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

February 8, 2021

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 2 - ISSUANCE OF AMENDMENT NO. 323 RE: TECHNICAL SPECIFICATION DELETIONS, ADDITIONS, AND RELOCATIONS (EPID L-2019-LLA-0284)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 323 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 16, 2019, as supplemented by letters dated June 17, 2020, and September 22, 2020.

The amendment revises several TS requirements by the addition, deletion, or relocation of certain TS limiting conditions for operation, Actions, and surveillance requirements. Relocated TSs are placed in the ANO-2 Technical Requirements Manual or the associated TS Bases.

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. 50-368

Enclosures:
1. Amendment No. 323 to NPF-6
2. Safety Evaluation

cc: Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 323
Renewed License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated December 16, 2019, as supplemented by letters dated June 17, 2020, and September 22, 2020, 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. NPF-6 is hereby amended to read as follows:

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

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 323, are hereby incorporated in the renewed license. The licensee 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

Jennifer L.
Dixon-Herrity

Digitally signed by
Jennifer L. Dixon-Herrity
Date: 2021.02.08
18:20:38 -05'00'

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. NPF-6 and
the Technical Specifications

Date of Issuance: February 8, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 323

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

Replace the following pages of Renewed Facility Operating License No. NPF-6 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	REMOVE	INSERT
3/4 1-3	3/4 1-3	3/4 7-18	3/4 7-18
3/4 1-4	---	3/4 7-27	---
3/4 3-25	3/4 3-25	3/4 7-28	---
3/4 3-26	3/4 3-26	3/4 7-38	---
3/4 3-26a	---	3/4 9-2	3/4 9-2
3/4 3-27	3/4 3-27	3/4 9-3	---
3/4 3-36	3/4 3-36	3/4 9-4	3/4 9-4
3/4 3-37	---	3/4 9-6	---
3/4 3-38	---	3/4 9-7	---
3/4 4-14a	3/4 4-14a	3/4 9-8	---
3/4 7-10	3/4 7-10	3/4 11-1	---
3/4 7-11	---	3/4 11-2	---
3/4 7-12	---	3/4 11-3	---
3/4 7-13	---	3/4 11-4	---
3/4 7-14	---	3/4 11-5	---
3/4 7-16a	---	6-7	6-7
3/4 7-16b	---		

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

(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 or instrument calibration or associated with radioactive apparatus or components; and

(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 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; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 3026 megawatts thermal. Prior to attaining this power level EOI shall comply with the conditions in Paragraph 2.C.(3).

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 323, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

Exemptive 2nd paragraph of 2.C.2 deleted per Amendment 20, 3/3/81.

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

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

2.C.(3)(a) Deleted per Amendment 24, 6/19/81.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN – $T_{avg} \leq 200$ °F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than that required above, immediately initiate and continue boration at ≥ 40 gpm of 2500 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to that specified in the CORE OPERATING LIMITS REPORT:

a. Within one hour after detection of an inoperable CEA(S) and at least once per 12 hours thereafter while the CEA(S) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. In accordance with the Surveillance Frequency Control Program by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area Monitor	1	Note 1	$\leq 1.5 \times 10^{-2}$ R/hr	$10^{-4} - 10^{1}$ R/hr	13
b. Containment High Range	2	1, 2, 3, & 4	Not Applicable	$1 - 10^{7}$ R/hr	18
2. PROCESS MONITORS					
a. Containment Purge and Exhaust Isolation	1	Note 3	≤ 2 x background	$10 - 10^{6}$ cpm	16
b. Control Room Ventilation Intake Duct Monitors	2	Note 2	≤ 2 x background	$10 - 10^{6}$ cpm	17,20,21

Note 1 – With fuel in the spent fuel pool or building.

Note 2 – MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

Note 3 – Applicable during:

a. PURGE of the Containment Building or,

b. Containment Building continuous ventilation operations when moving recently irradiated fuel assemblies or moving new fuel assemblies over recently irradiated fuel assemblies in the Containment Building.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 – With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 16 – With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, complete the following:

a. If moving recently irradiated fuel assemblies or moving new fuel assemblies over recently irradiated fuel assemblies within the Containment Building, secure the Containment Purge System or suspend the movement of recently irradiated fuel assemblies and movement of new fuel assemblies over recently irradiated fuel assemblies within the Containment Building.

b. If a Containment PURGE is in progress, secure the Containment Purge System.

c. If continuously ventilating the Containment Building, verify the associated SPING monitor operable or perform the applicable ACTION(s) of the Offsite Dose Calculation Manual; otherwise, secure the Containment Purge System.

ACTION 17 – In MODE 1, 2, 3, or 4, with no channels OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system (CREVS) in the recirculation mode of operation or be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 18 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the NRC within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

ACTION 19 – DELETED

ACTION 20 – In MODE 1, 2, 3, or 4 with the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, within 7 days restore the inoperable channel to OPERABLE status or initiate and maintain the CREVS in the recirculation mode of operation. Otherwise, be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 21 – During movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies with one or two channels inoperable, immediately place one OPERABLE CREVS train in the emergency recirculation mode or immediately suspend the movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area Monitor	SFCP	SFCP	SFCP	Note 1
b. Containment High Range	SFCP	SFCP Note 4	SFCP	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Purge and Exhaust Isolation	Note 2	Note 3	Note 3	In accordance with applicable Notes
b. Control Room Ventilation Intake Duct Monitors	SFCP	SFCP	SFCP Note 6	Note 5

Note 1 – With fuel in the spent fuel pool or building.

Note 2 – Within 8 hours prior to initiating Containment PURGE operations and in accordance with the Surveillance Frequency Control Program during Containment PURGE or continuous ventilation operations.

Note 3 – Within 31 days prior to initiating Containment PURGE operations and in accordance with the Surveillance Frequency Control Program during Containment continuous ventilation operations when moving recently irradiated fuel assemblies or moving new fuel assemblies over recently irradiated fuel assemblies in the Containment Building.

Note 4 – Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

Note 5 – MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies or movement of new fuel assemblies over irradiated fuel assemblies.

Note 6 - When the Control Room Ventilation Intake Duct Monitor is placed in an inoperable status solely for performance of this Surveillance, entry into associated ACTIONS may be delayed up to 3 hours.

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INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring channels less than required, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. The logarithmic neutron instrumentation, the startup channel instrumentation, and the reactor trip breaker indication are excluded from CHANNEL CALIBRATION.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by performance of a Reactor Coolant System water inventory balance in accordance with the Surveillance Frequency Control Program during steady state operation except when operating in the shutdown cooling mode*.

4.4.6.2.2 Primary to secondary leakage shall be verified to be ≤ 150 gallons per day through any one SG in accordance with the Surveillance Frequency Control Program*.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.6-1 shall be demonstrated OPERABLE by individually verifying leakage to be within its limit:

a. Prior to entering MODE 2 after each refueling outage,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

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* Not required to be performed until 12 hours after establishment of steady state operation.

MAIN STEAM ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided the isolation valve is maintained closed; otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam isolation valve shall be demonstrated OPERABLE by verifying full closure within 3 seconds when tested pursuant to the INSERVICE TESTING PROGRAM.

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SURVEILLANCE REQUIREMENTS

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤ 84 °F D.B.

b. In accordance with the Surveillance Frequency Control Program by verifying a system flow rate of 9900 cfm ± 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the system operates for at least 15 minutes.

b. In accordance with the Surveillance Frequency Control Program by verifying that on a control room high radiation signal, either actual or simulated, the system automatically isolates the control room and switches into a recirculation mode of operation.

c. By performing the required Control Room Emergency Ventilation filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

d. Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one or more of the above required monitors inoperable, immediately suspend positive reactivity additions.

AND

Suspend movement of fuel, sources, and reactivity control components within the reactor vessel.[1]

b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program, and

c. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of the movement of recently irradiated fuel assemblies or the movement of new fuel assemblies over recently irradiated fuel assemblies.

Note 1: Fuel assemblies, sources, and reactivity control components may be moved if necessary to restore an inoperable source range neutron flux monitor or to complete movement of a component to a safe condition.

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CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door is capable* of being closed,

b. A minimum of one door in each airlock is capable* of being closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed* by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Capable* of being closed by an OPERABLE containment purge and exhaust isolation system.

APPLICABILITY: During movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies within the Containment Building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies in the Containment Building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required conditions within 72 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during movement of recently irradiated fuel assemblies or movement of new fuel assemblies over recently irradiated fuel assemblies in the Containment Building.

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* Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. Administrative controls shall ensure that appropriate personnel are aware that when containment penetrations, including both personnel airlock doors and/or the equipment door are open, a specific individual(s) is designated and available to close the penetration following a required evacuation of containment, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and/or the equipment door be capable of being quickly removed.

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6.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. The volumetric examination per Regulatory Position C.4.b.1 will be performed on approximately 10-year intervals.

6.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Decay Tanks, the quantity of radioactivity contained in the Waste Gas Decay Tanks, and the quantity of radioactivity contained in unprotected temporary outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology contained in Revision 2 of NUREG-0800, "Standard Review Plan," Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure," Revision 0, July 1981. The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Decay Tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion),

b. A surveillance program to ensure that the quantity of radioactivity contained in each Waste Gas Decay Tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents, and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 323 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By application dated December 16, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19350B324), as supplemented by letters dated June 17, 2020, and September 22, 2020 (ADAMS Accession Nos. ML20169A558 and ML20266G333, respectively), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 2 (ANO-2).

The proposed changes would revise several TS requirements by the addition, deletion, or relocation of certain TS limiting conditions for operation (LCOs), Actions, and surveillance requirements (SRs). Relocated TSs would be placed in the ANO-2 Technical Requirements Manual (TRM) or the associated TS Bases. In the license amendment request (LAR) dated December 16, 2019, the licensee stated that the proposed TS changes are intended to provide added consistency between the ANO-2 TSs and the improved Standard Technical Specifications (STSs) of NUREG-1432, “Standard Technical Specifications – Combustion Engineering Plants,” Revision 4.

The supplemental letters dated June 17, 2020, and September 22, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on February 25, 2020 (85 FR 10733).

2.0 REGULATORY EVALUATION

The proposed TS changes evaluated in this safety evaluation (SE) are:

- TS 3.1.1.3, “Boron Dilution”
- TS 3.3.3.1, “Radiation Monitoring Instrumentation”
- TS 3.3.3.5, “Remote Shutdown Instrumentation”

- TS 3.4.6.2, “Reactor Coolant System Operational Leakage”
- TS 3/4.7.2, “Steam Generator Pressure/Temperature Limitation”
- TS 3/4.7.5, “Flood Protection”
- TS 3/4.7.9, “Sealed Source Contamination”
- TS 3/4.7.12, “Spent Fuel Pool Structural Integrity”
- TS 3.9.3.a, “Decay Time and Spent Fuel Storage”
- TS 3.9.5, “Communications”
- TS 3.9.6, “Refueling Machine Operability”
- TS 3.9.7, “Crane Travel – Spent Fuel Pool Building”
- TS 3.11.1, “Liquid Holdup Tanks”
- TS 3.11.2, Gas Storage Tanks”
- TS 3.11.3, “Explosive Gas Mixture”
- TS 6.5.8, “Explosive Gas and Storage Tank Radioactivity Monitoring Program”
- Editorial Changes

2.1 System Descriptions

2.1.1 Boron Dilution

Boron dilution is required to provide adequate mixing, prevent stratification, and ensure that reactivity changes will be gradual during boron concentration reductions in the reactor coolant system (RCS). The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

2.1.2 Radiation Monitoring Instrumentation

The main steam line (MSL) radiation monitoring system consists of remote detector assemblies, one attached on each MSL between the containment penetration and the MSL safety valves. The MSL radiation monitors detect radioactivity transported through either the MSL during events that involve steam generator (SG) tube leakage. The MSL radiation monitoring system is an Eberline Model radiation monitoring system.

