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

June 25, 2024

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. 333 RE: REVISION TO TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b" (EPID L-2023-LLA-0052)

Dear Site Vice President:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 333 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 5, 2023, as supplemented by letters dated January 11, 2024, and April 24, 2024.

The amendment revises the TSs to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met.

The changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b," dated July 2, 2018. The NRC staff issued a final model safety evaluation approving TSTF-505, Revision 2 on November 21, 2018.

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/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 333 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. 333
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 April 5, 2023, as supplemented by letters dated January 11, 2024, and April 24, 2024, 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) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 333, 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 180 days from the date of issuance. Implementation of the amendment shall also include the update of the Fire PRA model using the FLEX equipment failure rates in PWROG-18042-NP prior to implementing the RICT Program for ANO-2, as described in the supplemental letter dated April 24, 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennivine K. Rankin

Digitally signed by Jennivine K. Rankin
Date: 2024.06.25 09:48:54 -04'00'

Jennivine K. Rankin, 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: June 25, 2024

ATTACHMENT TO LICENSE AMENDMENT NO. 333

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.

Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
3/4 3-4	3/4 3-4
3/4 3-5b	3/4 3-5b
3/4 3-14	3/4 3-14
3/4 3-15a	3/4 3-15a
3/4 5-3	3/4 5-3
3/4 6-4	3/4 6-4
3/4 6-10	3/4 6-10
3/4 6-14	3/4 6-14
3/4 6-16	3/4 6-16
3/4 7-5	3/4 7-5
3/4 7-10	3/4 7-10
3/4 7-15	3/4 7-15
3/4 8-1	3/4 8-1
3/4 8-1a	3/4 8-1a
3/4 8-2	3/4 8-2
3/4 8-2a	3/4 8-2a
3/4 8-6	3/4 8-6
3/4 8-8	3/4 8-8
6-18b	6-18b
---	6-18c

(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) Maximum Power Level

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) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 333, 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) Additional Conditions

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.

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be manually bypassed above 10^{-4}% power; bypass shall be automatically removed before decreasing below 10^{-4}% power.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed before pressurizer pressure exceeds 500 psia.

(c) Trip may be manually bypassed below 10^{-2}% power; bypass shall be automatically removed before exceeding 10^{-2}% power. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 1% power; bypass shall be automatically removed before exceeding 1% power.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

ACTION STATEMENTS

ACTION 1 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 4 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 5 – With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, place the reactor trip breakers of the inoperable channel in the tripped condition within 1 hour or be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

ACTION 6 –

a. With one CEAC inoperable, operation may continue for up to 7 days or in accordance with the Risk Informed Completion Time Program provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group. After 7 days or after expiration of the Risk Informed Completion Time, whichever is longer, operation may continue provided that ACTION 6.b is met.

b. With both CEACs inoperable, operation may continue provided that:

   1. Within 1 hour the margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

   2. Within 4 hours:

      a) All CEA groups are withdrawn within the limits of Specifications 3.1.3.5 and 3.1.3.6.b, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to both CEACs inoperable.

      c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "OFF" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is below 400 psia; bypass shall be automatically removed before pressurizer pressure exceeds 500 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Remote manual not provided for RAS. These are local manuals at each ESF auxiliary relay cabinet.

ACTION STATEMENTS

ACTION 9 – With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 10 – With the number of channels OPERABLE one less than the Total Number of Channels, operation in the applicable MODES may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed for greater than 48 hours, the desirability of maintaining this channel in the bypassed condition shall be reviewed as soon as possible but no later than the next regularly scheduled OSRC meeting in accordance with the Quality Assurance Program Manual (QAPM). The channel shall be returned to OPERABLE status prior to startup following the next COLD SHUTDOWN.

If an inoperable Steam Generator ΔP or RWT Level – Low channel is placed in the tripped condition, remove the inoperable channel from the tripped condition within 48 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	Functional Unit Bypassed
1. Containment Pressure – NR	Containment Pressure – High (RPS) Containment Pressure – High (ESFAS) Containment Pressure – High-High (ESFAS)
2. Steam Generator 1 Pressure	Steam Generator 1 Pressure – Low Steam Generator 1 ΔP (ESFAS 1) Steam Generator 2 ΔP (ESFAS 2)
3. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (ESFAS 1) Steam Generator 2 ΔP (ESFAS 2)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 12 – With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 13 – With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ACTION 14 – With the number of OPERABLE 460 volt Degraded Voltage (Functional Unit 7.b) channels one less than the Total Number of Channels or with both 4.16 kv Loss of Voltage (Functional Unit 7.a) channels inoperable on a single bus:

a. Immediately declare the affected diesel generator inoperable, and

b. Restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS – $T_{avg} \geq 300$ °F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each sub-system comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) train,

b. One OPERABLE low-pressure safety injection (LPSI) train, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3 with pressurizer pressure ≥ 1700 psia.

ACTION:

a. With one ECCS subsystem inoperable due to an inoperable LPSI train, restore the inoperable train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program; otherwise, be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

b. With one or more ECCS subsystems inoperable due to conditions other than "a" above and 100% of ECCS flow equivalent to a single OPERABLE HPSI and LPSI train is available, restore the inoperable train(s) to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

c. With less than 100% ECCS flow equivalent to either the HPSI or LPSI trains within both ECCS subsystems, restore at least one HPSI train and one LPSI train to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to < 1700 psia within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable in one or more containment air locks[1,2]:

   1. Verify that at least the OPERABLE air lock door is closed in the affected air lock within one hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed[3].

   2. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

b. With the containment air lock interlock inoperable in one or more containment air locks[1]:

   1. Verify that at least one OPERABLE air lock door is closed in the affected air lock within one hour and restore the inoperable air lock interlock to OPERABLE status within 24 hours or lock an OPERABLE air lock door closed[4].

   2. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

c. With one or more air locks inoperable for reasons other than those addressed in ACTION a. or b.:

   1. Immediately initiate action to evaluate overall containment leakage per LCO 3.6.1.2.

   2. Verify that at least one door in the affected air lock is closed within one hour and restore the affected air lock to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

Otherwise, be in at least HOT STANDBY within the next six hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

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[1] Separate ACTION entry is allowed for each air lock.

[2] With both air locks inoperable, entry and exit is permissible for seven days under administrative controls.

[3] Entry and exit is permissible to perform repairs on the affected air lock components.

[4] Entry and exit is permissible under the control of a dedicated individual.

3/4.6.2 DEPRESSURIZATION, COOLING, AND pH CONTROL SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal (CSAS) and automatically transferring suction to the containment sump on a Recirculation Actuation Signal (RAS). Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With both containment spray systems inoperable (Note 1):

1. Within 1 hour verify both CREVS trains are OPERABLE, and

2. Restore at least one containment spray system to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

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

1. Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.

2. Verifying that the system piping is full of water from the RWT to at least elevation 505' (equivalent to > 12.5% indicated narrow range level) in the risers within the containment.

b. Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head when tested pursuant to the INSERVICE TESTING PROGRAM.

Note 1: ACTION b is not applicable when the second containment spray system is intentionally made inoperable.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent containment cooling groups shall be OPERABLE with two operational cooling units in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION[1]:

a. With one group of the above required containment cooling units inoperable and both containment spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program.

b. With two groups of the above required containment cooling units inoperable and both containment spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program. Restore both above required groups of cooling units to OPERABLE status within 7 days, or in accordance with the Risk Informed Completion Time Program, of initial loss.

c. With one group of the above required containment cooling units inoperable and one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days, or in accordance with the Risk Informed Completion Time Program, of initial loss.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Note 1: The containment spray systems may be considered OPERABLE with respect to ACTIONs a, b, and c above if solely inoperable due to containment accident generated and transported debris exceeding the analyzed limits and LCO 3.6.4.1, ACTION a, is being met.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Note: Enter applicable ACTION(s) for system(s) made inoperable by containment isolation valves.

With one or more isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours or in accordance with the Risk Informed Completion Time Program either:

a. Restore the inoperable valve(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration by use of at least one closed manual valve or blind flange; or

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

PLANT SYSTEMS

EMERGENCY FEEDWATER (EFW) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two EFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS:[1]

NOTE 1: Specification 3.0.4.b is not applicable.

NOTE 2: Only applicable if MODE 2 has not been entered following refueling.

NOTE 3: Not applicable when the turbine-driven EFW train is inoperable solely due to one inoperable steam supply.

NOTE 4: LCO 3.0.3 and all other LCO ACTIONS requiring MODE changes are suspended until one EFW train is restored to OPERABLE status.

a. With the turbine-driven EFW train inoperable in MODE 3 following refueling[2], OR with the turbine-driven EFW train inoperable due to one inoperable steam supply, restore the turbine-driven EFW train to OPERABLE status within 7 days or in accordance with the Risk Informed Completion Time Program.

b. With one EFW train inoperable for reasons other than ACTION a, restore the inoperable train to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program.

c. With the turbine-driven EFW train inoperable due to one inoperable steam supply AND the motor-driven EFW train inoperable, restore either the steam supply to the turbine-driven train OR the motor-driven EFW train to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program.

d. With ACTION a, b, or c not met, be in HOT SHUTDOWN within the next 12 hours.

e. With both EFW trains inoperable, immediately initiate action to restore one EFW train to an OPERABLE status.[3,4]

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 or in accordance with the Risk Informed Completion Time Program; 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.

Next page is 3/4 7-15

PLANT SYSTEMS

3/4.7.3 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Notes:

1. Enter applicable ACTION(s) of LCO 3.8.1.1, "AC Sources – Operating," for diesel generator made inoperable by service water system.

2. Enter applicable ACTION(s) of LCO 3.4.1.3, "Reactor Coolant System – Shutdown," if a required shutdown cooling loop is made inoperable by service water system.

