the major CLB control room assumptions as approved in License Amendment No. 238 with the exception of a few minor changes. Changes to the CLB assumptions include a uniform unfiltered inleakage after isolation of 82 cubic feet per minute (cfm), and a reduction in the time for control room isolation for the CREA from 11 minutes to 9.6 minutes, consistent with the timing for a reactor trip. These minor changes have no significant impact on the dose consequence analysis.

The licensee noted that the two control room emergency ventilation system trains differ in that one train consists of a single 4-inch charcoal bed and the other train consists of two 2-inch charcoal beds in series. Therefore, both trains have 4 inches of charcoal filtration, which, as described in RG 1.52, allows for crediting 99 percent removal for both elemental iodine and organic iodide.

The licensee evaluated the dose to the control room from the cloud of radioactive material outside the control room with credit for the considerable amount of concrete shielding present. This was accomplished by evaluating the whole body direct dose at the control room intake after applying a reduction in the dose conversion factors using attenuation factors based on the significant amount of control room shielding.

3.2 Atmospheric Dispersion Estimates

The licensee did not propose any changes to the CLB atmospheric dispersion coefficients for this LAR. The meteorological data, methodology, and previously approved atmospheric dispersion factors (χ/Q values) are discussed in the SE associated with License Amendment No. 238 regarding the adoption of the AST. For completeness, tables 2A through 2D of this SE are included to indicate the specific atmospheric dispersion coefficients used in the control room dose consequence analyses for the three accidents evaluated by the licensee. Table 2E of this SE shows the offsite atmospheric dispersion coefficients used for all the licensee's dose consequence analyses.

3.3 Thermal-Hydraulic Analysis Evaluations

The licensee's thermal-hydraulic calculations of the SGTR were performed with the RELAP5/MOD2-B&W computer code (Framatome Topical Report BAW-10164P A, Revision 6, "RELAP5/MOD2 B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."), which has been approved for use by the NRC for both loss-of-coolant accident (LOCA) and non-LOCA safety analysis on B&W-designed PWRs.

The NRC staff notes that the consideration of HPI flow during an SGTR does not have any effect on the MSLB and CRE accidents, so these two accidents did not require the licensee to perform an updated thermal-hydraulic analysis.

3.3.1 SGTR

For the SGTR thermal-hydraulic analysis, the licensee assumed that a double-ended rupture of a SG tube occurs with unrestricted discharge from each end while operating at full power. The initial break flow exceeds the normal makeup to the RCS and results in the primary system pressure decreasing. The licensee's analysis assumes no initial operator action, which allows the primary system pressure to decrease until a reactor trip occurs on low reactor coolant pressure. The main turbine trips because of the reactor trip. Closure of the turbine stop valves causes the MSL pressure to increase resulting in the MSSVs to open to relieve pressure. The primary side pressure continues to decrease until the engineered safety actuation system low pressure setpoint is reached and flow from two HPI pumps is started. The HPI flow is significantly larger than the break flow, so the primary side pressure and level begin to recover.

The licensee assumed a LOOP coincident with a reactor trip. The LOOP results in reactor coolant pump trip, main feedwater pump trip, and the loss of the condenser. Upon a main feedwater trip, the emergency feedwater actuates automatically to raise the SG level to 50 percent of the operating range.

While the licensee proposed to reduce the allowable PS leakage in LCO 3.4.13.d from 150 gpd through any one SG to 39 gpd, the licensee's calculations conservatively assume the current 150 gpd throughout the accident.

The licensee determined the limiting single failure to be the failure of the ADV on the unaffected SG to initially open, requiring the dispatch of an operator to manually open the ADV. This failure allows for extensive steam release from the affected SG. At 30 minutes, the operator uses the ADV on the affected SG to begin cooling down the plant. The majority of the decay heat and RCS sensible heat is removed by steaming the affected SG to the atmosphere. The assumed failure provides the latest isolation time of the affected SG as compared with the other ADV failure scenarios considered by the licensee. The licensee assumed that the operator manually opens the ADV on the unaffected SG at 60 minutes and isolates the affected SG at 70 minutes.