The detectors provide a signal to the instrument panel (indicator) in the control room and to the safety parameter display system. The monitors detect SG tube leakage and monitor radioactive effluent releases through the steam safety relief valves during accident conditions.

2.1.3 Remote Shutdown Instrumentation

The remote shutdown instrumentation ensures that sufficient capability is available to permit plant operators to shutdown and maintain the plant in hot standby (Mode 3) from locations outside of the control room and, in conjunction with procedures, reach cold shutdown (Mode 5) conditions if necessary. This capability is required in the event control room habitability is lost.

2.1.4 RCS Operational Leakage

The RCS leakage detection systems required by ANO-2 TS 3.4.6.2 are provided to monitor and detect leakage from the reactor coolant pressure boundary (RCPB).

2.1.5 Steam Generator Pressure/Temperature Limitation

The limitation on SG pressure and temperature ensures that the pressure-induced stresses in the SGs do not exceed the maximum allowable fracture toughness stress limits. The limitations in ANO-2 TS 3/4.7.2 are based on the parameters sufficient to prevent brittle fracture.

2.1.6 Flood Protection

The requirements for flood protection in ANO-2 TS 3/4.7.5 are provided to ensure that facility protective actions will be taken in the event of flood conditions.

2.1.7 Sealed Source Contamination

The limitation specified in ANO-2 TS 3/4.7.9 on removable contamination for sources requiring leak testing, including alpha emitters, is provided to ensure that leakage from byproduct, source and special nuclear material sources will not exceed allowable intake values.

2.1.8 Spent Fuel Pool Structural Integrity

The specified structural integrity inspections of the spent fuel pool (SFP), as called for in ANO-2 TS 3/4.7.12, are required to be performed to ensure that the SFP remains safe for use and that it will adequately resist the imposed loadings.

2.1.9 Decay Time and Spent Fuel Storage

The minimum requirement in ANO-2 TS 3.9.3.a for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel (RPV) is provided to ensure that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products.

2.1.10 Communications

The requirement for communications capability specified in ANO-2 TS 3.9.5 is provided to ensure that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during core alterations.

2.1.11 Refueling Machine Operability

The refueling machine in the reactor containment is a traveling bridge and trolley, which is located above the refueling canal and rides on rails set in the concrete on each side of the refueling canal. Motors on the bridge and trolley position the machine over each fuel assembly location within the reactor core or fuel transfer carrier. The hoist assembly contains a grappling device which, when rotated by the actuator mechanism, engages the fuel assembly to be removed. The hoist assembly and grappling device are raised and lowered by a cable attached to the hoist winch. After the fuel assembly has been raised into the refueling machine, the refueling machine transports the fuel assembly to its designated location.

The refueling machine moves fuel assemblies in and out of the core and between the core and the transfer equipment. The fuel transfer equipment tilts fuel assemblies from the vertical position to the horizontal position, shuttles them from the refueling pool in the containment through the containment wall into the SFP, and returns them to the vertical position. The spent

fuel handling machine handles fuel between the transfer equipment and the fuel storage racks in the SFP.

The operability requirements for the refueling machine in ANO-2 TS 3.9.6 are provided to ensure that: (1) the refueling machine will be used for movement of control element assemblies (CEAs) with fuel assemblies, and that it has sufficient load capacity to lift a fuel assembly, and (2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

2.1.12 Crane Travel – SFP Building

The SFP building crane is a traveling bridge and trolley, which is located above the SFP and rides on rails set in the concrete on each side of the SFP. Motors on the bridge and trolley position the machine over each spent fuel storage slot in the SFP. ANO-2 TS 3.9.7 is provided to restrict movement of loads in excess of the nominal weight of a typical spent fuel assembly and lifting equipment to ensure that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array.

2.1.13 Liquid Holdup Tanks

Liquid radioactive waste holdup tanks are used to store liquid radioactive waste. ANO-2 TS 3.11.1 limits potential radiological exposures to the public by limiting the radioactive content stored in unprotected temporary outdoor liquid storage tanks to 10 curies; excluding tritium and dissolved or entrained noble gases. Unprotected tanks are those tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or those tanks which do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

2.1.14 Gas Storage Tanks

Radioactive gas storage tanks are used to hold up radioactive gas for reduction by radioactive decay prior to release in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36a and TS limits. ANO-2 TS 3.11.2 is provided to limit the quantity of radioactivity contained in the gaseous radioactive waste storage tanks to 82,400 curies of noble gases (considered as Xenon-133).

2.1.15 Explosive Gas Mixture

Gaseous radioactive waste storage tanks have the potential to contain an explosive mixture of hydrogen and oxygen. To preclude potential explosive mixtures that could result in an offsite release, TS 3.11.3 places limits on the oxygen and hydrogen concentrations, to prevent an explosion that could rupture a gaseous radioactive waste storage tank.

2.1.16 Proposed Explosive Gas and Storage Tank Radioactivity Monitoring Program

New TS 6.5.8 would limit the quantity of radioactivity in both the gaseous radioactive waste storage tanks and in the unprotected temporary outdoor liquid storage tanks. The gaseous radioactivity quantities would be determined following the methodology in NRC Branch Technical Position (BTP) Effluent Treatment Systems Branch (ETSB 11-5). The liquid radwaste

quantities would be determined in accordance with the licensee-controlled Offsite Dose Calculation Manual (ODCM)[1].

2.2 Proposed TS Changes

The licensee's proposed TS revisions are summarized below. The proposed TS page markups are provided in Attachment 1 of the enclosures to the licensee's letters dated December 16, 2019, and September 22, 2020.

2.2.1 Proposed Revision to TS 3.1.1.3

The LCO and SR for boron dilution in ANO-2 TS 3/4.1.1.3 would be deleted.

2.2.2 Proposed Revision to TS 3.3.3.1

ANO-2 TS 3.3.3.1, Instrument 2.c from Table 3.3-6, its associated Action 19, SRs listed in Table 4.3-3, and TS Bases would be relocated to the ANO-2 TRM.

2.2.3 Proposed Revision to TS 3.3.3.5

ANO-2 TS LCO 3.3.3.5 and its Action statement would be revised by relocating TS Table 3.3-9, "Remote Shutdown Monitoring Instrumentation," to the TS 3.3.3.5 Bases. Also, SR 4.3.3.5 would be revised by replacing, "operations at the frequencies shown in Table 4.3-6," with the language, "in accordance with the Surveillance Frequency Control Program." The logarithmic neutron instrumentation, the startup channel instrumentation, and the reactor trip breaker indication are excluded from CHANNEL CALIBRATION."

2.2.4 Proposed Revision to TS 3.4.6.2

ANO-2 TS 3.4.6.2, SR 4.4.6.2.1.b, monitoring the reactor head flange leakoff temperature, would be deleted and SR 4.4.6.2.1.a would be revised accordingly.

2.2.5 Proposed Revision to TS 3/4.7.2

ANO-2 TS 3/4.7.2 and its associated TS Bases would be relocated to the ANO-2 TRM.

2.2.6 Proposed Revision to TS 3/4.7.5

ANO-2 TS 3/4.7.5 and its associated TS Bases would be relocated to the ANO-2 TRM.

2.2.7 Proposed Revision to TS 3/4.7.9

ANO-2 TS 3/4.7.9 and its associated TS Bases would be deleted.

2.2.8 Proposed Revision to TS 3/4.7.12

ANO-2 TS 3/4.7.12 and its associated TS Bases would be relocated to the ANO-2 TRM.

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[1] The ANO ODCM is included as an attachment to the ANO Annual Radioactive Release Report for 2019 (ADAMS Accession No. ML20118C140).

2.2.9 Proposed Revision to TS 3.9.3.a

ANO-2 TS 3.9.3.a and its associated TS Bases would be relocated to the ANO-2 TRM.

2.2.10 Proposed Revision to TS 3.9.5

ANO-2 TS 3.9.5 and its associated TS Bases would be deleted.

2.2.11 Proposed Revision to TS 3.9.6

ANO-2 TS 3.9.6 and its associated TS Bases would be relocated to the ANO-2 TRM.

2.2.12 Proposed Revision to TS 3.9.7

ANO-2 TS LCO 3.9.7 and its associated TS Bases would be relocated to the ANO-2 TRM.

2.2.13 Proposed Revision to TS 3.11.1

The ANO-2 TS 3/4.11.1 requirements would be relocated to the new TS Explosive Gas and Storage Tank Radioactivity Monitoring Program (i.e., new TS 6.5.8). The associated TS Bases would be deleted.

2.2.14 Proposed Revision to TS 3.11.2

ANO-2 TS 3/4.11.2 would be relocated to the new TS Explosive Gas and Storage Tank Radioactivity Monitoring Program. The associated TS Bases would be deleted.

2.2.15 Proposed Revision to TS 3.11.3

ANO-2 TS 3/4.11.3 would be relocated to new TS Explosive Gas and Storage Tank Radioactivity Monitoring Program. The associated TS Bases would be deleted.

2.2.16 Proposed Revision to TS 6.5.8

A new program entitled "Explosive Gas and Storage Tank Radioactivity Monitoring Program" (TS 6.5.8) would be added to TS Section 6.0, "Administrative Controls," for ANO-2.

2.2.17 Proposed Editorial Changes

The licensee proposed the following editorial changes to the ANO-2 TSs. TS pages 3/4 1-3, 3/4 3-27, 3/4 3-36, 3/4 7-10, 3/4 7-18, 3/4 9-2, and 3/4 9-4 would be renumbered and/or annotated with the next page number. TS pages 3/4 3-26a, 3/4 7-11 3/4 7-12, and 3/4 7-13, would be deleted.

2.3 Regulatory Requirements and Guidance

2.3.1 Regulatory Requirements

The NRC regulatory requirements in 10 CFR 50.36(a)(1) requires an applicant for an operating license to include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must include in the application, a "summary statement of the

bases or reasons for such specifications, other than those covering administrative controls." However, per 10 CFR 50.36(a)(1), these TS Bases "shall not become part of the technical specifications."

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c), and include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; (5) administrative controls. However, the regulation does not specify the particular requirements such as format and content to be included in a plant's TS categories.

As required by 10 CFR 50.36(c)(2)(i), the TSs will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action (also known as Required Actions or Actions) permitted by the TSs until the condition can be met.

As stated in 10 CFR 50.36(c)(3), "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

On July 22, 1993, the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement) (58 FR 39132), which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 on July 19, 1995 (60 FR 36953). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1 — Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 — A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3 — A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4 — A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Appendix A, "General Design Criteria [GDC][2] for Nuclear Power Plants," to 10 CFR Part 50, provides the minimum necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. The following criteria apply to this LAR:

- GDC 13, "Instrumentation and control," states:

  Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

- GDC 19, "Control room," states:

  A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

- GDC 25, "Protection system requirements for reactivity control malfunctions," states:

  The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

- GDC 26, "Reactivity control system redundancy and capability," states:

  Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operation occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

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[2] ANO-2 was originally designed to comply with the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits," published in July 1967. Sections 3.1.1 through 3.1.6 of the ANO-2 Safety Analysis Report provide a comparison with the Atomic Energy Commission GDC published as Appendix A to 10 CFR 50 in 1971. Each criterion is followed by a summary discussion of the design and procedures that are intended to meet the design objectives reflected in the criterion.

> The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

As discussed in part, in the *Federal Register* notice regarding the Final Rule for Technical Specifications under 10 CFR Part 50, dated July 19, 1995 (60 FR 36953):

> LCOs that do not meet any of the criteria, and their associated actions and surveillance requirements, may be proposed for relocation from the technical specifications to licensee-controlled documents, such as the FSAR [Final Safety Analysis Report]. The criteria may be applied to either standard or custom technical specifications.