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two service water loops shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. In accordance with the Surveillance Frequency Control Program during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on CCAS, MSIS and RAS test signals.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system and

b. Two separate and independent diesel generators each with:

1. A day fuel tank containing a minimum volume of 300 gallons of fuel,

2. A separate fuel storage system, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE: Specification 3.0.4.b is not applicable to diesel generators.

a. With one offsite A.C. circuit of the above required A.C. electrical power sources inoperable, perform the following:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and

2. Within 24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required features(s), declare required features(s) with no offsite power available inoperable when its redundant required features(s) is inoperable, and

3. Restore the offsite A.C. circuit to OPERABLE status within 72 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN. Startup Transformer No. 2 may be removed from service for up to 30 days as part of a preplanned preventative maintenance schedule. The 30-day allowance may be applied not more than once in a 10-year period.

LIMITING CONDITION FOR OPERATION

b. With one diesel generator of the above required A.C. electrical power source inoperable, perform the following:

   1. Demonstrate the OPERABILITY of both the offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and

   2. Within 4 hours from discovery of one required diesel generator inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) supported by the inoperable diesel generator inoperable when its redundant required feature(s) is inoperable, and

   3. Demonstrate the OPERABILITY of the remaining OPERABLE diesel generator within 24 hours by:

      i. Determining the OPERABLE diesel generator is not inoperable due to a common cause failure, or

      ii. Perform Surveillance Requirement 4.8.1.1.2.a.4 unless:

         a. The remaining diesel generator is currently in operation, or

         b. The remaining diesel generator has been demonstrated OPERABLE within the previous 24 hours, and

   4. Restore the diesel generator to OPERABLE status within 14 days (See Note 1) or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Note 1 – If the Alternate A.C. Diesel Generator (AACDG) is determined to be inoperable during this period, then a 72 hour restoration or Risk Informed Completion Time period is applicable until either the AACDG or the diesel generator is returned to operable status (not to exceed 14 days or the Risk Informed Completion Time from the initial diesel generator inoperability).

LIMITING CONDITION FOR OPERATION

c. With one offsite A.C. circuit and one diesel generator of the above required A.C. electrical power sources inoperable (see Note 2), perform the following:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. circuit by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and,

2. Within 4 hours from discovery of one required diesel generator inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) supported by the inoperable diesel generator inoperable if its redundant required feature(s) is inoperable, and

3. If the diesel generator became inoperable due to any cause other than preplanned preventative maintenance or testing, then

   i. Demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.8.1.1.2.a.4 within 8 hours, except when:

      a. The remaining diesel generator is currently in operation, or

      b. The remaining diesel generator has been demonstrated OPERABLE within the previous 8 hours, and

4. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or in accordance with the Risk Informed Completion Time Program, and

5. Restore the remaining inoperable A.C. Source to an OPERABLE status (Offsite A.C. Circuit within 72 hours or in accordance with the Risk Informed Completion Time Program, or Diesel Generator within 14 days or in accordance with the Risk Informed Completion Time Program (see b.4, Note 1)), based on the time of the initiating event that caused the inoperability.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

Note 2 – Enter applicable ACTIONs of LCO 3.8.2.1, "A.C. Distribution – Operating," when ACTION c is entered with no AC power to any train.

LIMITING CONDITION FOR OPERATION

d. With two offsite A.C. circuits of the above required A.C. electrical power sources inoperable, perform the following:

 1. Perform Surveillance Requirement 4.8.1.1.2.a.4 on the diesel generators within the next 8 hours except when:

  i. The diesel generators are currently in operation, or

  ii. The diesel generators have been demonstrated OPERABLE within the previous 8 hours, and

 2. Within 12 hours from discovery of two required offsite A.C. circuits inoperable concurrent with inoperability of redundant required feature(s), declare required feature(s) inoperable when its redundant required feature(s) is inoperable, and

 3. Restore one of the inoperable offsite A.C. circuits to OPERABLE status within 24 hours or in accordance with the Risk Informed Completion Time Program, and

 4. Restore both A.C. circuits within 72 hours or in accordance with the Risk Informed Completion Time Program of the initiating event,

 Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

e. With two diesel generators of the above required A.C. electrical power sources inoperable, perform the following:

 1. Demonstrate the OPERABILITY of the two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter, and

 2. Restore one of the inoperable diesel generators to OPERABLE status within 2 hours, and

 3. Restore the remaining inoperable diesel generator within 14 days or in accordance with the Risk Informed Completion Time Program (see b.4, Note 1) of the initiating event.

 Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

4160 volt Emergency Bus # 2A3

4160 volt Emergency Bus # 2A4

480 volt Emergency Bus # 2B5

480 volt Emergency Bus # 2B6

120 volt A.C. Vital Bus # 2RS1

120 volt A.C. Vital Bus # 2RS2

120 volt A.C. Vital Bus # 2RS3

120 volt A.C. Vital Bus # 2RS4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Note: Enter applicable ACTIONs of LCO 3.8.2.3, "DC Sources – Operating" for DC train(s) made inoperable by inoperable power distribution subsystems.

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or in accordance with the Risk Informed Completion Time Program; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

DC SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required full capacity chargers inoperable:

   i. Restore the battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, and

   ii. Verify battery float current ≤ 2 amps once per 12 hours.

b. With one DC electrical power subsystem inoperable for reasons other than ACTION 'a' above, restore the inoperable DC electrical power subsystem to OPERABLE status within 2 hours or in accordance with the Risk Informed Completion Time Program.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. LCO 3.0.4.a is not applicable when entering HOT SHUTDOWN.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 In accordance with the Surveillance Frequency Control Program by verifying that the battery terminal voltage is greater than or equal to the minimum established float voltage.

6.5.19 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system ACTIONs. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

c. Provisions to ensure that an inoperable supported system's allowed outage time is not inappropriately extended as a result of multiple support system inoperabilities, and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate ACTIONs of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate ACTIONs to enter are those of the support system.

6.5.20 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1 and 2;

6.5.20 Risk Informed Completion Time Program (continued)

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods approved for use with this program, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 333 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 April 5, 2023 (Reference 1), as supplemented by letters dated January 11, 2024 (Reference 2), and April 24, 2024 (Reference 3), Entergy Operations, Inc. (Entergy, the licensee) submitted a license amendment request (LAR) for Arkansas Nuclear One, Unit 2 (ANO-2).

The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b,” dated July 2, 2018 (Reference 4). The U.S. Nuclear Regulatory Commission (NRC, the Commission) issued a final model safety evaluation (SE) to be used when preparing a plant-specific SE of an LAR to adopt TSTF-505, Revision 2, on November 21, 2018 (Reference 5).

The licensee has proposed variations from the TS changes described in TSTF-505, Revision 2, which are described in attachment 1, “Evaluation of the Proposed Change,” and attachment 5, “ANO-2 Technical Specification TSTF-505 Cross-Reference,” of the LAR, and evaluated in section 3.2.1 of this SE.

The NRC staff participated in a regulatory audit in October 2023 (Reference 6) to ascertain the information needed to support its review of the application and to develop requests for additional information (RAIs), as needed. Following the regulatory audit, the licensee submitted a supplemental letter dated January 11, 2024, which included additional information resulting from the audit. On April 18, 2024, the staff issued an audit summary (Reference 7).

The supplemental letters dated January 11 and April 24, 2024, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on July 11, 2023 (88 FR 44166).

2.0 REGULATORY EVALUATION

2.1 Regulatory Review

2.1.1 Applicable Regulations

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 provides the general provisions for "Domestic Licensing of Production and Utilization Facilities." The general provisions include but are not limited to establishing the regulatory requirements that a licensee must adhere to for the submittal of a license application. The NRC staff has identified the following applicable sections within 10 CFR Part 50 for the staff's review of a licensee's application to adopt TSTF-505, Revision 2:

- 10 CFR 50.36, "Technical Specifications," paragraphs (c)(2), "Limiting conditions for operation," and (c)(5), "Administrative controls"

- 10 CFR 50.55a, "Codes and standards," paragraph (h), "Protection and safety systems"

- 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule)

2.1.2 Regulatory Guidance

NRC regulatory guides (RGs) provide one way to ensure that the codified regulations continue to be met. The NRC staff considered the following guidance, and industry guidance endorsed by the NRC, during its review of the proposed changes:

- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (Reference 8) and RG 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," December 2020 (Reference 9).

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 and Revision 3, January 2018 (References 10 and 11, respectively).

- RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," May 2011 and RG 1.177, Revision 2, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, January 2021 (References 12 and 13 respectively).

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs [Probabilistic Risk Assessments] in Risk-Informed Decisionmaking," March 2017 (Reference 14).

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), section 16.1, "Risk-Informed Decision Making: Technical Specifications," March 2007 (Reference 15) and section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (Reference 16).

- Nuclear Energy Institute (NEI) Topical Report NEI 06-09 Revision 0-A (NEI 06-09-A), “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines,” dated October 2012 (Reference 17), provides guidance for risk-informed TSs. The NRC staff issued a final model SE approving NEI 06-09 on May 17, 2007 (Reference 18).

The licensee’s submittal cites various revisions of RG 1.200, RG 1.174, and RG 1.177. The RGs have been updated to Revision 3 of RGs 1.200 and 1.174, and Revision 2 for RG 1.177. The updates do not include any technical changes that would impact the consistency with NEI 06-09-A; therefore, the NRC staff finds the updated revisions to the RGs also applicable for use in the licensee’s adoption of TSTF-505, Revision 2.

2.2 Description of the RICT Program

The TS LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee must shut down the reactor or follow any remedial or required action (e.g., testing, maintenance, or repair activity) permitted by the TSs until the condition can be met. The remedial actions (i.e., ACTIONS) associated with an LCO contain Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s) (CT). The CTs are referred to as the “front stops” in the context of this SE. For certain conditions, the TSs require exiting the Mode of Applicability of an LCO (e.g., shut down the reactor).