The NRC staff reviewed the licensee's SGTR thermal-hydraulic calculations during the regulatory audit. The licensee's calculations considered multiple cases including SGTR with and without a LOOP, as well as both early and delayed reactor trips. The dose calculations were performed with the SGTR with a LOOP and an early reactor trip. The NRC staff finds that the licensee's SGTR calculations were performed consistent with the approved methodology using appropriately conservative assumptions and remains consistent with the design basis.

3.3.2 Flashing Fraction Determination

The flashing fraction is used to quantify the amount of steam generated from a leak or break. The flashing fraction is defined as the ratio between the mass of generated steam and total mass of the leak or break flow that escapes from the primary loop into the SG secondary side. 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 an SGTR accident, any steam release through the MSSVs/ADVs in the MSL will result in an offsite release.

There are two different flashing fractions used by the licensee in the revised calculations. The first is the flashing fraction for the PS leakage flow and the second is for the SGTR break flow.

As specified in the supplemental letter dated December 2, 2021, the flashing fractions for the PS leakage are as follows:

Accident	**Current Licensing Basis Value**	**Proposed Value**
CRE	15 percent	100 percent
MSLB	100 percent (affected SG) 20 percent (intact SG)	100 percent (affected SG) 100 percent (intact SG)
SGTR	15 percent	100 percent

The CLB values of 15 percent and 20 percent were identified by the licensee as potentially non-conservative. To address the concern, the licensee proposed increasing these values to 100 percent. A flashing fraction of 100 percent is conservative, as all the leakage flow is converted to steam, which has a release path to the atmosphere outside the containment through either the MSSVs or ADVs. Therefore, the NRC staff finds the use of a flashing fraction of 100 percent acceptable for the PS leakage flow.

The CLB value for the SGTR break flow flashing fraction is 100 percent pre-trip and 15 percent post-trip. To address the potential non-conservatism, the licensee considered using a flashing fraction of 100 percent post-trip, however, the licensee concluded that this was too restrictive. Therefore, the licensee computed a new flashing fraction for the SGTR break flow. The new value accounts for the effects of: **[[**

]] Each of the individual effects are discussed in more detail in sections 3.3.2.1 through 3.3.2.4 of this SE. The licensee computed the steam generation rate for each of these four effects, then summed them to obtain a total steam generation rate. This was then used with the total break flow to determine an overall flashing fraction, which is then used in the dose calculations. The licensee's calculations to determine the flashing fractions were performed using a spreadsheet with inputs taken from the thermal-hydraulic analysis discussed in section 3.3.1 of this SE. Specifically, the calculation of the break flashing fraction considers the results of the SGTR scenario with a coincident LOOP for both the delayed and early reactor trip cases. The flashing fractions used in the dose calculations encompass both the delayed and early reactor trip cases.

The NRC staff reviewed the calculations performed by the licensee and performed independent confirmatory calculations as part of the regulatory audit. The NRC staff did not identify any calculational errors in the licensee's spreadsheet.

3.3.2.1 **[[**

]] The NRC staff reviewed the calculations during the regulatory audit and found that they were performed appropriately.

3.3.2.2 **[[**

]] Therefore, the NRC staff finds that the assumed number of wetted SG tubes is sufficiently conservative. The heat transfer from the SG tubesheet is discussed in section 3.3.2.3 of this SE.

[[

]]

The NRC staff reviewed the **[[** **]]** calculations performed by the licensee and finds the licensee's calculations and applied correlation to be acceptable for this specific SG geometry and fluid conditions**.**

[[

]] The NRC staff finds this calculation was performed appropriately.

[[

]]

Upon reviewing the **[[** **]]** calculations performed by the licensee, the NRC staff finds the licensee's calculations and applied correlation to be acceptable for this specific SG geometry and fluid conditions.

3.3.2.3 **[[**

]]

The NRC staff reviewed the [[ ]] calculations performed by the licensee and finds the calculations, correlations, and assumptions acceptable for this specific SG geometry and fluid conditions.

3.3.2.4 [[

]]

The NRC staff reviewed the licensee's calculation [[
]] and finds the calculations, correlations, and assumptions acceptable for this specific SG geometry and fluid conditions.