As discussed, in part, in the Final Policy Statement (58 FR 39138; July 22, 1993):

> When licensees submit amendment requests based on this Policy Statement, they should identify the location of and controls for the technical and administrative requirements of the relocated requirements. The NRC staff will carefully review these submittals to ensure the accountability and the acceptability of controls for each relocated requirement. Many of the requirements will be relocated to the FSAR and will be enforceable through 10 CFR 50.59. Other requirements will be relocated to more appropriate documents (e.g., Security Plan, QA [Quality Assurance] Plan) and controlled by the applicable regulatory requirements. The adequacy of controls for relocated requirements which do not fit in the above categories will be reviewed and approved by the NRC staff on a case-by-case basis to determine, among other things, whether an enforceable control method will need to be established.

As discussed in the LAR dated December 16, 2019, the licensee proposed to relocate certain TSs from the ANO-2 TSs, and place them in the TRM, which is considered part of the ANO-2 Safety Analysis Report (SAR) (ADAMS Accession No. ML19282B426), or the associated TS Bases, which is controlled by ANO-2 TS 6.5.14, “Technical Specifications (TS) Bases Control Program,” and therefore, both would be controlled in accordance with the requirements of 10 CFR 50.59, “Changes, tests, and experiments.”

The quantity of radioactive material contained in liquid radioactive waste storage tanks is restricted to provide assurance that, in the event of an uncontrolled release of the contents of the tanks, the resulting concentrations would be less than the limits of 10 CFR Part 20, “Standards for Protection against Radiation,” Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations from Release to Sewerage,” Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

2.3.2 Regulatory Guidance

NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements" (ADAMS Accession No. ML051400209), dated November 1980, Enclosure 3, "Clarification of TMI Action Plan Requirements," Section II.F.1, "Noble Gas Effluent Monitor," Attachment 1, includes, in part:

(1) A recommendation that the noble gas effluent monitors be installed with an extended range designed to function during accident conditions as well as during normal operating conditions, and

(2) Reference to NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (ADAMS Accession No. ML090060030), dated July 1979, which recommends, in part, that plants provide the high range radiation monitors for noble gases in plant effluent lines and a high range radiation monitor in the containment. The NRC staff notes that NUREG-0737 does not specifically require the identification of MSL radiation monitors as meeting the requirement for noble gas effluent monitors.

The NRC staff's guidance for review of TSs is in Section 16, Technical Specifications, of NUREG-0800, Revision 3, dated March 2010 (ADAMS Accession No. ML100351425). The guidance specifies that the NRC staff review should determine whether the content and format are consistent with the applicable STSs. Where TS provisions depart from the reference TSs, the NRC staff determines whether proposed differences are justified by uniqueness in plant design or other considerations.

Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," Revision 0, dated July 1981, provides guidelines on postulated radioactive releases due to a radioactive waste gas system leak or failure. This BTP is located in NUREG-0800, Section 11.3, "Gaseous Waste Management System," Revision 2, dated July 1981 (ADAMS Accession No. ML052350105).

NUREG-1432, Revision 4.0, includes Volume 1, "Specifications," and Volume 2, "Bases" dated April 2012 (ADAMS Accession Nos. ML12102A165 and ML12102A169, respectively). The format of the STSs addresses the categories required by 10 CFR 50.36, which comprise six sections covering the areas of: definitions, safety limits and limiting safety system settings, LCOs, SRs, design features, and administrative controls. Section 3.3.11 of Volume 1 of NUREG-1432, the "Reviewer's Notes" for Table 3.3.11-1, "Post Accident Monitoring Instrumentation," states:

> Table 3.3.11-1 shall be amended for each unit as necessary to list:
>
> 1. All Regulatory Guide 1.97, Type A instruments and
> 2. All Regulatory Guide 1.97, Category I, non-Type A instruments specified in the unit's Regulatory Guide 1.97, Safety Evaluation Report.

Regulatory Guide (RG) 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," dated May 1983 (ADAMS Accession No. ML003740282), describes a method acceptable to the NRC staff for complying with the NRC's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power

plant. These post-accident instruments provide the essential information needed by the operator to determine if the plant safety functions are being performed. RG 1.97 provides NRC guidance on acceptable methods to identify the importance and reliability of plant instrumentation needed for post-accident monitoring. RG 1.97 provides guidance for determining what instrumentation is needed to perform these functions and defines the relative importance of variables by a classification system using Type A through Type E, and as Category 1 through Category 3; with Type A and Category 1 variables being the most important variables.

The MSL monitors are included in RG 1.97, Revision 3, Table 3 as a Category 2, Type E variable. In accordance with NUREG-1432, Table 3.3.11-1, "Post Accident Monitoring Instrumentation," only Type A instruments and Category 1, non-Type A instruments need to be included in TSs. Since the MSL monitors are not a Type A or Category 1, non-Type A instruments, the MSL monitors are not instrumentation required to be included in TSs.

3.0 TECHNICAL EVALUATION

3.1 Proposed Revision to TS 3.1.1.3

The basis for the licensee to delete ANO-2 TS 3/4.1.1.3, as provided in the LAR, is that the minimum required reactor coolant flow rate through the RCS for the planned boron dilution operation (i.e., greater than or equal to (≥) 2000 gallons per minute (gpm)) can be assured under the other LCOs 3.4.1.1, 3.4.1.2, 3.4.1.3, and 3.9.8.1. The question of whether each LCO, associated Action, and SR will meet the original requirements as specified in TS 3/4.1.1.3 for boron dilution is reviewed and evaluated below.

3.1.1 LCO 3.4.1.1, Action and SR 4.4.1.1

ANO-2 TS LCO 3.4.1.1 requires both reactor coolant loops and both reactor coolant pumps (RCPs) in each loop to be in operation for Modes 1 and 2. The licensee stated that the design capacity of each RCP is 80,000 gpm (ANO-2 SAR Table 1.3-1), much greater than the 2000 gpm stipulated in TS LCO 3.1.1.3. Therefore, the licensee considered that a planned boron operation under ANO-2 TS LCO 3.4.1.1 would meet the boron dilution LCO (i.e., LCO 3.1.1.3) and proposed to delete ANO-2 LCO 3.1.1.3 for Modes 1 and 2 due to the available LCO 3.4.1.1. Since there are four RCPs required to be OPERABLE during Modes 1 and 2 and each RCP has a design capacity of 80,000 gpm, reasonable assurance exists that a reactor coolant flow rate greater than 2000 gpm through the RCS for the planned boron dilution is achievable during Modes 1 and 2.

The Action statement for ANO-2 TS LCO 3.4.1.1 describes that if fewer than the LCO-required RCPs are in operation, a unit trip and entry into Mode 3 shall occur within 1 hour of the loss of any RCPs. However, a planned boron dilution may be in progress when the action for LCO 3.4.1.1 is triggered (i.e., loss of RCPs), such that the reactor coolant flow rate through the RCS may be less than 2000 gpm. Therefore, the NRC staff issued Request for Additional Information (RAI)-SNSB-1 to obtain assurance from the licensee that the planned boron dilution under such conditions would be suspended immediately, consistent with the Action for ANO-2 TS LCO 3.1.1.3. In its supplement dated June 17, 2020, the licensee responded that the core protection calculators (CPCs) will become active when the reactor core power is greater than 10^{-2} percent (i.e., 0.01 percent) of rated power. The CPCs will automatically trip the reactor if less than four RCPs are operating during Modes 1 and 2. The NRC staff found the response

acceptable because the automatic trip of the reactor by the CPCs will serve the purpose of ensuring immediate measures to manage reactivity.

The SR for ANO-2 TS LCO 3.4.1.1 (SR 4.4.1.1) requires that the reactor coolant loops be verified to be in operation and circulating reactor coolant at least once per 12 hours. However, in contrast, the SR for ANO-2 TS LCO 3.1.1.3 requires the flow rate of reactor coolant through the RCS shall be determined to be ≥ 2000 gpm within 1 hour prior to the start of and at least once per hour during a planned boron dilution. If a planned boron dilution operates under TS 3/4.4.1.1, similar SR requirements for TS LCO 3.1.1.3 should be implemented, or justification provided for the increase in surveillance intervals. In RAI-SNSB-2, the NRC staff requested the licensee to address this issue. In its supplemental letter dated June 17, 2020, the licensee indicated that the RCPs' speeds and mass flow rates are two of the parameters under the CPCs' continuous monitoring, ensuring that the intent of the SR for TS LCO 3.1.1.3 is met. The NRC staff found this response acceptable.

3.1.2 LCO 3.4.1.2 and SR 4.4.1.2

ANO-2 TS LCO 3.4.1.2.a requires both reactor coolant loops to be operable, while each loop shall have at least one associated RCP in operation during Mode 3. The licensee stated that the design capacity for an RCP is 80,000 gpm (ANO-2 SAR Table 1.3-1), much greater than the 2000 gpm required in ANO-2 TS LCO 3.1.1.3. Therefore, the licensee considered that a planned boron operation under TS LCO 3.4.1.2.a would meet boron dilution LCO 3.1.1.3 and proposed to delete LCO 3.1.1.3 for Mode 3 due to the available TS LCO 3.4.1.2.a. Since there are at least two RCPs required to be operable during Mode 3 and each RCP has a design capacity of 80,000 gpm, reasonable assurance exists that a reactor coolant flow rate greater than 2000 gpm through the RCS for the planned boron dilution is achievable during Mode 3.

The Action statement for ANO-2 TS LCO 3.4.1.2.a describes that if less than the LCO-required reactor coolant loops are operable, the operators must act to either restore the required loops to operable status within 72 hours or be in hot shutdown within the next 12 hours. If there is at least one reactor coolant loop operable when entering into this action, the reactor coolant flow rate of greater than 2000 gpm through the RCS for the planned boron dilution is still achievable. If no reactor coolant loops are operable, then the licensee will enter the TS LCO 3.4.1.2.b Action statement, as discussed below.

ANO-2 TS LCO 3.4.1.2.b requires at least one reactor coolant loop to be in operation, while the loop shall have at least one associated RCP operable during Mode 3. The licensee stated that the design capacity for one RCP is 80,000 gpm (ANO-2 SAR Table 1.3-1), much greater than the 2000 gpm required in TS LCO 3.1.1.3. Therefore, the licensee considered that a planned boron operation under TS LCO 3.4.1.2.b would meet the boron dilution LCO (i.e., TS LCO 3.1.1.3) and proposed to delete the TS LCO 3.1.1.3 for Mode 3 due to the available TS LCO 3.4.1.2.b. Since there is at least one RCP required to be operable under TS LCO 3.4.1.2.b and one RCP has a design capacity of 80,000 gpm, reasonable assurance exists that a reactor coolant flow rate greater than 2000 gpm through the RCS for the planned boron dilution is achievable under TS LCO 3.4.1.2.b.

The Action statement for TS LCO 3.4.1.2.b describes that if no reactor coolant loop is in operation, then any planned boron dilution in operation under TS LCO 3.4.1.2.b will be suspended and corrective actions will immediately be initiated to return the required loop to operation. Since the action for TS LCO 3.4.1.2.b meets the intent of action for the boron dilution

LCO, the NRC staff finds it acceptable to delete the Action for the boron dilution LCO for Mode 3.