The licensee’s submittal requested approval to add a RICT Program to the Administrative Controls section of the TSs, and modify selected CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09-A. Consistent with table 1 of TSTF-505, Revision 2, for Conditions requiring additional technical justification, the licensee provided several plant-specific LCOs and associated Actions for which ANO-2 proposed to be included in the RICT Program, along with additional justification. The NRC staff review of these variations and the justification is provided in section 3.2.1 of this SE.

The licensee is proposing no changes to the design of the plant or any operating parameter, and no new changes to the design basis in the proposed changes to the TSs. The effect of the proposed changes, when implemented, will allow CTs to vary based on the risk significance of the given plant configuration (i.e., the equipment out of service at any given time), provided that the system(s) retain(s) the capability to perform the applicable safety function(s) without any further failures (e.g., one train of a two-train system is inoperable). These restrictions on inoperability of all required trains of a system ensure that consistency with the defense-in-depth philosophy is maintained by following existing guidance when the capability to perform TS safety function(s) is lost.

The proposed RICT Program uses plant-specific operating experience for component reliability and availability data. Thus, the allowances permitted by the RICT Program are directly reflective of actual component performance in conjunction with component risk significance.

3.0 TECHNICAL EVALUATION

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to demonstrate that the proposed licensing basis changes meet the five key principles provided in section C of RG 1.174, Revision 3, and the three-tiered approach outlined in section C of RG 1.177, Revision 2. These key principles and tiers are:

Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption….

Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.

Principle 3: The proposed licensing basis change maintains sufficient safety margins.

Principle 4: When the proposed licensing basis changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

- Tier 1: PRA Capability and Insights
- Tier 2: Avoidance of Risk-Significant Plant Configurations
- Tier 3: Risk-Informed Configuration Risk Management

Principle 5: The impact of the proposed licensing basis change should be monitored by using performance measures strategies.

3.1 Method of NRC Staff Review

Each of the key principles and tiers are addressed in NEI 06-09-A and approved in the final model SE issued by the NRC for TSTF-505, Revision 2. NEI 06-09-A provides a methodology for extending existing CTs, and to thereby delay exiting the operational mode of applicability or taking Required Actions if risk is assessed and managed within the limits and programmatic requirements established by a RICT Program. The NRC staff's evaluation of the licensee's proposed use of RICTs against the key safety principles of RGs 1.174 and 1.177 is discussed below.

3.2 Review of Key Principles

3.2.1 Key Principle 1: Evaluation of Compliance with Current Regulations

Paragraph 50.36(c)(2) of 10 CFR requires that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TS until the condition can be met.

The CTs in the current TSs were established using experiential data, risk insights, and engineering judgement. The RICT Program provides the necessary administrative controls to permit extension of CTs and, thereby, delay reactor shutdown or Required Actions if risk is assessed and managed appropriately within specified limits and programmatic requirements, and the safety margins and defense in depth remain sufficient. The option to determine the extended CT in accordance with the RICT Program allows the licensee to perform an integrated evaluation in accordance with the methodology prescribed in NEI 06-09-A and proposed TS 6.5.20, “Risk Informed Completion Time Program.” The RICT is limited to a maximum of 30 days (termed the “backstop”).

The typical CT is modified by the application of the RICT Program as shown in the following example. The changed portion is indicated in italics.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days *OR* *In accordance with the Risk Informed Completion Time Program*

In attachment 1, attachment 5, and enclosure 1, “List of Revised Required Actions to Corresponding Probabilistic Risk Analysis (PRA) Functions,” to the LAR, as supplemented, the licensee provided a list of the TSs, associated LCOs, and Required Actions for the CTs that included modifications and variations from the approved TSTF-505. The modifications and variations consisted of proposed changes to the Required Actions and CTs. Furthermore, consistent with table 1 of TSTF-505, Revision 2, for ANO-2 TS 3.3.2.1, “Engineered Safety Feature Actuation System Instrumentation”; TS 3.5.2, “ECCS [Emergency Core Cooling System] Subsystems – Tavg ≥ 300°F [degrees Fahrenheit]”; TS 3.6.1.3, “Containment Air Locks”; TS 3.6.2.1, “Containment Spray System”; TS 3.6.2.3, “Containment Cooling System”; and TS 3.7.1.5, “Main Steam Isolation Valves”; in sections 2.1–2.6 of enclosure 1 to the LAR, as supplemented, the licensee included additional technical justification to demonstrate the acceptability for including these TSs in the RICT Program. The NRC staff reviewed the proposed changes to the TSs, associated LCOs, Required Actions, and CTs provided by the licensee for the scope of the RICT Program and concluded, with the incorporation of the RICT Program, that the required performance levels of equipment specified in LCOs are not changed and only the required CTs for the Required Actions are modified, such that 10 CFR 50.36(c)(2) will continue to be met.

The licensee identified in section 2.3, “Optional Changes and Variations,” of attachment 1 to the LAR, variations from TSTF-505, Revision 2, for which the licensee is proposing to apply the RICT Program.

Based on the above review, the NRC staff concludes that the TSs, as amended by the proposed changes, will continue to meet the requirements of 10 CFR 50.36(c)(2) because the LCOs will continue to state the lowest functional capability or performance levels of equipment required for safe operation of the facility. The TSs will continue to stipulate that if an LCO is not met, the facility must be shut down, or other acceptable remedial actions must be taken. The staff concludes that the remedial actions, as amended by the proposed change, will ensure that facility operation remains safe during the time the LCOs are not met. Therefore, the proposed changes to the TSs are acceptable.

Based on the discussion provided above, the NRC staff finds that the proposed RICT Program provided in section 2.0 of this SE, LCOs, Required Actions, and CTs meet the first key principle of RGs 1.174 and 1.177.

3.2.2 Key Principle 2: Evaluation of Defense in Depth

In RG 1.174, Revision 2, the NRC identified the following considerations used for evaluation of how the licensing basis change is maintained for the defense-in-depth philosophy:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential CCFs [common cause failures].
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

The licensee requested to use the RICT Program to extend the existing CTs for the respective TS LCOs described in the LAR, as supplemented. For the TS LCOs in the LAR, as supplemented, the licensee provided a description and assessment of the redundancy and diversity for the proposed changes. The NRC staff's evaluation of the proposed changes for these LCOs assessed ANO-2's redundant or diverse means to mitigate accidents to ensure consistency with the plant licensing basis requirements using the guidance in RG 1.174, RG 1.177, and TSTF-505, to ensure adequate defense in depth (for each of the functions) to operate the facility in the proposed manner (i.e., that the changes are consistent with the defense-in-depth criteria).

Enclosure 1 to the LAR, as supplemented, provided information supporting the ANO-2 evaluation of the redundancy, diversity, and defense-in-depth for each TS LCO and TS Required Action as it relates to instrumentation and controls (I&C) and electrical power systems. The NRC staff confirmed that for the following TS LCOs, the above defense-in-depth criteria were applicable, except for the criteria for maintaining multiple fission product barriers:

- TS 3.3.1, "Reactor Protective Instrumentation"
- TS 3.3.2, "Engineered Safeguards Actuation System Instrumentation"
- TS 3.8.1, "AC [Alternating Current] Sources– Operating"; Conditions a, b, c, d and e

- TS 3.8.2, “AC Distribution - Operating”
- TS 3.8.2.3, “DC [Direct Current] Sources – Operating”; Condition b

For the TS LCOs specific to I&C (i.e., TS 3.3, “Instrumentation,” specifically: TSs 3.3.1 and 3.3.2), the NRC staff reviewed the specific trip logic arrangements, redundancy, backup systems, manual actions, and diverse trips specified for each of the protective safety functions and associated instrumentation, as described in the associated safety analysis report (SAR) (Reference 20) sections, and as reflected in the LAR, as supplemented. The staff verified that, in accordance with the ANO-2 SAR Amendment 30 and equipment and actions credited in enclosure 1 to the LAR, as supplemented, in all applicable operating modes, the affected protective feature would perform its intended function by ensuring the ability to detect and mitigate the associated event or accident when the CT of a channel is extended. Furthermore, the staff concludes that there is sufficient redundancy, diversity, and defense-in-depth, to protect against CCFs and potential single failure for the ANO-2 instrumentation systems evaluated in LAR enclosure 1, as supplemented, during a RICT. There is at least one diverse means specified by the licensee for initiating mitigating action for each accident event, thus providing defense-in-depth against a failure of instrumentation during the RICT for each TS LCO. The staff confirmed that the defense in depth specified by the licensee does not overly rely on manual actions as the diverse means; therefore, there is not over-reliance of programmatic activities as compensatory measures. Therefore, the staff finds that the intent of the plant’s design criteria (e.g., safety functions) for the above TS LCOs related to I&C are maintained.

ANO-2 SAR section 8.3.1.1.2, “Unit Auxiliary Transformer, Startup Transformers and 6900-Volt Systems,” states that AC offsite power connects to the Class 1E onsite power system through 4.16 kilovolt (kV) startup transformer (SU) 3 with SU 2 being an alternate offsite power source, specifically associated with proposed changes to TS LCO 3.8.1, Conditions a, c, and d, and TS LCO 3.8.2.1, concerning offsite power. SAR sections 8.3.1.1.3, “4,160-volt Auxiliary System,” and 8.3.1.1.8.6, “Redundant Bus Separation,” states the Class 1E onsite power system consists of two engineered safety features (ESF), redundant 4,160-volt buses, each backed by its diesel generator (DG) (specifically associated with proposed changes to TS LCO 3.8.1, Condition b and TS LCO 3.8.2.1, concerning DGs).