3.3.2.5 SGTR Flashing Fraction

The licensee computed the total steam generation rate as the sum of steam generation due to the following effects: [[

]] The total steam generation rate is shown on page 7 of enclosure 1 (proprietary information) of the supplemental letter dated June 2, 2022. This figure shows that the total steam generation rate [[
]]. This remains the case up through the time of reactor trip, resulting in a flashing fraction of 100 percent. After reactor trip, the steam generation rate drops below the break flow rate due to the changes in primary and secondary side flow rates, which result in less heat transfer from the SG tubes and superheated steam. The licensee calculated the flashing fraction as a function of time, as shown on page 2 of enclosure 1 of the letter dated June 2, 2022. [[

]]

The NRC staff finds the resulting flashing fraction acceptable, as the licensee's calculations conservatively accounted for the appropriate heat transfer mechanisms that convert the PS break flow into steam.

3.4 Human Factors Review

The NRC staff reviewed the LAR with respect to human factors considerations. In its supplemental letter dated June 2, 2022, the licensee stated that there would not be any substantive changes to the ANO-1 emergency operating procedures (EOPs) and that that the current operator actions required to respond to a SGTR will not be changed as a result of the LAR.

The NRC staff reviewed the LAR, as supplemented, in accordance with NUREG-0800, Section 15.0.1 and NUREG-0737. The NRC staff finds the request acceptable from a human factors perspective because there are no substantive changes to the EOPs or credited manual actions from the ANO-1 CLB.

3.5 Summary of the Technical Evaluation

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs at ANO-1. The NRC staff finds that the licensee used assumptions, inputs, and methods consistent with the regulatory requirements and guidance identified in section 2.0 of this SE. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria identified in section 2.0 of this SE. The NRC staff concludes that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. The NRC staff further finds that ANO-1, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated accidents and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs.

In addition, the NRC staff reviewed the licensee's thermal-hydraulic calculations of the SGTR accident, which considered multiple cases, including an SGTR with and without LOOP, as well as both early and delayed reactor trips. The NRC staff finds that the licensee's SGTR calculations were performed consistent with the approved methodology using appropriately conservative assumptions and remain consistent with the design basis. The NRC staff finds the computed flashing fraction acceptable for use in the offsite dose calculations for a SGTR accident.

Based on the above, the NRC staff concludes that the proposed changes provide for more restrictive operations regarding the allowable RCS specific activity and PS leakage, and that the proposed TS changes will ensure that the DBA dose consequence analyses meet their associated dose acceptance criteria. Therefore, the NRC staff finds that the proposed changes to TSs 3.4.12 and 3.4.13 assure continued operation of the facility in a safe manner and that the requirements of 10 CFR 50.36 and 10 CFR 50.67 will continue to be met. Therefore, the NRC staff finds the proposed TS changes to be acceptable.

Table 1
ANO-1 Radiological Consequences Expressed as TEDE [1]
(rem)

Design-Basis Accidents	EAB[2]	LPZ[3]	CR
MSLB Accident[4]	0.32	0.18	1.08
SGTR[4]	1.79	0.29	3.68
Dose Acceptance Criteria	25	25	5
CREA[6]	4.72	2.27	3.12
CREA[7]	3.29	2.20	4.96
Dose Acceptance Criteria	6.3	6.3	5
MSLB Accident[5]	1.26	0.72	2.55
SGTR[5]	2.45	0.41	2.31
Dose Acceptance Criteria	2.5	2.5	5

[1] total effective dose equivalent
[2] exclusion area boundary
[3] low population zone
[4] pre-existing spike
[5] accident-induced spike
[6] assumes containment release
[7] assumes secondary side release

Table 2A
ANO-1 SGTR CR Atmospheric Dispersion Coefficients (sec/m^3)