ANO-2 SR 4.4.1.2 requires that at least one reactor coolant loop be verified to be in operation and circulating reactor coolant at least once per 12 hours. However, in contrast, the SR for ANO-2 TS LCO 3.1.1.3 requires that the flow rate of reactor coolant through the RCS shall be determined to be ≥ 2000 gpm within 1 hour prior to the start of and at least once per hour during a planned boron dilution. If a planned boron dilution operates under ANO-2 TS 3/4.4.1.2, similar SRs for TS LCO 3.1.1.3 should be implemented, or justification provided for the increase in the surveillance interval. In its supplement dated June 17, 2020, the licensee stated the following:

> SR 4.4.1.2 requires at least one RCP to be in operation in Mode 3 (RCS temperature ≥ 300 °F [degrees Fahrenheit]). As stated in Entergy's Reference 1 license amendment request (LAR), the flowrate of one RCP is significantly greater than the 2000 gpm minimum flow assumed for performance of boron dilution activities. The loss of an RCP would result in Control Room alarm and, if the only RCP operating at the time, rising RCS temperature. Operations personnel would recognize the loss of RCS flow within moments, if not immediately, and the ACTIONS of LCO 3.4.1.2 entered at that time, which require boron dilutions to be suspended. Since loss of RCS flow in Mode 3 will be promptly recognized and appropriate action taken, it is not necessary to stipulate a one-hour flow verification requirement within SR 4.4.1.2.

The NRC staff reviewed the licensee's response and finds it acceptable because there are means available for the operator to promptly recognize the loss of RCS flow and meet the intent of the original boron dilution surveillance (SR 4.1.1.3).

3.1.3 LCO 3.4.1.3 and SR 4.4.1.3

ANO-2 TS LCO 3.4.1.3.a requires at least two of the coolant loops from the reactor coolant loops or shutdown cooling (SDC) loops be operable, while at least one of the cooling loops shall be in operation during Modes 4 and 5. The licensee considered that a planned boron operation under TS LCO 3.4.1.3.a would meet the boron dilution LCO (i.e., LCO 3.1.1.3) and proposed to delete the ANO-2 TS LCO 3.1.1.3 for Modes 4 and 5 due to the available TS LCO 3.4.1.3.a. Since there is at least one associated coolant pump (RCP) operable during Modes 4 and 5, each with a design capacity of 80,000 gpm, there is reasonable assurance that a reactor coolant flow rate greater than 2000 gpm through the RCS for the planned boron dilution is achievable during Modes 4 and 5.

The licensee also stated that when a low pressure safety injection (LPSI) pump is used for SDC, the relevant plant procedures will require the LPSI pump to operate at flows greater than (>) 2000 gpm in order to avoid cavitation, vibration, and bearing wear. In addition, when a containment spray pump is used for SDC, the relevant plant procedures require maintaining a flow rate between 2300 and 2500 gpm. Therefore, there is reasonable assurance that a reactor coolant flow rate greater than 2000 gpm through the RCS for the planned boron dilution is achievable during SDC operation.

Based on the above, the NRC staff finds that the licensee's proposed change to delete ANO-2 TS LCO 3.1.1.3 for Modes 4 and 5 acceptable because the use of an RCP or SDC loop will provide the minimum required reactor coolant flow rate of 2000 gpm through the RCS for boron dilution during Modes 4 and 5.

The Action statement for LCO 3.4.1.3.a prescribes that if less than two coolant loops are in operation, then a corrective action will be taken immediately to return the required coolant loops to operable status and initiate action to make at least one SG available for decay heat removal via natural circulation. Since the Action for LCO 3.4.1.3.a meets the intent of the action for the boron dilution LCO, the NRC staff finds it acceptable to delete the Action for the boron dilution LCO for Modes 4 and 5.

ANO-2 LCO 3.4.1.3.b requires at least one of the reactor coolant loops or SDC loops be in operation while the operable loop shall have at least one associated pump operable during Modes 4 and 5. Since there is either one RCP in operation with a design capacity of 80,000 gpm or one SDC loop in operation with a required operation flow rate of > 2000 gpm, reasonable assurance exists that a reactor coolant flow rate greater than 2000 gpm through the RCS for the planned boron dilution is achievable during Modes 4 and 5.

The Action statement for ANO-2 TS LCO 3.4.1.3.b describes that if no coolant loop is in operation, then any planned boron dilution operation will be suspended, and corrective action immediately initiated to return the required loop to operation. Since the Action for TS LCO 3.4.1.3.b meets the intent of the action for the boron dilution LCO, the NRC staff finds it acceptable to delete the Action for the boron dilution LCO for Modes 4 and 5.

ANO-2 SRs 4.4.1.3.1 through 4.4.1.3.4 require that the associated SDC loop(s), coolant pump(s), SG(s) or RCP loop(s) be verified to be in operation and circulating reactor coolant per the inservice testing program, once per 7 days and at least once per 12 hours, respectively. However, in contrast, the SR for ANO-2 TS LCO 3.1.1.3 requires the flow rate of reactor coolant through the RCS to be determined to be ≥ 2000 gpm within 1 hour prior to the start of and at least once per hour during a planned boron dilution. If a planned boron dilution operates under TS 3/4.4.1.3, similar SRs for TS LCO 3.1.1.3 should be implemented, or justification provided for the increase in surveillance intervals. The NRC staff issued RAI-SNSB-2, requesting the licensee to address this issue. In its supplement dated June 17, 2020, the licensee responded stating that any loss of an RCP or SDC loop(s) (i.e., the purpose for surveillance) would be readily recognized by operations personnel via control room alarm. The NRC staff determined that the licensee's response was acceptable and that because any loss of an RCP or SDC loop would be recognized immediately due to the control room alarm, once per hour verification of RCS flowrate is not necessary. Therefore, the NRC staff concludes that this change is acceptable.

3.1.4 LCO 3.9.8.1 and SR 4.9.8.1

ANO-2 TS LCO 3.9.8.1 requires that at least one SDC loop be in operation during Mode 6. The licensee evaluated and concluded that a planned boron operation under TS LCO 3.9.8.1 would meet the boron dilution LCO (i.e., LCO 3.1.1.3) for Mode 6 and proposed to delete TS LCO 3.1.1.3 for Mode 6 due to the available TS LCO 3.9.8.1. The licensee stated that when a LPSI pump is used for SDC, the relevant plant procedures will require the LPSI pump to operate at flow rates > 2000 gpm in order to avoid cavitation, vibration, and bearing wear. In addition, when a containment spray pump is used for SDC, the relevant plant procedures require maintaining a flow rate between 2300 and 2500 gpm. The NRC staff finds the licensee's proposed deletion of TS LCO 3.1.1.3 for Mode 6, due to the availability of SDC required by LCO 3.9.8.1, acceptable because the use of SDC will provide the reactor coolant flow rate of greater than 2000 gpm through the RCS, as required for boron dilution.

The Action statement for LCO 3.9.8.1.a describes that if no SDC loop is in operation, then the planned boron dilution operation under LCO 3.9.8.1 will be suspended. Since this Action meets the intent of the Action for the boron dilution LCO, the NRC staff finds it acceptable to delete the Action for the boron dilution LCO for Mode 6.

ANO-2 SR 4.9.8.1 requires that one SDC loop be verified to be in operation and circulating reactor coolant at a flow rate of ≥ 2000 gpm at least once per 24 hours. However, in contrast, the SR for TS LCO 3.1.1.3 (i.e., SR 4.1.1.3) requires that the flow rate of reactor coolant through the RCS be determined to be ≥ 2000 gpm within 1 hour prior to the start of and at least once per hour during a planned boron dilution. If a planned boron dilution operates under TS 3/4.9.8.1, similar SRs for TS LCO 3.1.1.3 should be implemented, or justification provided for the increase in surveillance interval. Based on the response provided by the licensee for RAI-SNSB-2, the NRC staff found that this is not an issue because any loss of an SDC loop(s) (i.e., the purpose for the surveillance) would be readily recognized by operations personnel via a control room alarm. Therefore, a once per hour verification of RCS flowrate is not necessary.

3.1.5 Summary of NRC Staff Evaluation of Proposed Revision to TS 3.1.1.3

The NRC staff reviewed the licensee's evaluation and RAI responses concerning the proposed deletion of boron dilution TS 3/4.1.1.3 and concludes that adequate justification is provided, and the concerns raised during review have been resolved. Based on the above evaluation, the NRC staff finds that the proposed TS change concerning boron dilution will continue to meet the requirements of GDCs 25 and 26. Specifically, operating the boron system with a rate of reactivity change resulting from planned boron dilution that is readily recognized by the operations personnel will assure that acceptable fuel design limits will not be exceeded. Therefore, the NRC staff finds that this change is acceptable.

3.2 Proposed Revision to TS 3.3.3.1

3.2.1 Background

In accordance with Generic Letter (GL) 83-37, "NUREG-0737 Standard Technical Specifications," dated November 1, 1983, the licensee requested changes to ANO Units 1 and 2 TSs by letter dated September 28, 1992 (ADAMS Accession No. ML20106C897), as supplemented by letter dated January 26, 1993 (ADAMS Legacy Library Accession No. 9302020186), to satisfy the guidance in Item II.F.1.1 of NUREG-0737 to have noble gas effluent monitors monitoring the effluent pathway through the steam dump valves. The changes include LCOs and SRs for each unit's MSL radiation monitors.

In response to the requirements of NUREG-0737, the NRC approved the addition of MSL radiation monitors as noble gas effluent monitors into the ANO-2 TS, by NRC letter dated March 6, 1993 (ADAMS Accession No. ML021270024). In this letter, the NRC staff stated, in part:

> In response to the GL 83-37 request for licensees to submit proposed TSs as appropriate for NUREG-0737 items, the licensee proposed by letter dated March 16, 1984, that no TS changes for NUREG-0737, Item II.F.1.1 (noble gas monitors) were necessary since TS requirements for noble gas monitors were previously submitted under the Radiological Effluent Technical Specifications (RETs) effort. RETs were subsequently approved by Amendments 88 and 60 to the ANO-1 and ANO-2 TSs, respectively. However, the RETs did not include TSs for the Main Steam Line

Radiation Monitors. Accordingly, by letter dated September 28, 1992, the licensee submitted proposed TSs for these monitors.

As a result of the NRC staff's approval of the licensee's September 1992 request, SRs for Instrument 2.c., now exist in the current ANO-2 TS 3/4.3.3, "Monitoring Instrumentation" Radiation Monitoring Instrumentation LCO.

3.2.2 Licensee's Methodology

In the LAR, the licensee used information from plant licensing documents, NRC regulatory requirements, the guidance contained within RG 1.97, the analyses contained in the ANO-2 SAR, and the STSs contained in NUREG-1432. The licensee reviewed and compared the requirements and guidance from each of these documents to identify whether the affected MSL radiation instruments in ANO-2 TS Tables 3.3-6 and 4.3-3 need to be considered as Type A or non-Type A but Category 1, and requested to revise ANO-2 TS 3.3.3.1.

3.2.3 Technical Evaluation

The NRC staff reviewed ANO-2 SAR, Section 7.5.2.5, "Analysis of Post-Accident Monitoring Instrumentation," and Table 7.5-3, "R.G. 1.97 Post Accident Monitoring Variables," to evaluate the variables and criteria of the MSL radiation monitors in Table 3.3-6 of the ANO-2 TSs.

The NRC staff also reviewed the licensee's proposed TS revision using the approach described below to determine whether the licensee's proposed revision of ANO-2 TS LCO 3.3.3.1 satisfies the regulatory requirements and the review criteria in Section 2.3 of this SE.

Based on the four criteria of 10 CFR 50.36(c)(2)(ii), the NRC staff reviewed the LAR and its references to verify whether the required functionality for the MSL radiation monitors meets one or more LCO requirements described within these four criteria. The NRC staff's assessment of the MSL radiation monitor functionality as it relates to the four criteria are addressed in Section 3.2.3.2 of this SE.

Based on the requirements of GDC 13 described in the ANO-2 SAR, the NRC staff reviewed the LAR to ensure that the proposed change still conforms with the ANO-2 licensing basis to include sufficient instrumentation to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions as appropriate.

Based on the guidance in NUREG-0737, the NRC staff also evaluated whether any changes to TSs associated with accident monitoring instrumentation are needed.