ANO-2 SAR section 8.3.1.1.3, states that AC offsite power connects to the Class 1E onsite power system through 4.16 kV SU 1 with SU 2 being an alternate offsite power source, which is applicable to the proposed changes to STS 3.8.1, Conditions a, c, and d and STS 3.8.9, Condition a concerning offsite power. The Class 1E onsite power system consists of two fully redundant buses (one bus per train), which are backed by DGs for safe shutdown. This is applicable to proposed changes to STS 3.8.1, Conditions b and d, and STS 3.8.9, Condition a, for the onsite power system and its AC sources. SAR section 8.3.2.1.3, “DC Control Centers,” also states that the 125 volt (V) DC system consists of two redundant DC buses with only one required for its safe shutdown. This is applicable to the proposed change to STS 3.8.4, Condition a, and STS 3.8.9, Condition c concerning the 125 V DC system. SAR section 8.3.1.1.6, “120-Volt Uninterruptable AC Power System,” shows that the 120 VAC (volt alternating current] Vital AC system has four redundant distribution panels (two per train) with each panel supplied by one inverter, but only two inverters associated with one train are required for safe shutdown. This is applicable to the proposed changes to STS 3.8.7, Condition and STS 3.8.9, Condition b.

SAR section 8.3.2, “DC Power Systems,” states that the Class 1E 125 V DC system consists of two independent, physically and electrically separated 125 V batteries (one battery and two battery chargers (one normally operating and one spare per train)) with only one train required for safe shutdown, specifically associated with proposed changes to TS LCO 3.8.2.3.b. SAR section 8.3.1.1.6, states that the 120 volt uninterruptible AC power system consists of six inverters and four distribution panels (one per channel) with each panel supplied by one inverter, but only two inverters associated with two channels (see SAR table 8.3-10, “120-Volt Vital AC System Single Failure Analysis”) are required for safe shutdown, specifically associated with TS LCO 3.8.2.1 concerning 120 V AC vital buses.

The NRC staff also reviewed the electrical power systems design for a potential LOF for each proposed electrical RICT based on TSTF-505 and did not identify a LOF for any electrical power system. The staff reviewed the LAR and its supplements: (1) to verify that each affected electrical LCO can be entered voluntarily or involuntarily based on NEI 06-09-A; and (2) to evaluate if the affected electrical power systems for those LCOs could perform their safety functions (assuming no additional failures other than for the LCO being implemented) for the proposed RICTs. Based on its evaluation to verify no LOF for any electrical proposed RICT, the staff finds that the ANO-2 electrical power systems would function as intended for the proposed TS changes.

The NRC staff verified that the design success criteria in LAR table E1-1, “In Scope TS/LCO Conditions to Corresponding PRA Functions,” for each of the electrical TS 3.8 LCO Conditions reflect the minimum operable electrical power sources to support their safety functions to mitigate postulated design-basis accidents, safely shutdown the reactor, and maintain the reactor in a safe shutdown condition. The staff also verified that there are RICT estimates for each of those TS 3.8 LCO Conditions in LAR table E1-2, “In Scope TS/LCO Conditions RICT Estimate,” consistent with NEI 06-09-A.

In enclosure 12, “Risk Management Action Examples,” to the LAR, the licensee provided examples of risk management actions (RMAs) that are representative of actual RMAs that may be considered during a RICT Program entry for any of the proposed changes to TS 3.8 LCO Conditions to reduce the risk impact and ensure adequate defense in depth. The NRC staff evaluated the RMA examples provided in enclosure 12, section 4 and determined they had the required level of detail and additional RMAs identified within them to reduce the risk impact and ensure adequate defense in depth, including the electrical example for an inoperable DG. Based on that review, the staff determined that those examples provide reasonable assurance that the actual RMAs implemented to monitor and control risk for each LCO (for specific structures, systems, and components (SSC(s))) will be of similar quality and specific to that LCO.

The NRC staff reviewed the licensee’s proposed electrical TS LCO changes and supporting documentation. Based on the evaluations above, the staff finds that each LCO’s reduced redundancy, the CT extensions, as allowed by the RICT Program, are acceptable because (a) the capacity and capability of the remaining operable electrical systems to perform their safety functions (assuming no additional failures) is maintained, and (b) the licensee’s identification and implementation of RMAs as compensatory measures, in accordance with the RICT Program, would be effective.

The NRC staff notes that while in a TS LCO condition, the redundancy of the function will be temporarily relaxed and, consequently, the system reliability would be degraded accordingly. The staff examined the design information from the ANO-2 SAR and the risk-informed TS LCO conditions for the affected safety functions. Based on this information, the staff confirmed that under any given design-basis accident evaluated in the ANO-2 SAR, the affected protective features maintain adequate defense in depth.

Considering that the CT extensions will be implemented in accordance with the NEI 06-09-A guidance, which also considers RMAs and the redundancy of the offsite and onsite power system, the NRC staff finds that the plant will maintain adequate defense in depth. Therefore, the staff finds that the TS LCOs proposed by the licensee in attachment 5 to the LAR, as supplemented, are acceptable for the RICT Program.

The NRC staff reviewed all the TS LCOs proposed by the licensee in attachment 5 to the LAR, as supplemented, and concludes that the proposed changes do not alter the ways in which the ANO-2 systems fail, do not introduce new CCF modes, and the system independence is maintained.

The NRC staff finds that extending the CTs associated with the TS LCOs proposed by the licensee in attachment 5 to the LAR, as supplemented, with the RICT Program following loss of redundancy, but maintaining the capability of the system to perform its safety function, is an acceptable reduction in defense-in-depth during the proposed RICT period provided that the licensee identifies and implements compensatory measures in accordance with the RICT Program during the extended CT.

Based on the above, the NRC staff finds that the licensee's proposed changes are consistent with the NRC-endorsed guidance described in NEI 06-09-A and satisfy the second key principle in RGs 1.177 and 1.174. Additionally, the staff concludes that the changes are consistent with the defense-in-depth philosophy as described in RG 1.174.

3.2.3 Key Principle 3: Evaluation of Safety Margins

Paragraph 50.55a(h) of 10 CFR requires in part, that "[p]rotection systems of nuclear power reactors of all types must meet the requirements specified in this paragraph." Section 2.2.2, "Technical Specification Change Maintains Sufficient Safety Margin (Principle 3)," of RG 1.177 states, in part, that sufficient safety margins are maintained when:

a. Codes and standards … or alternatives approved for use by the NRC are met….

b. Safety analysis acceptance criteria in the final safety analysis report (FSAR) are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties….

The licensee is not proposing to change any quality standard, material, or operating specification in this application. In the LAR, the licensee proposed to add a new program, "Risk Informed Completion Time Program," in section 6.0, "Administrative Controls," of the ANO-2 TSs, which requires adherence to NEI 06-09-A.

The NRC staff evaluated the effect on safety margins when the RICT is applied to extend the CT up to a backstop of 30 days in a TS condition with sufficient trains remaining operable to fulfill the TS safety function. Although the licensee will be able to have design-basis equipment out of service longer than the current TSs allow, any increase in unavailability is expected to be insignificant and is addressed by the consideration of the single failure criterion in the design-basis analyses. Acceptance criteria for operability of equipment are not changed and, if sufficient trains remain operable to fulfill the TS safety function, the operability of the remaining train(s) ensures that the current safety margins are maintained. The staff finds that if the specified TS safety function remains operable, sufficient safety margins would be maintained during the extended CT of the RICT Program.

Safety margins are also maintained if PRA functionality is determined for the inoperable train, which would result in an increased CT. Credit for PRA functionality, as described in NEI 06-09-A, is limited to the inoperable train, loss of offsite power (LOOP), or component.

Based on the above, the NRC staff finds that the design-basis analyses for ANO-2 remain applicable and unchanged, that sufficient safety margins would be maintained during the extended CT, and that the proposed changes to the TSs do not include any change in the standards applied or the safety analysis acceptance criteria. The staff concludes that the proposed changes meet 10 CFR 50.55a(h) and, therefore, the third key principle of RGs 1.177 and 1.174.

3.2.4 Key Principle 4: Change in Risk Consistent with the Safety Goal Policy Statement

Proposed TS 6.5.20, “Risk Informed Completion Time Program,” states, in part, that the RICT “must be implemented in accordance with NEI 06-09-A, Revision 0, ‘Risk-Managed Technical Specifications (RMTS) Guidelines.’”

NEI 06-09-A provides a methodology for a licensee to evaluate and manage the risk impact of extensions to TS CTs. Permanent changes to the fixed TS CTs are typically evaluated by using the three-tiered approach described in SRP section 16.1; RG 1.177, Revision 2; and RG 1.174, Revision 3. This approach addresses the calculated change in risk as measured by the change in core damage frequency (CDF) and large early release frequency (LERF), as well as the incremental conditional core damage probability and incremental conditional large early release probability; the use of compensatory measures to reduce risk; and the implementation of a configuration risk management program (CRMP) to identify risk-significant plant configurations.

The NRC staff evaluated the licensee’s processes and methodologies for determining that the change in risk from implementation of RICTs would be small and consistent with the intent of the Commission’s Safety Goal Policy Statement.[1] In addition, the staff evaluated the licensee’s proposed changes against the three-tiered approach in RG 1.177, Revision 2, for the licensee’s evaluation of the risk associated with a proposed TS CT change. The results of the staff’s review are discussed below.

3.2.4.1 Tier 1: PRA Capability and Insights

Tier 1 evaluates the impact of the proposed changes on plant operational risk. The Tier 1 review involves two aspects: (1) scope and acceptability of the PRA models and their application to the proposed changes, and (2) a review of the PRA results and insights described in the licensee’s application.