Time (Hours)	Release from MSSV	Releases from ADV	Values used in SGTR Analysis	Comments on Values used in SGTR Analysis
0.0 - 0.16	1.90×10^{-2}	4.10×10^{-3}	4.10×10^{-3}	ADV values used for condenser releases
0.16 - 1.0	1.90×10^{-2}	4.10×10^{-3}	1.90×10^{-2}	MSSV release after reactor trip and assumed LOOP. Cooldown is started using the intact SG but operators notice ADV on intact SG fails to open, so cooldown is started using ruptured SG. Operator is dispatched to address ADV on intact SG.
1.0 - 2.0	1.90×10^{-2}	4.10×10^{-3}	4.10×10^{-3}	ADV values used after failed ADV block valve is manually opened
2.0 - 8.0	1.23×10^{-2}	2.59×10^{-3}	2.59×10^{-3}	
8.0 - 24	5.83×10^{-3}	1.12×10^{-3}	1.12×10^{-3}	
24 - 96	3.80×10^{-3}	8.32×10^{-4}	8.32×10^{-4}	
96 - 720	3.10×10^{-3}	5.91×10^{-4}	5.91×10^{-4}	

Table 2B
ANO-1 MSLB Control Room Atmospheric Dispersion Coefficients (sec/m^3)

Time (Hours)	Release from MSSV	Release from ADV	Release from Main Steam Pipe	Values used in MSLB Analysis	Comments on Values used in MSLB Analysis
0.0 - 0.5	1.90×10^{-2}	4.10×10^{-3}	3.15×10^{-3}	3.15×10^{-3}	Used for secondary coolant release in the faulted loop
0.0 - 0.5	1.90×10^{-2}	4.10×10^{-3}	3.15×10^{-3}	1.90×10^{-2}	The normally closed motor-operated block valve isolating the ADV on the intact loop fails to open so that the releases are via the MSSV until an operator manually opens the ADV block valve at 30 minutes.
0.5 - 2.0	1.90×10^{-2}	4.10×10^{-3}	3.15×10^{-3}	4.10×10^{-3}	ADV releases continue for the duration of the accident.
2.0 - 8.0	1.23×10^{-2}	2.59×10^{-3}	2.16×10^{-3}	2.59×10^{-3}	
8.0 - 24	5.83×10^{-3}	1.12×10^{-3}	8.90×10^{-4}	1.12×10^{-3}	
24 - 96	3.80×10^{-3}	8.32×10^{-4}	6.61×10^{-4}	8.32×10^{-4}	
96 - 720	3.10×10^{-3}	5.91×10^{-4}	5.01×10^{-4}	5.91×10^{-4}	

Table 2C
ANO-1 CREA CR Atmospheric Dispersion Coefficients (sec/m^3)
Secondary Side Release Scenario

Time (Hours)	Release from MSSV	Releases from ADV	Values used in CREA Analysis	Comments on Values used in CREA Secondary Side Release Analysis
0.0 - 0.5	1.90×10^{-2}	4.10×10^{-3}	1.155×10^{-2}	Since the releases from each loop are approximately equal, the value used is the average of the MSSV and ADV values.
0.5 - 2.0	1.90×10^{-2}	4.10×10^{-3}	4.10×10^{-3}	The normally closed motor-operated block valve on one loop is assumed to fail to open so that the releases are via the MSSV after the trip until an operator manually opens the ADV block valve after 30 minutes.
2.0 - 8.0	1.23×10^{-2}	2.59×10^{-3}	2.59×10^{-3}	ADV releases continue for the duration of the accident.
8.0 - 24	5.83×10^{-3}	1.12×10^{-3}	1.12×10^{-3}	
24 - 96	3.80×10^{-3}	8.32×10^{-4}	8.32×10^{-4}	
96 - 720	3.10×10^{-3}	5.91×10^{-4}	5.91×10^{-4}	

Table 2D
ANO-1 CREA CR Atmospheric Dispersion Coefficients (sec/m^3)
Containment Release Scenario

Time (Hours)	Containment Release Values used in CREA Analysis	Comments on Values used in CREA Containment Release Analysis
0.0 - 2.0	3.55×10^{-3}	The highest values of the dispersion factors for a diffuse release from the ANO-1 containment to either control room intake are applied.
2.0 - 8.0	2.49×10^{-3}	
8.0 - 24	9.85×10^{-4}	
24 - 96	8.30×10^{-4}	
96 - 720	6.31×10^{-4}	

Table 2E
ANO-1 Offsite Atmospheric Dispersion Coefficients (sec/m^3)