In addition, the NRC staff considered the guidance of RG 1.97, Revision 3, to evaluate whether the MSL radiation monitor falls under Type A or Category 1 of RG 1.97.

3.2.3.1 Evaluation of the Proposed Relocation of Instrument 2.c. from TS Table 3.3-6 to the TRM

The licensee proposed to relocate Instrument 2.c. from TS Table 3.3-6 to the TRM.

Instrument	Minimum Channel Operable	Applicable Modes	Alarm/Trip Setpoint	Measurement Range	Action
2. Process Monitors c. Main Steam Line Radiation Monitors	1/Steam Line	1, 2, 3, & 4	Not Applicable	$10^{-4} – 10^{4}$ mR/hr	19

The acceptance of the proposed relocation is based on an evaluation as to whether this function meets any one of the four criteria of 10 CFR 50.36(c)(2)(ii), as described in Section 3.2.3.2 of this SE. In the LAR, the licensee stated that the MSL radiation monitors are not considered a Type A or Category 1 variable in accordance with the ANO-2 basis related to RG 1.97. The NRC staff's evaluation of this statement is described in Section 3.2.3.3 below.

3.2.3.2 Evaluation Regarding Categorization Under 10 CFR 50.36(c)(2)(ii)

Installed instrumentation that is used to detect and indicate a significant abnormal degradation of the RCPB in the control room meets Criterion 1 in 10 CFR 50.36(c)(2)(ii)(A). ANO-2 TS 3.4.6.1 requires the containment sump level, containment atmosphere particulate radioactivity, and containment atmosphere gaseous radioactivity monitors to detect RCS leakage into containment. The MSL radiation monitors are not included in TS 3.4.6.1 because they cannot detect RCS leakage into the containment.

The MSL radiation monitors detect radioactivity transported through either of the MSLs during events that involve SG tube leakage. ANO-2 TS 3.4.5 requires the SG tube integrity to be maintained and verified in accordance with ANO-2 TS 6.5.9, "Steam Generator (SG) Program." The MSL radiation monitors are not used in the SG program nor are they used to verify SG tube integrity. ANO-2 TS 3.4.6.2 requires RCS operational leakage to be limited to 150 gallons per day primary-to-secondary leakage through any one SG, and the leakage is verified to be less than or equal to this limit in accordance with the surveillance frequency control program (SFCP), 12 hours after establishment of steady-state operation. The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the SG program is an effective measure for minimizing the frequency of SG tube ruptures. For primary-to-secondary leakage determination, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. RCS level and pressure are the primary means for detecting a significant degradation of the RCPB and SG level, and feed rates are used to determine the affected SG. Although the MSL radiation monitors can detect radioactivity from SG tube leakage, they cannot determine the amount of leakage, only the amount of radioactivity released from the RCS. The NRC staff finds that the MSL radiation monitors provide defense-in-depth for primary-to-secondary leakage detection in the SG, but do not meet Criterion 1 in 10 CFR 50.36(c)(2)(ii)(A) to be included in the ANO-2 TSs because they cannot determine the amount or significance of the abnormal degradation of the RCPB.

Process variables, design features, or operating restrictions that are an initial condition of a design-basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier meet Criterion 2 in 10 CFR 50.36(c)(2)(ii)(B). The MSL radiation monitors are provided for monitoring effluent radioactive release under accident conditions. The monitors do not provide direct input to the reactor protective system or the engineered safety features actuation system functions, nor are the monitors a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The NRC staff finds that the MSL radiation monitors do not meet Criterion 2 in 10 CFR 50.36(c)(2)(ii)(B) to be in the ANO-2 TSs because they are not an initial condition of any DBA or transient analysis at ANO-2.

Structures, systems, or components (SSCs) that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier meet Criterion 3 in 10 CFR 50.36(c)(2)(ii)(C). The MSL radiation monitors are not part of a primary success path which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The instruments monitor potential radioactive release during an event that involves a SG tube leak but are not a primary success path used to determine the response of plant SSCs with respect to accident mitigation. With respect to SG tube leakage, the affected SG is determined by monitoring SG level and/or SG feed rates, both of which are included in ANO-2 TS 3.3.3.6, "Post-Accident Instrumentation," as Type A and/or Category 1 variables. The NRC staff finds that the MSL radiation monitors do not meet Criterion 3 in 10 CFR 50.36(c)(2)(ii)(C) for inclusion in the ANO-2 TSs because they are not part of any primary success path and do not function or actuate to mitigate a DBA or transient at ANO-2.

SSCs for which operating experience or probabilistic safety assessment (PSA) (also known as probabilistic risk assessment (PRA)) has shown to be significant to public health and safety meet Criterion 4 in 10 CFR 50.36(c)(2)(ii)(D). The MSL radiation monitors have not been shown to be risk significant to public health and safety by either operating experience or PRA. As discussed in the Final Policy Statement discussion of Criterion 4, it is the Commission policy that licensees retain the following systems in their TS LCOs, Action statements, and SRs, which operating experience and PSA have generally shown to be significant to public health and safety: reactor core isolation cooling/isolation condenser, residual heat removal, standby liquid control, and recirculation pump trip. The MSL radiation monitors are not listed in the discussion of Criterion 4 in the Final Policy Statement, nor has operating experience or the ANO-2 PRA shown them to be risk significant to public health and safety. Therefore, the NRC staff finds that the MSL radiation monitors do not meet Criterion 4 in 10 CFR 50.36(c)(2)(ii)(D).

At ANO-2, the TRM is considered part of the SAR, a licensee-controlled document; therefore, 10 CFR 50.59 will govern any changes to these relocated requirements. The TRM is Section 13.8 in the ANO-2 SAR. The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. Therefore, the NRC staff concludes that sufficient regulatory controls exist for the TRM.

The NRC staff finds that the MSL radiation monitors can be relocated from ANO-2 TSs because as discussed above, they do not meet the criteria of items that are required to be included in the TS LCOs as stated in 10 CFR 50.36(c)(2)(ii). The NRC staff concludes that the proposed change to relocate TS 3.3.3.1, Table 3.3-6 Instrument 2.c, its associated Action 19, SRs that are listed in Table 4.3-3, and the associated TS Bases to the ANO-2 TRM is acceptable.

3.2.3.3 Evaluation Regarding Categorization Within RG 1.97, Revision 3

The MSL radiation detectors were installed to monitor for a potential post-accident radioactive release via the main steam system. The MSL radiation monitors can also detect radioactivity from SG tube leaks, but they cannot determine the amount of leakage, only the amount of radioactivity released from the RCS. The MSL radiation monitors are designed with purpose and classification that are consistent with the capability of other non-TS instruments such as the main condenser offgas radiation monitor. The NRC staff reviewed the guidance of RG 1.97, Revision 3, and determined that the Effluent Radioactivity Noble Gas Effluent from Condenser Air Removal System Exhaust Monitor is a Type C, Category 3 variable in Table 3 of RG 1.97 and in Table 7.5-3 of the ANO-2 SAR. However, the MSL radiation monitors do not provide the primary information that is required for safety systems to accomplish their safety functions, and they are not type A or Category 1 variables of RG 1.97.

RG 1.97, Revision 3, identifies the "Vent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves" as a Type E variable to be monitored.

The LAR states, in part:

> NUREG-0737 specifically listed [pressurized water reactor (PWR)] steam safety valve discharge/atmospheric steam dump valve discharge as an effluent pathway which should be monitored. In response to this requirement, MSL [radiation monitor system (RMS)] was installed at ANO-2 and subsequently added to the ANO-2 TSs under Amendment 145 in March of 1993.
>
> … ANO-2 TS 3.3.3.1 requires a MSL radiation monitor to be operable on each MSL with alarm/trip setpoints within the specified limits.

The NRC staff understands that the licensee uses the MSL radiation monitors to satisfy both the NUREG-0737 and RG 1.97 Revision 3 guidance for monitoring the vent from SG Safety Relief Valves/Atmospheric Dump Valves. Also, RG 1.97 states that these monitors are considered Type E and Category 2 variables. Further, the licensee committed in the ANO-2 SAR Table 7.5-3 to implement these parameters as Type E and Category 2 variables.

These MSL radiation monitors do not provide the primary information needed to permit the control room operating personnel to take preplanned actions for which no automatic control is provided, nor are they required for safety systems to accomplish their safety functions for DBA events, and they are not classified as Type A or Category 1 variables in Table 3 of RG 1.97. These instruments monitor potential radioactive release during an event that involves an SG tube leak. The SG tube leakage is monitored by the "Steam Generator Level" instrument, which is a Type D, Category 1 variable.

Based on the description above, the NRC staff concludes that the MSL radiation monitors are not Type A or Category 1 because these monitors are not used to detect and indicate in the control room a significant abnormal degradation of the RCPB and therefore they do not meet the definition within RG 1.97.

In the LAR, the licensee concluded that the requirements for the ANO-2 MSL radiation monitoring system may be relocated to the ANO-2 TRM, along with the appropriate supporting Bases. The NRC staff reviewed the ANO-2 "Technical Requirements Manual Draft Markup"

(Attachment 4 of the LAR) and verified that the licensee has submitted its plan to relocate the MSL radiation monitor function (Function 19) to Section 3.3.2 of the TRM. Accordingly, the NRC staff determined that the licensee plans to preserve appropriate controls for maintaining the ability to monitor effluents from the Steam Generator Vent and Atmospheric Dump Valves.

Based on the above discussion, the NRC staff verified that the relocation of the MSL radiation monitors from Table 3.3-6 of the ANO-2 TSs to the TRM is acceptable because the function of the MSL radiation monitors does not fulfill any of the 10 CFR 50.36c(2)(ii) criteria. Additionally, this function does not need to be included in the TSs associated with post-accident monitoring instruments, which have been determined to be Type A or Category 1 instruments. This proposed change will also be consistent with the STSs in NUREG-1432.

3.2.3.4 Evaluation of the Proposed Deletion of Action 19 From the Table Notation of Table 3.3-6

In accordance with the relocation of Instrument 2.c. from TS Table 3.3-6 to the TRM as discussed in Section 3.2.3.1 of this SE, the licensee proposed to remove Action 19 from the Table Notation of TS Table 3.3-6.

The NRC staff verified that Action 19 only applies to Instrument 2.c of TS Table 3.3-6. This proposed change will not affect actions of the other instruments in the ANO-2 TSs. Further, this proposed change is consistent with the STSs in NUREG-1432. Overall, this proposed change continues to satisfy the criteria of 10 CFR 50.36(c)(2)(ii) and the requirements of GDCs 13 and 19, and is acceptable because Action 19 only applies to Instrument 2.c, which will no longer be located in TS Table 3.3-6.

3.2.3.5 Evaluation of the Proposed Deletion of Instrument 2.c from TS Table 4.3-3

In accordance with the relocation of the Instrument 2.c. from TS Table 3.3-6 to the TRM, as discussed in Section 3.2.3.1 of this SE, the licensee proposed to delete the Instrument 2.c. from the associated Surveillance TS Table 4.3-3.

Instrument	Channel Check	Channel Calibration	Channel Functional Test	Mode in Which Surveillance Required
2. Process Monitors c. Main Steam Line Radiation Monitors	SFCP	SFCP	SFCP	1,2,3, & 4

The NRC staff verified that since Instrument 2.c will be removed from TS Table 3.3-6, the associated SR for Instrument 2.c, will not be needed in TS Table 4.3-3, and therefore this proposed change is consistent with the relocation of Instrument 2.c. This proposed change will not affect other SRs for other instruments in the ANO-2 TSs and continues to satisfy the criteria of 10 CFR 50.36(c)(2)(ii) and the requirements of GDCs 13 and 19. In addition, this proposed change is consistent with the STS in NUREG-1432. Therefore, based on the above, this proposed deletion is acceptable.