In enclosure 2, “Information Supporting Consistency with Regulatory Guide 1.200, Revision 2,” and enclosure 4, “Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PRA Models,” to the LAR, the licensee identified the following modeled

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[1] Commission’s Safety Goal Policy Statement, “Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,” published in the *Federal Register* on August 4, 1986 (51 FR 28044), as corrected, and republished, on August 21, 1986 (51 FR 30028).

hazards and alternate methodologies that the licensee proposed to be used in the ANO-2 RICT Program to assess the risk contribution for extending the CT of a TS LCO:

- Internal Events PRA (IEPRA) model (includes internal floods)
- Internal Fire Events PRA (FPRA) model
- Seismic Hazard: a CDF penalty of 5.45×10^{-6} per year, and a LERF penalty of 2.59×10^{-6} per year
- Extreme Winds and Tornado Missile Hazards: a CDF penalty of 5×10^{-6} per year and a LERF penalty of 5×10^{-7} per year
- Other External Hazards: screened out from RICT Program based on appendix 6A of the American Society of Mechanical Engineers / American Nuclear Society (ASME/ANS) PRA Standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S 2008, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (ASME/ANS 2009 PRA Standard) (Reference 21)

3.2.4.1.1 PRA Scope

The guidance in RG 1.174, Revision 2, states that "[t]he scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process." The NRC's SE for NEI 06-09-A states that the PRA models should conform to the guidance in RG 1.200, Revision 1. The current version is RG 1.200, Revision 3, which clarifies the current applicable ASME/ANS 2009 PRA Standard. RG 1.200, Revision 3, is a recent update that does not include any technical challenges that would impact the plant's consistency with NEI 06-09-A; therefore, RG 1.200, Revision 2 is also acceptable for the implementation of the RICT Program. For external hazards for which a PRA is not available, the guidance in NEI 06-09-A allows for the use of bounding analysis of the risk contribution of the hazard for incorporation into the RICT calculation or justification for why the hazard is not significant to the RICT calculation.

The NRC staff evaluated the PRA acceptability information provided by the licensee in enclosure 2 to the LAR, including industry peer review results and the licensee's self-assessment of the PRA models for internal events, including internal flooding, and fire, against the guidance in RG 1.200, Revision 2. The licensee screened out all external hazard events, except for seismic and extreme winds and tornado missile, as described in section 3.2.4.1.3 of this SE, as insignificant contributors to the RICT calculations. The ANO-2 PRA model with modifications is used as the CRMP model, as described in section 3.2.4.1.7 of this SE. In addition, the licensee provided a bounding estimate of the seismic and tornado missile CDFs and LERFs and will include those CDF and LERF values, per sections 3.9 and 4.2 of enclosure 4 to the LAR, in the change-in-risk used to calculate RICTs consistent with the guidance in NEI 06-09-A.

The NRC staff finds that the ANO-2 scope of modeled PRA hazards, and those hazards for which a modeled PRA is not available where the licensee has proposed use of alternative methods, are commensurate with the RICT application for use in the integrated decision-making process, consistent with RG 1.174, Revision 3.

3.2.4.1.2 Evaluation of PRA Acceptability for Internal Events and Internal Fires

IEPRA (Includes Internal Flooding)

In enclosure 2, section 3, “Scope and Technical Adequacy of ANO-2 Internal Events and Internal Flooding PRA Model,” to the LAR, the licensee explains that the IEPRA model was subjected to a full-scope peer review in July 2008 against RG 1.200, Revision 1. The licensee conducted a “ANO-2 Technical Specification TSTF-505 Cross-Reference,” self-assessment of the changes to the supporting requirements of the ASME/ANS PRA standards made between RG 1.200, Revision 1, and RG 1.200, Revision 2. The NRC staff concluded in the SE for the licensee’s TSTF-425 application (Reference 22) that the licensee self-assessment addressed changes between the two regulatory guides.

Subsequently, the licensee conducted a focused-scope peer review (FSPR) of the internal flooding model in February/March 2017 and in August 2019 a FSPR of the LERF PRA model, both against RG 1.200, Revision 2. An IEPRA focused-scope peer review was conducted in December 2021, against RG 1.200, Revision 3. Afterward, the licensee conducted several independent assessments in September/October of 2019 and May 2022 to close the finding-level facts and observations (F&Os) using the Appendix X process documented in the NEI letter to the NRC “Final Revision of Appendix X to NEI 05-04/07-12/12-16, ‘Close-out of Facts and Observations,” dated February 21, 2017 (Reference 23). All finding-level F&Os were reviewed and closed using this NRC-accepted process. Hence, the LAR does not identify any open finding-level F&Os.

The NRC staff finds that the ANO-2 IEPRA (that includes internal flooding) was appropriately peer reviewed consistent with RG 1.200, Revision 2, and that all finding-level F&Os have been closed consistent with the Appendix X process guidance, as accepted, with conditions by the staff. Therefore, the staff concludes that the IEPRA (that includes internal flooding) is acceptable for use in the RICT Program.

Internal FPRA

In enclosure 2, section 4, “Scope and Technical Adequacy of ANO-2 Fire PRA Model,” to the LAR, the licensee confirmed that the ANO-2 internal FPRA model received a full-scope peer review in June 2009 using the ASME/ANS 2009 PRA Standard, and RG 1.200, Revision 2. Subsequent to the peer review, focused-scope peer reviews were conducted in 2011, 2012, 2014, and 2016. A subsequent independent assessment for closure of F&Os using the Appendix X process, as accepted, with conditions by the NRC staff, was performed in September/October 2019 and December 2021, which resulted in closure of all finding-level F&Os. Hence, the LAR does not identify any open finding-level F&Os.

The NRC staff finds that the ANO-2 FPRA was appropriately peer reviewed consistent with RG 1.200, Revision 2, and that all finding-level F&Os have been closed consistent with the Appendix X process guidance, as accepted, with conditions by the staff. Therefore, the staff concludes that the FPRA is acceptable for use in the RICT Program.

3.2.4.1.3 Evaluation of External Hazards

Evaluation of Seismic Hazard

The licensee's approach for including the seismic risk contribution in the RICT calculation is to add a penalty seismic CDF and a penalty seismic LERF to each RICT calculation. The proposed CDF estimate is based on using the plant-specific seismic hazard curves developed in response to the Near-Term Task Force (NTTF) Recommendation 2.1 (Reference 24), and a plant-level high confidence of low probability of failure (HCLPF) capacity of 0.3g referenced to peak ground acceleration (PGA). The uncertainty parameter for seismic capacity was represented by a composite beta factor (β_c) of 0.4. The calculated seismic CDF penalty is 5.45×10^{-6} per year. The NRC staff finds that the method to determine the baseline seismic CDF is acceptable because it is consistent with the approach used in NRC Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (Reference 25). The staff performed an independent convolution using the input parameters identified by the licensee to confirm the proposed seismic CDF penalty.

Concerning the proposed seismic LERF estimate, the licensee explains in the LAR that an estimate of the seismic LERF is obtained by convolving the estimated seismic CDF (as described above) with a limiting fragility for containment integrity, also assumed to be 0.3g PGA HCLPF. The calculated seismic LERF is 2.59×10^{-6} per year. The NRC staff finds that the licensee's approach to determining that a seismic LERF estimate is acceptable because the use of a 0.3g PGA HCLPF as the limiting fragility for containment integrity is conservative.

The licensee addressed the incremental risk associated with seismic-induced LOOP in its application dated April 5, 2023. The seismic LOOP frequency across the entire hazard interval is 6.9×10^{-5} per year. The seismic LOOP frequency across the entire hazard interval is about 5.2 percent of the total internal events 24-hour non-recovered LOOP frequency of 1.3×10^{-3} per year. The NRC staff evaluated the licensee's analysis and finds that it adequately addresses the impact of a seismically-induced LOOP on risk and that the exclusion of the impact of a seismically-induced LOOP on risk from the non-recovered LOOP frequency has an insignificant impact on the RICT calculations.

The NRC staff finds that, during RICTs for SSCs credited in the design basis to mitigate seismic events, the licensee's proposed methodology captures the risk associated with seismically induced failures of redundant SSCs because such SSCs are assumed to be fully correlated. By assuming full correlation, the seismic risk for those RICTs will not increase if one of the redundant SSCs is unavailable because simultaneous failure of all redundant trains would be assumed in a seismic PRA. During RICTs for SSCs that are not credited in seismic events, the proposed methodology for considering seismic risk contributions is conservative because the seismically induced failure of such SSCs would not result in a risk increase associated with the plant configuration during the RICT, but the seismic penalty is still included in the calculation. During RICTs for SSCs that are credited in seismic events, the proposed methodology is acceptable for this application because the plant-level HCLPF value used for the RICT calculations provides a conservative estimate of HCLPF values for all the credited SSCs.

In summary, the NRC staff finds the licensee's proposal to use the seismic CDF penalty of 5.45×10^{-6} per year, and a seismic LERF penalty of 2.59×10^{-6} per year to be acceptable for the licensee's RICT Program for ANO-2, because (1) the licensee used the most current site-specific seismic hazard information, (2) the licensee used an acceptably low plant level HCLPF value of 0.3g and a combined beta factor of 0.4 consistent with the information for ANO-2 in the GI-199 evaluation, (3) the licensee determined a seismic LERF penalty based on its estimate of seismic CDF combined with using a containment integrity fragility of 0.3g PGA HCLPF, and (4) adding baseline seismic risk to RICT calculations, which assumes the fully correlated failures, is acceptable for this application.