Time (Hours)	EAB	LPZ
0.0 - 2.0	6.8×10^{-4}	1.1×10^{-4}
2.0 - 8.0	N/A	1.1×10^{-4}
8.0 - 24		1.1×10^{-5}
24 - 96		4.0×10^{-6}
96 - 720		1.3×10^{-6}

Table 3
ANO-1 Control Room Data Assumptions

Control Room Volume	400,000 ft^3
Normal Air Intake	35,200 cfm
Recirculation Filter Flow	1,667 cfm
Recirculation Filter Efficiency	
Elemental iodine	95 percent
Organic iodide	95 percent
Aerosols/Particulates	99 percent
Filtered Intake Flow	333 cfm
Intake Filter Efficiency	
Elemental iodine	99 percent
Organic iodide	99 percent
Aerosols/Particulates	99 percent
Isolation Time Based on Control Room Intake High Radiation	10 seconds
Time to Detect High Radiation in Control Room Intake	
SGTR	11 minutes
MSLB	0 seconds
CREA	0 seconds
Unfiltered Inleakage After Isolation	82 cfm
Control Room Operator Breathing Rate	
0 - 720 hours	3.5 x10-4 m^3/sec
Control Room Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4

Table 4
ANO-1 Data and Assumptions for the SGTR Accident

RCS Mass	537,000 lbm
Secondary System Mass	60,000 lbs per SG
Time of Reactor Trip	9.6 minutes
Percentage of Failed Fuel Rods	0 percent
PS Leak Rate	150 gpd per SG[1]
Reactor Coolant Noble Gas DEX	2200 μCi/gm
Reactor Coolant Equilibrium Iodine Inventory	0.25 μCi/gm DEI
Secondary System Equilibrium Iodine Inventory	0.1 μCi/gm DEI
Pre-Accident Iodine Spike Activity	6 μCi/gm DEI
Concurrent Iodine Spike Appearance Multiplier	335 times equilibrium rate
Duration of Concurrent Iodine Spike	8 hours
Fraction of PS Leakage that is Vaporized	100 percent
Fraction of Ruptured Flow that Flashes	
Before Reactor Trip (T=0 to T=580 seconds)	100 percent
From T=580 to T=645 seconds	100 percent
From T=645 to T=1080 seconds	65 percent
From T=1080 seconds to isolation	40 percent
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbs/ft^3
Single Active Failure	Failure of ADV block valve to open
Response Time to Open ADV Block Valve	30 minutes
Initial Ruptured Tube Flow Rate	36.83 lbs per second
Time to Isolate Ruptured SG	70 minutes
Time to Isolate Intact SG	237.8 hours
SG Iodine Partition Coefficient	100
SG Moisture Carryover	0.1 percent
Condenser Partition Coefficient	10,000
Iodine Species Released to Environment	97 percent elemental 3 percent organic

[1]The assumed PS value of 150 gpd in the SGTR analysis is conservative relative to the revised TS value of 39

Table 5
ANO-1 Data and Assumptions for the MSLB Accident

RCS Mass	537,000 lbm
Secondary System Mass	60,000 lbs per SG
Time of Reactor Trip	0 seconds
Percentage of Failed Fuel Rods	0 percent
PS Leak Rate	0.5 gpm per SG(1)
Reactor Coolant Noble Gas DEX	2200 μCi/gm DEX
Reactor Coolant Equilibrium Iodine Inventory	1.0 μCi/gm DEI(2)
Secondary System Equilibrium Iodine Inventory	0.1 μCi/gm DEI
Pre-Accident Iodine Spike Activity	60 μCi/gm DEI(2)
Concurrent Iodine Spike Appearance Rate	500 Xs(3) equilibrium appearance
Duration of Concurrent Iodine Spike	8 hours
Fraction of PS Leakage that is Vaporized	100 percent
Density used for Leakage Volume-to-Mass Conversion	62.4 lbs/ft^3
Single Active Failure	Failure of ADV block valve to open
Response Time to Open ADV Block Valve	30 minutes
Time to Isolate Affected SG	251.8 hours
Time to Isolate Intact (Unaffected) SG	237.8 hours
ntact SG Iodine Partition Coefficient	100
SG Moisture Carryover	0.1 percent
Condenser Partition Coefficient	10,000
odine Species Released to Environment	97% elemental; 3% organic