3.3 Proposed Revision to TS 3.3.3.5

ANO-2 TS LCO 3.3.3.5 requires the remote shutdown instrumentation channels shown in TS Table 3.3-9 to be operable with readouts displayed external to the control room and SR 4.3.3.5 requires "[e]ach remote shutdown monitoring instrumentation channel to be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in [TS] Table 4.3-6."

The licensee proposed to revise ANO-2 TS LCO 3.3.3.5 to state that "The remote shutdown monitoring instrumentation channels shall be operable with readouts displayed external to the control room," and TS 3.3.3.5 Action to state that "With the number of OPERABLE remote shutdown monitoring channels less than required, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours." The revision would delete references to TS Table 3.3-9 from TS LCO 3.3.3.5 and its associated Action. Current TS Table 3.3-9 lists the "Readout Location," "Measurement Range," and "Minimum Channels Operable" for "Remote Shutdown Monitoring Instrumentation." The proposed change relocates these requirements to the TS Bases document for TS 3.3.3.5, a licensee-controlled document.

The licensee proposed to revise SR 4.3.3.5 to state that:

> Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program. The logarithmic neutron instrumentation, the startup channel instrumentation, and the reactor trip breaker indication are excluded from CHANNEL CALIBRATION.

The revision would delete reference to TS Table 4.3-6 from SR 4.3.3.5 for ANO-2. Current SR 4.3.3.5 requires "Each remote shutdown monitoring instrumentation channel to be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in TS Table 4.3-6," and current TS Table 4.3-6 lists the SFCP is the frequency for every instrument Channel Check and all the instrument Channel Calibrations with the exception of the Logarithmic Neutron Channel, Startup Channel, and Reactor Trip Breaker Indication. The Channel Calibration is not applicable to these channels in current TS Table 4.3-6. The proposed change relocates these requirements to the TS 3.3.3.5 Bases, a licensee-controlled document.

The proposed revision to ANO-2 SR 4.3.3.5 incorporates the current SR frequency requirements listed in TS Table 4.3-6 and relocates TS Table 4.3-6 to the TS 3.3.3.5 Bases. The NRC staff finds there are no changes to the current SR frequencies and with the current SR frequency requirements incorporated into SR 4.3.3.5, the TS Table 4.3-6 is no longer necessary and may be relocated. The NRC staff finds the proposed revision to SR 4.3.3.5 and relocation of TS Table 4.3-6 to be acceptable because there is no change to the SR frequencies.

The remote shutdown capability is described in ANO-2 SAR Sections 3.1.2, 7.4.1.5, and 15.1.26, and the remote shutdown panel is illustrated in Figure 7.4-3. The definition of operable in the ANO-2 TSs states that a system shall be operable or have operability when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling and seal water, lubrication, and other auxiliary equipment that

are required for the system to perform its function(s) are also capable of performing their related support function(s). This definition provides adequate guidance for determining what instrumentation and controls are necessary for a particular remote shutdown function.

Additionally, ANO-2 SR 4.3.3.5 still requires the remote shutdown instrumentation to be tested, which is sufficient to assure that the system is operable. Relocation of the table and its variables to the TS Bases is acceptable because the TS Bases are consistent with the design basis as described in the updated SAR. Therefore, the table and its variables can be relocated to the TS Bases without a significant safety impact. Changes to the TS Bases are evaluated in accordance with 10 CFR 50.59 requirements and TS 6.5.14.

Furthermore, precedent for the relocation has been established with TS Task Force (TSTF) Traveler TSTF-266, "Eliminate the Remote Shutdown System Table of Instrumentation and Controls," and Revision 2 of NUREG-1432, which incorporated TSTF-266 into the Improved STSs. Moreover, relocation of TS Table 3.3-9 and its variables to the TS Bases is acceptable because its inclusion in the TSs does not fall within the criteria for mandatory inclusion in the TSs in 10 CFR 50.36(c)(2)(ii). The NRC staff finds that sufficient regulatory controls exist under the regulations to maintain the effect of the provisions in the TS Bases.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and appropriate remedial measures are specified if the LCO is not met; the requirements of 10 CFR 50.36(c)(3) continue to be met because the SR ensures that the necessary quality of remote shutdown instrumentation is maintained, that facility operation will be within safety limits, and that the LCOs will be met. Therefore, the proposed change to relocate TS Table 3.3-9 to the TS 3.3.3.5 Bases, remove its reference from TS LCO 3.3.3.5 and its associated Action, incorporate the current SR frequency requirements listed in TS Table 4.3-6 into SR 4.3.3.5, and relocate TS Table 4.3-6 to TS 3.3.3.5 Bases are acceptable.

3.4 Proposed Revision to TS 3.4.6.2

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB meets Criterion 1 in 10 CFR 50.36(c)(2)(ii)(A). The ANO-2 reactor vessel is a right circular cylinder with two hemispherical heads. The lower head is welded to the vessel shell and the upper closure head can be removed to provide access to the reactor internals. Sealing is accomplished by using two silver-plated, nickel chromium iron alloy, self-energized O-rings. The space between the two rings is monitored to detect any inner-ring coolant leakage. An increase in the reactor head flange leakoff temperature indicates that reactor coolant is leaking past the inner O-ring to the reactor drain tank. The RCPB consists of both the inner and outer reactor head O-rings, and thus, the reactor head flange leakoff temperature indication does not monitor the complete RCPB. The NRC staff finds the reactor head flange leakoff temperature indication does not meet Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) to be in ANO-2 TSs because it does not detect a significant abnormal degradation of the RCPB.

Process variables, design features, or operating restrictions that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier meet Criterion 2 in 10 CFR 50.36(c)(2)(ii)(B). The reactor head flange leakoff temperature indication is not an initial condition to any DBA or transient analysis

described in the ANO-2 SAR. Therefore, the NRC staff finds that the reactor head flange leakoff temperature indication does not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) to be in ANO-2 TSs because it is not an initial condition of any DBA or transient analysis at ANO-2.

SSCs that are part of the primary success path and that function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier meet Criterion 3 in 10 CFR 50.36(c)(2)(ii)(C). The reactor head flange leakoff temperature indication does not function or actuate to perform any accident or transient mitigation function described in the ANO-2 SAR. Therefore, the NRC staff finds that the reactor head flange leakoff temperature indication does not meet Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) to be in ANO-2 TSs because it is not part of any primary success path and does not function or actuate to mitigate a DBA or transient at ANO-2.

SSCs for which operating experience or PSA has shown to be significant to public health and safety meet Criterion 4 in 10 CFR 50.36(c)(2)(ii)(D). The reactor head flange leakoff temperature indication monitors the inner reactor head flange O-ring and not the RCPB. The reactor head flange leakoff temperature indication has not been shown to be risk significant to public health and safety by either operating experience or PRA. As discussed in the Final Policy Statement discussion of Criterion 4, it is the Commission policy that licensees retain in their TSs LCOs, Action statements and SRs for the following systems, which operating experience and PSA have generally shown to be significant to public health and safety: reactor core isolation cooling/isolation condenser, residual heat removal, standby liquid control, and recirculation pump trip. The reactor head flange leakoff temperature indication is not listed in the discussion of Criterion 4 in the Final Policy Statement, nor has operating experience or the ANO-2 PRA shown it to be significant to public health and safety. Therefore, the NRC staff finds that the reactor head flange leakoff temperature indication does not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D).

Based on the above, the NRC staff finds that SR 4.4.6.2.1.b reactor head flange leakoff temperature indication can be deleted from ANO-2 TSs because it does not meet the criteria of items which are required to be included in the TS LCOs as stated in 10 CFR 50.36(c)(2)(ii).

As stated in 10 CFR 50.36(c)(3), “Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.” RCS leakage is continuously monitored by leak detection instruments as required by ANO-2 TS 3.4.6.1, “Leakage Detection Systems.” The leak detection instruments include a Containment Atmosphere Particulate Radioactivity Monitor, a Containment Atmosphere Gaseous Radioactivity Monitor, and the Containment Sump Level Monitor. Non-TS leak detection indications displayed in the control room include containment humidity, temperature, and pressure, along with quench tank and reactor drain tank level indications. In addition to the leak detection instruments in TS 3.4.6.1, TS 3.4.6.2 requires RCS leakage to be verified by performance of an RCS water inventory balance (SR 4.4.6.2.1.a). Reactor head flange leakage, which is collected in the reactor drain tank, is quantified as identified leakage, which is determined by performance of an RCS water inventory balance and limited to 10 gpm by TS 3.4.6.2. The initial RCS water inventory balance is required to be performed within 12 hours following RCS steady-state operation and once per 72 hours thereafter in accordance with the SFCP. The reactor head flange leakage is accounted for by RCS water inventory balance and can be detected by the various leakage monitoring systems.

The NRC staff finds that SR 4.4.6.2.1.b is not required to assure that facility operation will be within safety limits, and that TS 3.4.6.1 or TS 3.4.6.2 LCO will be met. TS 3.4.6.2 leakage detection function is maintained without the use of the reactor head flange leakoff temperature indication. Deletion of the SR 4.4.6.2.1.b has no impact on the ability of leakage detection instrumentation to detect RCS leakage and, therefore, does not meet the requirements of 10 CFR 50.36(c)(3). The NRC staff finds that the requirements of 10 CFR 50.36(c)(3) will continue to be met by SR 4.4.6.2.1.a, along with SR 4.4.6.2.2 and SR 4.4.6.2.3.

The NRC staff concludes that 10 CFR 50.36(c)(2)(ii) and 10 CFR 50.36(c)(3) will continue to be met and that reactor head flange leakoff temperature monitoring is not required to be retained in the ANO-2 TSs. Therefore, the proposed change to delete SR 4.4.6.2.1.b is acceptable.

With the deletion of ANO-2 SR 4.4.6.2.1.b, the licensee proposed to remove the letter “a” from SR 4.4.6.2.1.a and combine it with the first sentence, as this would be the only remaining requirement in SR 4.4.6.2.1. The NRC staff finds that the removal of the letter “a” and combining with the first sentence is an editorial change because it affects the format and style of the SR and does not, of itself, alter existing SRs. Therefore, the NRC staff finds that the proposed change is acceptable.

3.5 Proposed Revision to TS 3/4.7.2

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). SG pressure and temperature limits are not a form of instrumentation nor an SSC, and therefore, do not meet Criteria 1, 3, or 4 of 10 CFR 50.36(c)(2)(ii). The SG pressure and temperature limits are operating restrictions. However, these operating restrictions are not an initial condition for a DBA or transient analysis. Therefore, SG pressure and temperature limits do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. Since TS 3/4.7.2 does not satisfy these criteria, TS 3/4.7.2 and its associated TS Bases may be relocated to the ANO-2 TRM.

The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. The NRC staff concludes that sufficient regulatory controls exist for the TRM, TS 3/4.7.2 does not satisfy any of the criteria of 10 CFR 50.36(c)(2)(ii), and relocation to the TRM is acceptable.

3.6 Proposed Revision to TS 3/4.7.5

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Flood level/flood protection requirements are not a form of instrumentation nor an SSC, and therefore, do not meet Criteria 1, 3, or 4 of 10 CFR 50.36(c)(2)(ii). The flood level/flood protection requirements are operating restrictions. However, these operating restrictions are not an initial condition for a DBA or transient analysis. Therefore, flood level/flood protection do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. Since TS 3/4.7.5 does not satisfy these criteria, TS 3/4.7.5 and its associated TS Bases may be relocated to the ANO-2 TRM.

The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. The NRC staff concludes that sufficient regulatory controls exist for the TRM, TS 3/4.7.5 does not satisfy any of the criteria of 10 CFR 50.36(c)(2)(ii), and relocation to the TRM is acceptable.