Evaluation of Extreme Winds and Tornado Hazards

In section 4 of enclosure 4 to the LAR, as supplemented, the licensee discusses its evaluation of the extreme wind and tornado impact on this application. The licensee concluded that the extreme winds hazard can generally be screened from consideration for the TSTF-505 application because the frequency of tornadoes having wind speeds that exceed the design basis of 300 miles per hour is much less than 1 x 10^{-6} per year; tropical storms such as hurricanes are not a concern for ANO-2 given its inland location; and the risk from straight winds is bounded by that from tornadoes. Although the tornado missiles hazard is screened for total risk, it does not screen for all configurations; therefore, the licensee proposed a penalty factor to account for tornado missile risk in the RICT.

In its supplemental letter dated January 11, 2024, the licensee provided additional information on the risk evaluation of tornado missiles and the development of the associated penalty factors. The proposed tornado missile CDF and LERF penalty estimates are based on using the ANO-2 Conservative Tornado Risk Model (CTRM), which is developed by making changes to the ANO-2 Tornado Missile Risk Evaluator (TMRE) model that was previously approved by the NRC staff by letter dated June 30, 2020 (Reference 26). In its letter dated January 11, 2024, the licensee provides changes that were made to the TMRE to improve realism in the CTRM and explains that the most significant change was the elimination of certain targets that were determined to be adequately protected from tornado-generated missiles and therefore, conforming, after the approval of the TMRE, based on additional plant-specific walkdowns. This change impacted 35 percent of the targets from those reported in the NRC SE issued by letter dated June 30, 2020, concerning the ANO LAR to incorporate TMRE into the ANO-2 licensing basis. Other significant changes between the TMRE and the CTRM were: (1) recent cable data from the recent FPRA model update that provided more realistic target data, (2) reduction in the degree of correlation of target failures in a single room due to a single missile by refining exposed equipment failure probabilities and impacted SSCs in accordance with NEI 17-02, “Tornado Missile Risk Evaluator (TMRE) Industry Guidance Document” (Reference 27) guidance (e.g., defining multiple failure scenarios rather than assume that all SSCs in certain rooms are failed by a single missile); and (3) incorporate realistic SSC fragilities for several targets using the Tornado Missile Probabilistic Risk Assessment (TORMIS) methodology (Reference 28).

To develop the penalty factors, the CTRM was quantified for all LCO configurations proposed to be included within the RICT Program and for several risk significant combinations of LCO configurations. For all but one plant configuration associated with LCOs to be included in the RICT Program, the licensee proposed a tornado missile CDF penalty of 5 x 10^{-6} per year, which was determined by the licensee to be bounding of all LCOs and plant configurations except for unavailability of the Red Train of the DC system (LCO 3.8.2.3, Condition b). For LCO 3.8.2.3, Condition b, the calculated RICT penalty would be 5.8 x 10^{-6} per year. However, the licensee proposed to apply the same CDF penalty of 5 x 10^{-6} per year, since the penalty value of 5.8 x 10^{-6} per year would only change the RICT by 1 minute. For all plant configurations associated with LCOs to be included in the RICT Program, the licensee proposed a tornado missile LERF penalty of 5 x 10^{-7} per year, which was determined by the licensee to be bounding for all LCOs and plant configurations.

The NRC staff reviewed the licensee's evaluation provided in section 4 of enclosure 4 to the LAR, as supplemented, and finds that the licensee's determination of CDF and LERF tornado missile penalties acceptable for this application because: (1) the approach used by the licensee to develop the penalties includes appropriate inputs and assumptions for this application, (2) the penalties bound the results of a tornado missile risk assessment for all LCOs encompassed by the RICT Program, and (3) the estimated LCO-specific tornado missile penalty factors would be added in their entirety to the delta-risk calculations for RICT determinations.

Evaluation of Other External Hazards

In addition to the seismic and the extreme winds and tornado hazards discussed above, the licensee confirmed that other external hazards for ANO-2 have insignificant contribution and proposed that these hazards be screened out from the RICT Program. For external floods, the licensee's conclusions in table E4-5, "Other External Hazards Disposition," of enclosure 4 to the LAR regarding insignificant risk contribution are based on the "Flooding Hazard Re-evaluation Report – Required Response for Near-Term Task Force (NTTF) Recommendation 2.1" and "Focused Evaluation for External Flooding" reports for ANO-2 (References 29 and 30, respectively). The licensee provided its assessment of other external hazard risk for the RICT Program in LAR enclosure 4. The hazards assessed in the LAR are those identified for consideration in non-mandatory appendix 6-A of the ASME/ANS 2009 PRA Standard, which provides a guide for identification of most of the possible external events for a plant site.

The NRC staff reviewed the information in the LAR, as supplemented, and finds that the contributions from external flooding and other external hazards have an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs because they either do not challenge the plant or they are bounded by the external hazards analyzed for the plant. Furthermore, the staff also finds that plant procedures exist to ensure that flood protection features will be available during RICTs to manage the external flooding risk in the RICT Program. For all other external hazards, the staff notes that the preliminary screening criteria and progressive screening criteria used and presented in LAR table E4-5 are the same criteria that were presented in supporting requirements for screening external hazards EXT-B1, EXT-B2, and EXT-C1 of the ASME/ANS 2009 PRA Standard.

3.2.4.1.4 PRA Results and Insights

The proposed change implements a process to determine TS RICTs rather than specific changes to individual TS CTs. NEI 06-09-A delineates that periodic assessment be performed of the risk incurred due to operation beyond the "front stop" CTs resulting from implementation of the RICT Program and comparison to the guidance of RG 1.174, Revision 3, for small increases in risk. In enclosure 5, "Baseline Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)," to the LAR, the licensee provided the estimated total CDF and LERF to demonstrate that they meet the 1E-4/year CDF and 1E-5/year LERF criteria of RG 1.174 consistent with the guidance in NEI 06-09-A, and that these guidelines will be satisfied for implementation of a RICT.

The licensee has incorporated NEI 06-09-A into the new proposed TS 6.5.20. The estimated current total CDF and LERF for ANO-2 PRAs meet the RG 1.174, Revision 3 guidelines;

therefore, the NRC staff finds that the PRA results and insights to be used by the licensee in the RICT Program will continue to be consistent with NEI 06-09-A.

3.2.4.1.5 Key Assumptions and Uncertainty Analyses

The licensee considered PRA modeling uncertainties and their potential impact on the RICT Program and identified, as necessary, the applicable RMAs to limit the impact of these uncertainties. In enclosure 9, “Key Assumptions and Sources of Uncertainty,” to the LAR, the licensee discussed the identification of key assumptions and sources of uncertainty, along with providing the dispositions for impact on the risk-informed application of applicable sensitivities. The licensee evaluated the ANO-2 PRA model to identify the key assumptions and sources of uncertainty for this application consistent with the RG 1.200, Revision 2, definitions, using sensitivity and importance analyses to place bounds on uncertain processes, to identify alternate modeling strategies, and to provide information to users of the PRA.

In response to APLA Question 01 in the LAR supplement dated January 11, 2024, the licensee identified several systems that contain digital I&C components (i.e., certain components of the reactor protection system (RPS), certain components of the engineered safety features actuation system (ESFAS), and common feedwater), but these are not explicitly modeled based on their minimal contribution to system unavailability or unreliability, which is consistent with supporting requirement SY-A15 of the of the 2009 ASME/ANS PRA standard. However, the licensee stated that some digital components are modeled. The NRC staff considers that there exists a level of uncertainty related to PRA modeling of digital components given the lack of industry data. In the supplemental letter dated January 11, 2024, the licensee provided the results of a sensitivity study for the uncertainty related to the modeled digital components that demonstrated that this source of uncertainty did not significantly impact any RICT calculations. The NRC staff finds that the ANO-2 PRA modeling of digital components does not impact this application.

In response to APLA Question 07 in the LAR supplemental letter dated January 11, 2024, the licensee expanded on the dispositions to certain of the identified key assumptions and sources of uncertainty, namely (1) LAR table E9-1, Item Number 12, “Containment sump/strainer performance,” (2) LAR table E9-2, Item No. 3, “FPRA Cable Selection,” and (3) LAR enclosure 9, section 4, “Assessment of Level 2 Epistemic Uncertainty Impacts.”

Regarding APLA Question 07a in the LAR supplemental letter dated January 11, 2024, FPRA Cable Selection, the licensee provided the results of a sensitivity study that revised the “always failed” assumption for the related SSCs to “always available.” The results of the sensitivity demonstrated that this source of uncertainty is minimal and that the assumption “always failed” is conservative. The NRC staff finds that the uncertainty related to unidentified cable selection does not impact this application.

Regarding APLA Question 07b in the LAR supplemental letter dated January 11, 2024, Level 2 Epistemic Uncertainty Impacts, specifically the ANO-2 use of the probability of burst (POB) calculation related to steam generator tube ruptures (SGTRs), the licensee performed a sensitivity study that used a factor of 10 increase in the POB value that demonstrated a 55 percent impact on the RICT calculation for TS LCO 3.7.1.5, “Each main steam isolation valve shall be OPERABLE.” In the supplemental letter dated January 11, 2024, the licensee describes the SGTR POB approach not as an assumption, but a consensus model approach that utilizes a statistical analysis of site-specific steam generator tube wear. However, the licensee does state that this statistical analysis is based on a small data set and therefore a level of uncertainty exists. A basis for this approach being a consensus model is that the POB is described in NUREG/CR-6365, “Steam Generator Tube Failures” (Reference 31). In its response to APLA RAI 01 in the LAR supplemental letter dated April 24, 2024, the licensee provided an updated sensitivity analysis using a factor of 3 multiplier, which represents the 95th confidence interval of the original ANO-2 site-specific analysis. When the factor of 3 multiplier was applied to the industry consensus data of $2.7x10^{-2}$, the results demonstrated a 16 percent impact on the RICT for TS 3.7.1.5. Subsequently, the licensee performed a site-specific update to its POB value and determined a POB of $1.4x10^{-3}$. When the factor of 3 multiplier is applied to site-specific value data, the sensitivity to POB is lowered to where it does not significantly impact the RICT for TS 3.7.1.5. The NRC staff determined that the ANO-2 site specific calculation for its POB and updated sensitivity analysis that the uncertainty related to the POB calculation does not impact this application.