(1) The assumed PS value of 0.5 gpm in the MSLB analysis is conservative relative to the revised TS value of 39 gpd.
(2) MSLB assumed values for primary coolant concentrations are conservative relative to the revised TS values.
(3) "500 times" the equilibrium appearance rates

Table 6
ANO-1 Data and Assumptions for the CRE Accident

Time of Reactor Trip	0 seconds
Percentage of Failed Fuel Rods	14 percent
Peaking Factor of Damaged Rods	1.8
Failure Mechanism	Departure from Nucleate Boiling
Fraction of Core Inventory in Fuel Rod Gap	
Noble Gases	10 percent
Iodine Gases	10 percent
Alkali Metals	12 percent
Primary-to-Secondary Leakage Case	
RCS Mass	537,000 lbm
PS Leak Rate	39 gpd per SG
Fraction of PS Leakage that is Vaporized	100 percent
Density used for Leakage Volume-to-Mass Conversion	62.4 lbs/ft^3
Time to Isolate SGs	38.25 hours
Single Active Failure	Failure of ADV block valve to open
Response Time to Open ADV Block Valve	30 minutes
odine Chemical Species Released to Environment	97% elemental; 3% organic
Containment Leakage Case	
Containment Volume	1,810,000 ft^3
Containment Leakage Rate	
0 – 24 hours	0.2 percent per day
After 24 hours	0.1 percent per day
Containment Natural Aerosol Deposition Rate	0.1 per hour for 69 hours
Containment Spray Removal	No Removal Credit
odine Chemical Species	
Aerosol (Particulate)	95 percent
Elemental	4.85 percent
Organic	0.15 percent

4.0 FINAL NSHC DETERMINATION

The NRC staff proposed to find that the requested amendment involves NSHC in its *Federal Register* notice of November 3, 2022 (87 FR 66328). The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves NSHC if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

An evaluation of the issue of NSHC is presented 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 control room 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 accident previously evaluated?

   Response: No.

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

   Therefore, the proposed changes do 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 control room 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 changes do not involve a significant reduction in a margin of safety.

Based on the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that NSHC is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment on November 4, 2022. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, published in the *Federal Register* on November 3, 2022 (87 FR 66328), and there has been no comment on such finding. Additionally, the Commission has made a final determination that no significant hazards consideration is involved for the proposed amendment as discussed above in section 4.0 of this SE. Accordingly, the amendment meets 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 amendment.

7.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Parillo, NRR
N. Amini, NRR
R. Beaton, NRR
K. Martin, NRR

Date: December 9, 2022

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - ISSUANCE OF AMENDMENT NO. 278 RE: REVISION TO TECHNICAL SPECIFICATIONS 3.4.12 and 3.4.13 BASED ON REVISED DOSE CALCULATIONS (EPID L-2021-LLA-0181) DATED DECEMBER 9, 2022

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RidsRgn4MailCenter Resource
JParillo, NRR
NAmini, NRR
RBeaton, NRR
KMartin, NRR

ADAMS Accession Nos.
ML22257A157 (Proprietary)
ML22263A191 (Non-Proprietary)

***by email**

OFFICE	NRR/DORL/LPL4/PM*	NRR/DORL/LPL4/LA*	NRR/DRA/ARCB/BC*	NRR/DSS/STSB/BC(A)*
NAME	TWengert	PBlechman	KHsueh	DWoodyatt
DATE	9/13/2022	11/1/2022	7/22/2022	7/22/2022
OFFICE	NRR/DORS/IOLB/TL*	NRR/DSS/SCPB/BC*	NRR/DNRL/NCSG/BC*	NRR/DSS/STSB/BC*
NAME	BGreen	BWittick	SBloom	VCusumano
DATE	8/5/2022	11/4/2022	11/8/2022	11/3/2022
OFFICE	OGC - NLO*	NRR/DORL/LPL4/BC*	NRR/DORL/LPL4/PM*	
NAME	BAyersman	JDixon-Herrity	TWengert	
DATE	11/28/2022	12/1/2022	12/9/2022	

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