3.7 Proposed Revision to TS 3/4.7.9

The NRC staff evaluated existing TS 3/4.7.9 to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Sealed source contamination requirements are not a form of instrumentation nor an SSC, and therefore, do not meet Criteria 1, 3, or 4 of 10 CFR 50.36(c)(2)(ii). The sealed source contamination requirements are not process variables, design features, or operating restrictions that are an initial condition for a DBA or transient analysis. Therefore, sealed source contamination requirements do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. The NRC staff finds that ANO-2 TS 3/4.7.9 does not meet any of the four criteria of 10 CFR 50.36(c)(2)(ii), which require mandatory inclusion in TSs. Therefore, TS 3/4.7.9 can be deleted from the ANO-2 TSs.

3.8 Proposed Revision to TS 3/4.7.12

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). The structural integrity requirements for the SFP are not monitored or controlled during plant operation; the integrity is verified through implementation of a visual inspection. The structural integrity of the SFP is not a form of instrumentation nor an SSC, and therefore, does not meet Criteria 1, 3, or 4 in 10 CFR 50.36(c)(2)(ii). The structural integrity requirements for the SFP are not a process variable, design feature, or operating restriction that is an initial condition for a DBA or transient analysis. Therefore, the SFP structural integrity requirements do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. Since TS 3/4.7.12 does not satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii), TS 3/4.7.12 and its associated TS Bases may be relocated to the ANO-2 TRM.

The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. The NRC staff concludes that sufficient regulatory controls exist for the TRM, TS 3/4.7.12 does not satisfy any of the criteria of 10 CFR 50.36(c)(2)(ii), and relocation to the TRM is acceptable.

3.9 Proposed Revision to TS 3.9.3.a

The licensee proposed to relocate ANO-2 TS 3.9.3.a and its associated TS Bases to the ANO-2 TRM. This TS prohibits the movement of irradiated fuel during refueling outages until 100 hours post-shutdown. As described in ANO-2 SAR Section 15.1.23, "Fuel Handling Accident," the fuel handling accident (FHA) analysis assumes that the reactor has been subcritical for at least 100 hours when the FHA occurs. The 100-hour limit is a plant-specific value that allows the radioactive decay of short-lived fission products, reducing the radioactive source term available for release in the event of an FHA. Since the implementation of ANO-2 Amendment No. 322 (ADAMS Accession No. ML20240A280), ANO-2 TS 3.9.3.a is no longer required because refueling operations before 100 hours post-shutdown are allowed and controlled per various ANO-2 TSs that apply during the movement of fuel (e.g., LCOs applicable to containment penetrations). Since Amendment No. 322, the 100-hour decay time limit is represented by the phrase "recently irradiated fuel," in the Applicability statements throughout ANO-2 TS LCOs, which are applicable during refueling operations. Additionally, as described in Amendment No. 322, the phrase, "recently irradiated fuel," is numerically defined in ANO-2 TS Bases Section 3/4 3.3.1 per TSTFs 51 and 471. This approach maintains public health and safety, as demonstrated by ANO-2's FHA analysis results satisfying NRC acceptance criteria that are based on 10 CFR 50.67, "Accident source term," while allowing the licensee operational flexibility to perform refueling operations without encountering overly burdensome licensing

basis restrictions. Based on the above, the NRC staff finds that relocating ANO-2 TS 3.9.3.a and its associated TS Bases to the ANO-2 TRM is acceptable.

The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. The NRC staff concludes that sufficient regulatory controls exist for the TRM and relocation to the TRM is acceptable.

3.10 Proposed Revision to TS 3.9.5

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). The design basis FHAs, as described in the updated SAR Section 15.1.23, do not take credit for the direct personnel communications required by this specification. Communications are not a form of instrumentation nor an SSC, and therefore, do not meet Criteria 1, 3, or 4 of 10 CFR 50.36(c)(2)(ii). Communications are not process variables, design features, or operating restrictions that are an initial condition for a DBA or transient analysis. Therefore, communications do not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs. The NRC staff finds that ANO-2 TS 3.9.5 does not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii), which require mandatory inclusion in TSs.

Therefore, since this TS does not fulfill any of the criteria in 10 CFR 50.36(c)(2)(ii) for items for which TSs must be established, the NRC staff finds that removing ANO-2 TS 3.9.5 along with the associated TS Bases is acceptable.

3.11 Proposed Revision to TS 3.9.6

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB meets Criterion 1 in 10 CFR 50.36(c)(2)(ii)(A). The operability requirements related to the refueling machine are applicable during movement of the CEAs or fuel assemblies within the RPV, which can only take place with the reactor head removed and the RCS depressurized. None of these operability requirements apply to instrumentation used to detect abnormal degradation of the RCPB. Therefore, the NRC staff finds that the refueling machine operability requirements do not meet Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) to be in ANO-2 TSs because the system/equipment does not detect degradation of the RCPB.

Process variables, design features, or operating restrictions that are an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier meet Criterion 2 in 10 CFR 50.36(c)(2)(ii)(B). Two DBAs that could involve the refueling machine have been identified in ANO-2 SAR Sections 15.1.15 and 15.1.23.

Section 15.1.15 of the ANO-2 SAR provides an analysis of an accident related to inadvertent loading of a fuel assembly into the improper position. ANO-2 TS 3.9.6 contains operability requirements for refueling machine capacity or excessive lifting force. These operability requirements are not related to the process used to ensure fuel assemblies are moved into the proper position in the core. TS 3.9.6 does not prevent the misloading of a fuel assembly or otherwise involve the initiating conditions for this accident. Therefore, the NRC staff concludes that TS 3.9.6 does not meet Criterion 2 for the inadvertent loading of a fuel assembly into the improper position DBA.

As noted in SAR Section 15.1.23, the applicable accident involving the refueling machine is the postulated FHA, which assumes a fuel assembly is dropped as an initial condition. The event consists of the drop of a fuel assembly either in the auxiliary building or inside containment. The refueling machine is not used to transport the fuel assembly into the SFP in the auxiliary building, so it does not play a role in the postulated FHA in this area of the plant. For the FHA inside the containment, the fuel assembly would be assumed to drop from the refueling machine. The effects and consequences of the design-basis FHA involving the refueling machine occurring inside the containment are also described in ANO-2 SAR Section 15.1.23. The radiological consequences of the FHA were determined in accordance with the guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792). The ANO-2 FHA assumed that two entire fuel assemblies, the dropped fuel assembly and the dropped-on fuel assembly, are damaged as a result of the drop. The failure of all fuel rods in both assemblies is a conservative assumption that provides a bounding condition for determining the potential radiological consequences.

ANO-2 TS 3.9.6 specifies that the refueling machine be operable with a minimum capacity for movement of fuel assemblies and an overload cutoff to protect the reactor vessel internals from excessive uplift force. Since the FHA assumes the refueling machine drops a fuel assembly, the capacity of the refueling machine is not an initial condition associated with the design-basis FHA. Similarly, operation of the overload cutoff is not an initial condition of the design-basis FHA because the assumption that all fuel rods are damaged in two assemblies bounds the potential damage to a fuel assembly that could result from excessive uplift forces. Therefore, the NRC staff concludes that ANO-2 TS 3.9.6 does not meet Criterion 2 for an FHA.

Section 9.1.4, "Fuel Handling System," of the ANO-2 SAR describes interlocks and operational constraints provided with the refueling machine design that interrupts hoisting of a fuel assembly if the load increases above the overload setpoint. As an additional protective feature, the hoisting load is visually displayed so that the operator can manually terminate the operation if an overload or other unsafe condition occurs.

The NRC staff determined that ANO-2 TS 3.9.6 is unrelated to inadvertent fuel loading accidents and does not represent an initial condition for fuel drop accident scenarios, and the consequences of such scenarios are bounded by those already considered in the ANO-2 SAR. Therefore, the NRC staff finds that ANO-2 TS 3.9.6 does not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) to be in ANO-2 TSs because it is not an initial condition of any DBA or transient analysis at ANO-2.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) applies to SSCs that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The operability requirements related to the refueling machine are not SSCs that are part of the primary success path and which function to actuate or mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Similarly, the refueling machine is used solely during refueling operations with the reactor head removed and does not actuate to mitigate a DBA or transient. Therefore, the NRC staff finds that ANO-2 TS 3.9.6 does not meet Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) to be in ANO-2.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) applies to SSCs for which operating experience or PSA has shown to be significant to public health and safety. The refueling machine has not been shown to be risk significant to public health and safety by either operating experience or PRA.

As discussed in the Final Policy Statement discussion of Criterion 4, it is the Commission policy that licensees retain in their TSs LCOs, Action statements and SRs for the following systems, which operating experience and PSA have generally shown to be significant to public health and safety: reactor core isolation cooling/isolation condenser, residual heat removal, standby liquid control, and recirculation pump trip. The refueling machine is not listed in the discussion of Criterion 4 in the Final Policy Statement, nor has operating experience or the ANO-2 PRA shown them to be risk significant to public health and safety; therefore, the NRC staff finds that the ANO-2 TS 3.9.6 does not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D).

The NRC staff concludes that TS 3.9.6 may be relocated from ANO-2 TSs because it does not meet the criteria of items that are required to be included in the TS LCOs as stated in 10 CFR 50.36(c)(2)(ii). The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. The NRC staff concludes that sufficient regulatory controls exist for the TRM and relocation to the TRM is acceptable.

3.12 Proposed Revision to TS 3.9.7

The NRC staff evaluated the existing TS to determine whether it meets any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii) and described in Section 2.3.1 of this SE, as follows:

The restrictions on the weight and travel of crane loads are applicable over irradiated spent fuel within the refueling building. The instrumentation included in this SR applies to crane travel over the SFP. Failed or inoperable instrumentation does not impact or result in any abnormal degradation of the reactor coolant boundary. The NRC staff finds that the crane travel and lift restriction does not meet Criterion 1 of 10 CFR 50.36(c)(2)(ii) to be in ANO-2 TS because it does not detect degradation of the RCPB.

The TS load restrictions are provided to disallow travel over the SFP with loads in excess of the weight of a typical spent fuel assembly and lifting equipment. Because the crane is designed in accordance with NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants,” dated May 1979 (ADAMS Accession No. ML110450636), a postulated heavy load drop (i.e., dry storage cask) is not a credible event, the structural integrity of the SFP cask will not be impaired, and no safety related components will be impacted. ANO-2 SAR Section 15.1.23 analyzes FHAs involving lifts of spent fuel over or within the SFP. Therefore, the NRC staff finds that the crane travel and lift restriction does not meet Criterion 2 of 10 CFR 50.36(c)(2)(ii) and is not required to be in ANO-2 TSs because it is not an initial condition of any DBA or transient analysis.

Crane travel and load limit is not an initial condition of the FHA analysis and is not required to mitigate an FHA or any other DBA or transient relating to fission product barrier integrity. With ANO-2 using a single failure proof crane, a postulated heavy load drop (i.e., cask) is not a credible event. Therefore, the NRC staff finds that the crane travel and lift restriction does not meet Criterion 3 of 10 CFR 50.36(c)(2)(ii) and is not required to be in ANO-2 TS because it is not part of any primary success path and does not function or actuate to mitigate a DBA or transient.

The restrictions on the weight and travel of crane loads over irradiated spent fuel is confined within the fuel building. Criterion 4 of 10 CFR 50.36(c)(2)(ii) requires a TS be established for, “[a] structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.” The LCO proposed for

deletion is an operating restriction and not an SSC, and therefore, does not meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The NRC staff concludes that TS 3.9.7 can be relocated from ANO-2 TS because it does not meet the criteria of items which are required to be included in the TS LCO as stated in 10 CFR 50.36(c)(2)(ii). The NRC staff considers the TRM to have an appropriate level of control under 10 CFR 50.59 for the requirements proposed for relocation. In addition, the proposed change will also allow the ANO-2 TSs to be consistent with NUREG-1432 and the NRC Final Policy Statement. Based on the foregoing, the NRC staff concludes that relocation to the TRM is acceptable.