Regarding APLA Question 07c in the LAR supplemental letter dated January 11, 2024, concerning containment sump/strainer performance, the licensee states that ANO-2 developed failure modes for both medium and large loss-of-coolant accidents with failure probabilities of 1E-04 and 2E-04, respectively. In the January 11, 2024 letter, the licensee also stated that it performed walkdowns to support the assumptions used in calculating the failure probabilities. Specifically, none of the reactor coolant pipes were above the recirculation sump strainers. The licensee has also installed three-dimensional box strainers to address the possibility of strainer clogging, which was found satisfactory by staff in the closeout of Generic Letter 2004-02 for ANO, Units 1 and 2 (Reference 32). The NRC staff determines that the installation of the box strainers and the ANO-2 PRA model incorporating probabilities that exclude the use of these strainers conservatively addresses this issue. Therefore, this issue does not impact the staff’s evaluation of this LAR.

Based on the NRC staff’s review of the licensee’s dispositions provided in enclosure 9 to the LAR, as supplemented, the staff finds that the licensee performed an adequate assessment to identify the potential sources of uncertainty, and that the identification of the key assumptions and sources of uncertainty was appropriate and consistent with the guidance in NUREG-1855, Revision 1 and associated Electric Power Research Institute (EPRI) Technical Report (TR)-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments” (Reference 33) and EPRI TR-1026511, “Practical Guidance of the Use on Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on Treatment of Uncertainty” (Reference 34). Therefore, the staff finds the licensee has satisfied the guidance in RG 1.177, Revision 2, and RG 1.174, Revision 3, and that the identification and treatment of assumptions and treatment of model uncertainties for risk evaluation of extended CTs is appropriate for this application and is consistent with the guidance in NEI 06-09-A, and therefore acceptable.

3.2.4.1.6 PRA Scope and Acceptability Conclusions

As stated in enclosure 2 to the LAR, the licensee has subjected the PRA models to the peer review processes and submitted the results of the peer review. The NRC staff reviewed the peer-review history, which included the results and findings, the licensee’s resolutions of peer review findings, and the identification and disposition of key assumptions and sources of uncertainty. The NRC staff concludes that: (1) the licensee’s PRA models are acceptable to support the RICT Program, and (2) the key assumptions for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2 and NUREG-1855, Revision 1. Additionally, the staff finds that the licensee’s approach for considering the impact of seismic events, non-seismic external hazards and other hazards using alternative methods is acceptable.

Based on the above conclusions discussed in sections 3.2.4.1.1 through 3.2.4.1.5 of this SE, the NRC staff finds that the licensee has satisfied the intent of Tier 1 in RG 1.177, Revision 2 and RG 1.174, Revision 3 for determining the PRA acceptability, and that the scope of the PRA models (i.e., IEPRA, FPRA, and the use of a bounding analysis for seismic events) is appropriate for this application.

3.2.4.1.7 Application of PRA Models in the RICT Program

The ANO-2 base PRA models that are determined to be acceptable in section 3.2.4.1.6 of this SE will be modified as an application-specific PRA model (i.e., CRMP tool), that will be used to analyze the risk for an extended CT. The CRMP model produces results (i.e., risk metrics) that are consistent with the NEI 06-09-A guidance. Throughout the entirety of the LAR and associated supplemental letters as discussed below, and specifically table E1-1, the licensee provided all information to support the requested LCO actions proposed for the ANO-2 RICT Program consistent with all the limitations and conditions prescribed in section 4.0 of NEI 06-09-A.

In LAR enclosure 8, “Attributes of the Real-Time Risk Model,” section 2, “Translation of Baseline PRA Model for Use in Configuration Risk,” the licensee explains that the CRMP model credits systems that are shared between units. In response to APLA Question 02 in the LAR supplemental letter dated January 11, 2024, the licensee identified and described all credited shared systems and equipment and described the PRA modeling for a dual unit event. The licensee included the common feedwater system, alternate AC diesel generator, cross tie to the ANO-2 4160 V vital buses 2A3 and 2A4 through 2A9 from the ANO-2 vital 4160 V buses, SU2, 500 kV grid, Instrument Air, backup DC power to non-vital buses 2A1, 2A2 (4160 V), and 2H1 and 2H2 (6900 V); and portable flexible equipment, included as part of their diverse and flexible coping strategies (FLEX) program. The NRC staff finds that the modeling of these shared systems in the CRMP model is acceptable because the calculated RICTs are not significantly impacted by over-crediting them in dual unit events.

Enclosure 8, section 2, to the LAR identifies several specific modifications that are made to the baseline PRA model to produce the CRMP model, or the real time risk (RTR) tool, that is used to make the RICT calculations. In response to APLA Question 03 in the supplemental letter dated January 11, 2024, the licensee provided additional details on how adjustments to the CRMP model are made to reflect changing conditions that could affect the model and associated RICT calculations, such as seasonal variations and time in the core cycle that could impact success criteria. The licensee stated that there are no seasonal variations that currently need to be accounted for in the CRMP model and that any identified need to account for seasonal variations in the future are addressed by the CRMP model update process. The licensee explained that the CRMP model does have settings for various emergent weather-related conditions that can be adjusted in real-time if needed, and that plant operators are trained to make these adjustments. Regarding time in the core cycle, the licensee further explained that the CRMP model to be used for the RICT Program will ensure that this assumption is treated conservatively. The NRC staff finds that the licensee’s CRMP model is in accordance with NEI 06-09-A with respect to the treatment of changing plant conditions, such as the weather and seasonal variations, because it is capable of being adjusted in real-time to account for changing plant conditions or assesses these conditions conservatively for the RICT calculations.

LAR enclosure 1, table E1-1, identifies each TS LCO proposed to be included in the RICT Program, describes whether the systems and components involved in the TS LCO are implicitly or explicitly modeled in the PRA, and compares the design basis and PRA success criteria. For certain TS LCO conditions, the table explains that the associated SSCs are not modeled in the PRAs but will be represented using a surrogate event that fails the function performed by the SSC.

In response to APLA Question 06a in the supplemental letter dated January 11, 2024, the licensee provided additional clarification for the LCO for TS 3.3.1.1, “Functional Unit 11.A - ‘Minimum of three channels of Reactor Protection System (RPS) Matrix Logic shall be operable,’ Action 1, ‘With the number of channels operable one less than required by the Minimum channels operable, restore within 48 hours.’” The licensee stated that the logic matrix trip signal de-energizes four logic matrix relays that in turn interrupt power to one of the four trip paths that de-energized the solid state relays (SSRs). It is these SSRs that will be used as the surrogate for this LCO. The NRC staff finds that the SSR surrogate bounds the function of the logic matrix relays and is consistent with the guidance of NEI 06-09-A.

In response to APLA Question 06b in the LAR supplemental letter dated January 11, 2024, the licensee provided additional clarification for the LCO for TS 3.3.1.1, “Functional Unit 14, ‘Two Control Element Assembly Calculators (CEACs) shall be operable,’ Action 6.a, ‘With one CEAC inoperable, restore within 7 days.’” The licensee stated that there are four sets of analog inputs to each bistable channel and the surrogate would be a limiting analog input that would fail one of the four inputs to each RPS channel. The NRC staff finds that the limiting analog input surrogate bounds the function of Functional Unit 14 and is consistent with the guidance of NEI 06-09-A.

In response to APLA Question 06c in the LAR supplemental letter dated January 11, 2024, the licensee provided additional clarification for the LCO for TS 3.3.2.1, “Engineered Safety Features Actuation System (ESFAS) Instrumentation,” “Functional Units 1.a, 2.a, 3.a, 4.a, 5.a, and 6.a, ‘Two sets of two Manual Trip buttons shall be operable,’” and “Functional Unit 8.a, ‘Two sets of two Manual Trip buttons per Steam Generator (SG) shall be operable,’ Action 9, ‘With one channel inoperable, restore channel within 48 hours.’” The licensee stated where the manual initiation is not modeled in the PRA the surrogate would be failing one train of the automatic function or master relay. The NRC staff finds that the automatic function or master relay surrogate bounds the function of the manual initiation and is consistent with the guidance of NEI 06-09-A.

In response to APLA Question 06d in the LAR supplemental letter dated January 11, 2024, the licensee provided additional clarification for the LCO for TS 3.3.2.1, “Functional Units 1.d.1, 2.c.1, 3.c.1, 4.c.1, 5.d.1, 6.c.1, and 8.d.1, ‘Minimum of three ESF Matrix Logic channels shall be operable,’ Action 12, ‘With one channel inoperable, restore channel within 48 hours.’” The licensee stated that the surrogate to be used is the downstream trip relay that would fail an ESFAS logic load group. The NRC staff finds that the downstream trip relay surrogate bounds the function of the ESFAS Units and is consistent with the guidance of NEI 06-09-A.

In response to APLA Question 08 in the LAR supplemental letter dated January 11, 2024, the licensee provided details on how the ANO-2 PRA modeling of FLEX addresses the NRC staff uncertainty concerns regarding equipment failure probabilities and operator actions. Entergy confirmed that the human reliability analysis methodology for FLEX operator actions addressed all of the staff concerns listed in its May 6, 2022, memorandum (Reference 35). Regarding equipment failure data, Entergy confirmed that the ANO-2 IEPRA incorporates the industry data provided in the Pressurized Water Reactor Owners Group (PWROG)-18042-NP, "Flex Equipment Data Collection and Analysis: (Reference 36) and will incorporate into the ANO-2 FPRA prior to implementing the RICT Program. In APLA RAI 02, the NRC staff requested the incorporation of PWROG-18042-NP data into the FPRA as an implementation item. In its response dated April 24, 2024, the licensee confirmed that the FPRA shall be updated using the FLEX equipment failure rates in PWROG-18042-NP prior to implementing the RICT Program for ANO-2 within the amendment implementation period. The NRC staff will include this item in its implementation statement on the license page of this amendment.