3.13 Proposed Revisions to TS 3.11.1, 3.11.2, 3.11.3 and the Addition of TS 6.5.8

3.13.1 Proposed Revision to TS 3.11.1

Existing ANO-2 TS 3.11.1 requires that the quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. The existing TS defines applicability of this TS to those outdoor temporary tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents, and do not have overflows and surrounding area drains connected to the liquid radwaste treatment system. The associated SR specifies that the quantity of radioactive material contained in each unprotected outside temporary radioactive liquid storage tank shall be determined to be within the specified limits and sampled at least once per 7 days when radioactive materials are being added to the tank.

The proposed Administrative Control TS 6.5.8 requirement does not contain a specific curie limit for the affected tanks, but specifies that the liquid radwaste quantities shall be determined in accordance with the ODCM. As indicated in the licensee's application, ODCM 2.3.1, "Radioactive Liquid Effluents," Required Action A.1, requires action to be initiated immediately to restore tank contents to within limits. The ODCM and related implementing procedures along with automated analysis reports provide for sampling and ensuring tank contents are verified and limits maintained. In accordance with the proposed TS 6.5.8, the potential dose impact to a member of the public is limited to that described in 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest public potable water supply and the nearest surface water supply. As noted by the licensee, ANO does not currently have any outdoor unprotected liquid radwaste storage tanks on site.

The licensee's proposed relocation of ANO-2 TS 3.11.1 to a licensee-controlled ODCM does not change the licensee's requirement to limit the quantity of radioactive material, such that radioactive material released during a tank failure does not exceed the annual average concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest public potable water supply and the nearest surface water supply in an unrestricted area.

As discussed above, the licensee currently does not have any applicable tanks, sampling is performed by procedure, and the proposed new TS 6.5.8 program is consistent with the existing ANO-1 TS 5.5.12 program. The proposed TS 6.5.8 also includes a surveillance program to ensure that the quantity of radioactivity stored in the unprotected temporary outdoor liquid holdup tanks are limited to the above criteria. The NRC staff determined that TS 3.11.1 need not be retained as an LCO, and that the deletion of the LCO and transfer of the information to licensee-controlled documents are acceptable.

3.13.2 Proposed Deletion of TS 3.11.2

The requirements of ANO-2 TS 3.11.2 include an LCO for a specific curie limit for the quantity within gas storage tanks as well as actions to suspend additions and reduce the tank contents to within this limit when the tank limit is exceeded. This LCO also contains an action requiring the licensee to describe events leading to exceedance of limits in its Radioactive Effluent Release Report defined in TS 6.6.3. The associated SR specifies that the quantity of radioactive material contained in each gas storage tank shall be determined to be within the TS curie limit.

The limits and requirements of TS 3.11.2 are intended to ensure that the uncontrolled release of the radioactivity contained in a waste gas storage tank would not result in a whole-body dose equivalent in excess of 0.5 rem at the nearest exclusion area boundary.

The changes proposed in this LAR include the relocation of the TS requirements and the addition of program requirements in the Administrative Controls section of the TSs to address the requirements of existing TS 3.11.2. The proposed TS 6.5.8 program will contain similar criteria to ensure that the quantity of radioactivity contained in each waste gas decay tank is less than the amount that would result in a whole-body exposure of ≥ 0.5 rem to any individual in an unrestricted area.

The TS 6.5.8 program commits to the gaseous radioactivity quantities being determined following the methodology in BTP ETSB 11-5. The NRC staff notes that licensee's LAR dated December 16, 2019, provided an incorrect reference to ADAMS Accession No. ML052350110. The licensee subsequently corrected the ADAMS Accession No. to ML052350105 in its letter dated September 22, 2020, in response to an NRC RAI. The use of BTP ETSB 11-5 is consistent with SAR Section 15.1.16, "Waste Gas Decay Tank Leakage or Rupture," analysis and NUREG-1432, Revision 4, Section 5.5.12.

Based on the discussion above, the numerical limit on the quantity of radioactive material contained in gas storage tanks is not required to be included in the TSs in accordance with the criteria in 10 CFR 50.36(c)(2). This numerical limit would be moved to the ODCM and controlled under 10 CFR 50.59. Additionally, the proposed changes to the TSs are consistent with the previously approved NRC guidance in NUREG-1432, Revision 4. In addition, TS 6.5.8 would also establish a surveillance program to ensure the quantity of radioactivity is limited accordingly. Based on the above, the NRC staff finds that ANO-2 TS 3.11.2 does not satisfy the screening criteria of 10 CFR 50.36(c)(2)(ii), and consistent with the guidance of the STS, the relocation of TS 3.11.2 to the ODCM is acceptable.

3.13.3 Proposed Deletion of TS 3.11.3 and Table 3.11-3

The NRC staff addressed the relocation of the explosive gas monitoring TS to licensee-controlled documents in NRC GL 95-10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation," dated December 15, 1995 (ADAMS Accession No. ML031070178). In GL 95-10, the NRC staff noted that explosive gas monitoring instrumentation requirements address detection of possible precursors to the failure of a waste gas system but do not prevent or mitigate DBAs or transients that assume a failure of or present a challenge to a fission product barrier.

Existing TS 3.11.3 requires that the concentration of the hydrogen/oxygen shall be limited in the waste gas storage tanks to Region "A" of TS Figure 3.11-1. The associated SR specifies that

the concentration of hydrogen/oxygen in the waste gas holdup system shall be within these limits, with the waste gas system in operation, by continuously monitoring with the hydrogen/oxygen monitors required operable per TS Table 3.11-3. This table contains criteria for explosive gas monitoring instrumentation.

The NRC staff compared the existing TS 3.11.3 against the criteria provided in 10 CFR 50.36(c)(2)(ii) to determine whether its inclusion as a TS LCO is required. The NRC staff determined that the condition limiting the oxygen concentration in the gas decay tank did not meet any of the criteria for the following reasons:

- The explosive gas monitoring instrumentation is not installed instrumentation used to detect and indicate in the control room, a significant abnormal degradation of the RCPB.

- The concentration or quantity of gases in the gas decay tank system is not a process variable that is an initial condition of a DBA or transient analysis, and failure of the gas decay tank is not a constituent of any such accident, because the hydrogen and oxygen concentrations are controlled to prevent conditions that could cause failure of the gas decay tank system.

- The explosive gas monitoring instrumentation is not part of a primary success path that mitigates a design basis accident or transient because the instrumentation functions to maintain normal conditions that avoid potential precursors to a gas decay tank system failure.

- The gas decay tank system monitoring instrumentation is not significant to public health and safety.

As noted in the licensee's LAR dated December 16, 2019, both mixture control and Figure 3.11-1 are established in plant operation procedures. Furthermore, the transfer of TS 3.11.3, Table 3.11-3 and Figure 3.11-1 to licensee-controlled documents is consistent with NRC guidance provided in GL 95-10. Based on the above, the NRC staff determined that TS 3.11.3 need not be retained as an LCO, and that deletion of the LCO and transfer of the information to licensee-controlled documents are acceptable.

3.13.4 Proposed Addition of TS 6.5.8

Proposed TS 6.5.8 for ANO-2 would provide general programmatic controls in lieu of the current TSs proposed for relocation. TS 6.5.8, which is essentially a plant-specific version of TS 5.5.12 from NUREG-1432, is similar to the established ANO-1 program and contains the following safety provisions:

- Quantities of gaseous radioactivity shall be determined in accordance with BTP ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure";

- Quantities of liquid radioactivity shall be determined in accordance with ODCM;

- A surveillance program to ensure that the quantity of radioactivity contained in each Waste Gas Decay Tank is less than the amount that would result in a whole body

exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of each tank's contents; and

- A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The programmatic requirements for the Explosive Gas and Storage Tank Radioactivity Monitoring Program included in proposed TS 6.5.8 are consistent with the model TS 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," included in NUREG-1432. ANO currently has a program implemented for ANO-1 that is consistent with NUREG-1432 and the proposed ANO-2 program. Therefore, proposed new TS 6.5.8 satisfies the guidance of Section 16.0, "Technical Specifications," of NUREG-0800 and is acceptable.

3.14 Proposed Editorial Changes

The licensee proposed the following editorial changes:

- Renumber the annotation to the next page on TS page 3/4 9-4,
- Annotate the next page number on TS pages 3/4 1-3, 3/4 3-27, 3/4 3-36, 3/4 7-10, 3/4 9-2,
- Delete the annotation to the next TS page on TS page 3/4 7-18, and
- Delete TS pages 3/4 3-26a, 3/4 7-11, 3/4 7-12, and 3/4 7-13.

The NRC staff evaluated the proposed changes to the TS page numbering and annotations to the next page number and has determined that these changes are editorial because there are no changes to the TS requirements. The NRC staff also evaluated the TS page deletions and determined that the TS pages are either void of any TS requirements or the TS requirements have been moved to the previous TS page and therefore, the proposed changes are editorial because there are no changes to the TS requirements. In addition, the annotation on TS page 3/4 3-27 identified above was previously incorporated into the ANO-2 TSs in Amendment No. 322, which was approved by the NRC staff by letter dated October 30, 2020 (ADAMS Accession No. ML20240A280).

The NRC staff concludes that these proposed changes are editorial and do not change any TS requirements and therefore, are acceptable.

3.15 TS Bases Changes

In accordance with 10 CFR 50.36(a)(1), the licensee submitted TS Bases changes that correspond to the proposed TS changes for information only. The licensee will make supporting changes to the TS Bases in accordance with TS 6.5.14.

3.16 Summary of the NRC Staff Evaluation

The NRC staff has reviewed the licensee's application with the supporting documentation. Based on its review, the NRC staff concludes that the proposed relocations from TSs are acceptable because TS LCOs do not meet the criteria of items that are required to be included in the TS LCOs as stated in 10 CFR 50.36(c)(2)(ii), that sufficient regulatory controls exist for the TRM and TS Bases, and that relocation to the TRM or TS Bases is acceptable. The NRC staff also concludes that the proposed TS deletions are acceptable because the TS LCOs do not meet the criteria of items which are required to be included in the TS LCOs as stated in 10 CFR 50.36(c)(2)(ii). The NRC staff further concludes that the proposed changes as discussed above in Section 3.0 are acceptable and there is reasonable assurance that 10 CFR 50.36(c)(2) and 10 CFR 50.36(c)(3) will continue to be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment on December 21, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components 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 February 25, 2020 (85 FR 10733), and there has been no public comment on such finding. 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.

6.0 CONCLUSION

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

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G. Curran
S. Garry
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T. Wengert

Date: February 8, 2021

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 - ISSUANCE OF AMENDMENT NO. 323 RE: TECHNICAL SPECIFICATION DELETIONS, ADDITIONS, AND RELOCATIONS (EPID L-2019-LLA-0284) DATED FEBRUARY 8, 2021

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ADAMS Accession No. ML20351A153 *By memorandum

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DSS/STSB/BC*	NRR/DEX/EICB/BC*
NAME	TWengert	PBlechman w/comments	VCusumano	MWaters
DATE	12/22/2020	12/16/2020	3/20/2020	7/10/2020
OFFICE	NRR/DSS/SNSB/BC*	NRR/DSS/SCPB/BC*	NRR/DRA/ARCB/BC(A)*	NRR/DRO/IOLB/BC*
NAME	SKrepel	BWittick	KHsueh (DGarmon-Candelaria for)	CCowdrey
DATE	8/12/2020	8/25/2020	10/2/2020	11/17/2020
OFFICE	NRR/DEX/ESEB/BC	OGC - NLO	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM
NAME	JColaccino	TJones	JDixon-Herrity	TWengert
DATE	12/23/2020	2/8/2021	2/8/2021	2/8/2021

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