For emergent conditions in which the extent of condition evaluation for inoperable SSCs is not complete prior to exceeding the CT, the requirement in TSTF-505, Revision 2, is to either (a) numerically account for the increased probability of CCF or (b) to implement RMAs not already credited in the RICT calculation that support redundant or diverse SSCs that perform the functions of the inoperable SSCs and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs. The NRC staff finds that numerically accounting for an increased probability of failure, in accordance with RG 1.177, Revision 2, will shorten the estimated RICT based on the particular SSCs involved thereby limiting the time when a CCF could affect risk. Alternatively, implementing actions that can increase the availability of other mitigating SSCs or decrease the frequency of demand on the affected SSCs will decrease the likelihood that a CCF could affect risk. The staff finds that both methods minimize the impact of CCF because they either limit the exposure time, help ensure the availability of alternate SSCs, or decrease the probability of plant conditions requiring the safety function to be performed.

For planned conditions, the licensee states in LAR enclosure 8, section 6, that adjustments to CCF grouping and associated probabilities (ANO-2 uses alpha factors to calculate CCFs) are not necessary when a component is taken out of service for preventive maintenance because (1) "[t]he component is not out-of-service for reasons subject to a potential CCF…" and (2) "[t]he net failure probability for the in-service components includes the CCF contribution of the out-of-service component." The licensee also states, in part, that "the CCF events that are related to the out-of-service component are retained" and that this is conservative.

Section 3.3.6, "Common Cause Failure Consideration," of NEI 06-09-A states, in part, that "[f]or all RICT assessments of planned configurations, the treatment of common cause failures in the quantitative CRM [configuration risk management] tools may be performed by considering only the removal of the planned equipment and not adjusting common cause failure terms." However, RG 1.177 states that when a component is rendered inoperable in order to perform preventative maintenance, the CCF contributions in the remaining operable components should be modified to remove the inoperable component and to only include CCF of the remaining components. The NRC staff finds that the CCF contribution from the out-of-service component is conservatively retained in the following ways: (1) the independent failure rate used in the PRA models includes both independent and dependent failure events (i.e., the dependent failures should be subtracted from the total population of failures to calculate the independent failure rate) and (2) the CCF event probabilities that include the out-of-service component are retained. The staff also finds, however, that this simplification produces both conservative and non-conservative effects. The CCF probability estimates are uncertain and retaining precision in the calculation of these estimates using a more refined approach will not necessarily improve the accuracy of the results. Therefore, the staff finds that the licensee's method is acceptable because, consistent with NEI 06-09-A, the calculations reasonably include CCFs after removing one train for maintenance consistent with the accuracy of the estimates.

The NRC staff did not identify any insufficiencies in the information or the CRMP tool (RTR model) as described in the LAR, as supplemented. Furthermore, as stated in attachment 1 to the LAR, regarding the ANO-2 design criteria, the licensee stated that "[t]he proposed change does not change the design, configuration, or method of operation of the plant." The staff finds that the ANO-2 PRA models and CRMP tool used will continue to reflect the as-built, as-operated plant consistent with RG 1.200, Revision 2, for ensuring PRA acceptability is maintained. Therefore, the staff finds that the proposed application of the ANO-2 RICT Program is appropriate for use in the adoption of TSTF-505 for performing RICT calculations.

3.2.4.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

As described in RG 1.177, Revision 2, the second tier evaluates the capability of the licensee to recognize and avoid risk-significant plant configurations that could result if equipment, in addition to that associated with the proposed change, is taken out of service simultaneously or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The limits established for entry into a RICT and for RMA implementation are consistent with the NEI guidance of Nuclear Management and Resources Council (NUMARC) 93-01, Revision 4F (Reference 37), endorsed by RG 1.160, Revision 4 (Reference 38), as applicable to plant maintenance activities.

Based on the licensee's incorporation of NEI 06-09-A in the TSs as discussed in LAR attachment 1, the use of RMAs as discussed in LAR enclosure 12, and because the proposed changes are consistent with the Tier 2 guidance of RG 1.177, Revision 2, the NRC staff finds the licensee's RICT Program requirements and criteria are consistent with the principle of Tier 2 to avoid risk-significant configurations and, therefore, that its Tier 2 program is acceptable and supports the proposed implementation of the RICT Program.

3.2.4.3 Tier 3: Risk-Informed Configuration Risk Management

Tier 3 of RG 1.177, Revision 2, provides that a licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.

The proposed RICT Program establishes a CRMP, or RTR model, based on the underlying PRA models. In enclosure 8 to the LAR, the licensee explains the adjustments to PRA models (e.g., adjustments to maintenance unavailability) to ensure the proper use of models in the RTR model calculations. The RTR model is then used to evaluate configuration-specific risk for planned activities associated with the RMTS extended CT and emergent conditions that may arise during an extended CT. This required assessment of configuration risk, along with the implementation of compensatory measures and RMAs, is consistent with the principle of Tier 3 for assessing and managing the risk impact of out-of-service equipment.

In enclosure 8 to the LAR, the licensee confirmed that future changes made to the baseline PRA models and changes made to the online model (i.e., RTR) are controlled and documented by plant procedures. In enclosure 10, "Program Implementation," to the LAR, the licensee identified the attributes that the RICT Program procedures will address, which are consistent with NEI 06-09-A. The NRC staff finds that the licensee has identified appropriate administrative controls consistent with NEI 06-09-A and 10 CFR 50.36(c)(5).

The NRC staff reviewed the description of the training program provided in the LAR and concluded that the program is consistent with the training requirements set forth in NEI 06-09-A.

Therefore, the staff finds that the licensee has proposed acceptable administrative controls for the PRA and personnel implementing the RICT Program and will establish appropriate programmatic and procedural controls for its RICT Program, consistent with the guidance of NEI 06-09-A, section 3.2.1, “RMTS Process Control and Responsibilities.”

Based on the licensee’s incorporation of NEI 06-09-A in the TSs, as discussed in LAR attachment 1; use of RMAs, as discussed in LAR enclosure 12; and because the proposed changes are consistent with the Tier 3 guidance of RG 1.177, Revision 2, the NRC staff finds that the licensee’s Tier 3 program is acceptable and supports the proposed implementation of the RICT Program.

3.2.4.4 Key Principle 4 Conclusions

The licensee has demonstrated the technical acceptability and scope of its PRA models and alternative methods. This includes considering the impact of seismic events, non-seismic external hazards, and other hazards, and that the models can support implementation of the RICT Program for determining extensions to CTs. The licensee has made proper consideration of key assumptions and sources of uncertainty. The risk metrics are consistent with the approved methodology of NEI 06-09-A and the acceptance guidance in RG 1.177 and RG 1.174. The RICT Program will be controlled administratively through plant procedures and training and follows the NRC-approved methodology in NEI 06-09-A. The NRC staff concludes that the RICT Program satisfies the fourth key principle of RG 1.177 and is, therefore, acceptable.

3.2.5 Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring

RG 1.177, Revision 2 and RG 1.174, Revision 3, establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the availability of SSCs impacted by the change. Revision 3 of RG 1.174 states, in part, monitoring performed in conformance with the Maintenance Rule (10 CFR 50.65), can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. In enclosure 11, “Monitoring Program,” to the LAR, states that the SSCs in the scope of the RICT Program are also in the scope of 10 CFR 50.65 for the Maintenance Rule. The Maintenance Rule monitoring programs will provide for evaluation and disposition of unavailability impacts, which may be incurred from implementation of the RICT Program.

In response to APLA Question 05 in the LAR supplemental letter dated January 11, 2024, the licensee confirmed that the Maintenance Rule monitoring program incorporates the use of performance criteria to evaluate SSC performance as described in NUMARC 93-01, Revision 4F, endorsed by RG 1.160, Revision 4.

NEI 06-09-A specifies that the cumulative risk associated with the use of RMTS beyond the front-stop for equipment out of service is to be monitored. In enclosure 11, “Monitoring Program,” to the LAR, the licensee states that the cumulative risk is calculated at least every refueling cycle, not to exceed 24 months. The NRC staff finds that this periodicity is consistent with NEI 06-09-A.

The NRC staff concludes that the RICT Program satisfies the fifth key principle of RG 1.177 and RG 1.174 because: (1) as described in enclosure 11 to the LAR, the RICT Program will monitor the average annual cumulative risk increase as described in NEI 06-09-A, and use this average annual increase to ensure that the program, as implemented, meets RG 1.174 guidance for small risk increases: and (2) all affected SSCs are within the Maintenance Rule program, which is used to monitor changes to the reliability and availability of these SSCs.

3.3 Technical Conclusion

The NRC staff has evaluated the proposed changes against each of the five key principles in RG 1.177, Revision 2 and RG 1.174, Revision 3, and evaluated the optional variations from the approved TSTF-505 discussed in section 3.2.1 of this SE. The staff concludes that the changes proposed by the licensee satisfy the key principles of risk-informed decision-making identified in RG 1.174, and RG 1.177 and, therefore, the requested adoption of the proposed changes to the TSs and associated guidance is acceptable to assure the regulatory requirements of 10 CFR Part 50 identified in section 2.1 of this SE 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 May 28, 2024. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 July 11, 2023 (88 FR 44166), 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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Date: June 25, 2024

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 - ISSUANCE OF AMENDMENT NO. 333 RE: REVISION TO TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2, “PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b” (EPID L-2023-LLA-0052) DATED JUNE 25, 2024

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