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

April 29, 2015

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISE TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENT TO ELIMINATE MOVEMENT OF CONTROL ELEMENT ASSEMBLY 18 FOR THE REMAINDER OF OPERATING CYCLE 24 (TAC NO. MF5698)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 302 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 6, 2015, as supplemented by letter dated February 24, 2015.

The amendment requested to eliminate exercising Control Element Assembly (CEA) 18 for the remainder of operating Cycle 24, currently scheduled to end on September 20, 2015. The proposed amendment would modify a Note to TS Surveillance Requirement (SR) 4.1.3.1.2 such that CEA 18 may be excluded from SR performance for the remainder of Cycle 24.

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

Sincerely,

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 302 to NPF-6
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 302
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 February 6, 2015, as supplemented by letter dated February 24, 2015, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-6 is hereby amended to read as follows:

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

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 302, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications

3. The license amendment is effective as of its date of issuance and shall be implemented immediately.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Renewed Facility
Operating License No. NPF-6
Technical Specifications

Date of Issuance: April 29, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 302

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Renewed Facility Operating License No. NPF-6 and 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 1-18	3/4 1/18

(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. 302, 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.

ACTION: (Continued)

e. With more than one CEA misaligned from any other CEA in its group by more than 7 inches (indicated position), be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours.

4.1.3.1.2 Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days. (Note 1)

Note 1 - Movement of CEA 18 is not required for the remainder of Cycle 24. If an outage of sufficient duration occurs prior to the end of Cycle 24, maintenance activities will be performed to restore the CEA.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 302 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated February 6, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15041A068), as supplemented by letter dated February 24, 2015 (ADAMS Accession No. ML15055A385), Entergy Operations, Inc. (Entergy), the licensee for Arkansas Nuclear One, Unit 2 (ANO-2), requested to revise the ANO-2 Technical Specifications (TS). The proposed revisions would eliminate exercising Control Element Assembly (CEA) 18 for the remainder of operating Cycle 24, currently scheduled to end September 20, 2015. The proposed amendment would modify a Note to Surveillance Requirement (SR) 4.1.3.1.2, such that CEA 18 may be excluded from SR performance for the remainder of Cycle 24.

The licensee requested the amendment due to a degraded upper gripper coil (UGC) located on top of the reactor vessel head, which is the coil that normally holds the CEA in place. Should the UGC fail during CEA movement, the CEA would drop into the core, resulting in a reactivity transient and power reduction, and could result in a plant shutdown. The licensee requests to delay the CEA surveillance until it can be repaired during the upcoming refueling outage in the fall of 2015.

The supplemental letter dated February 24, 2015, provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 3, 2015 (80 FR 11475).

2.0 REGULATORY EVALUATION

2.1 System Description

The CEAs for ANO-2 are used for reactivity control. The CEAs are divided into nine groups, of which two are Shutdown Groups designated as Banks A and B. The CEA 18 is in Shutdown Bank B. The Shutdown Groups are the first withdrawn during startup and the last inserted

during a planned shutdown. During power operation, insertion of the Shutdown Groups is prohibited, except to complete performance of the SR 4.1.3.1.2 CEA exercise test. Since CEA 18 is within Shutdown Group B, this CEA must remain fully withdrawn at all times when the reactor is critical, except during performance of SR 4.1.3.1.2.

2.2 Applicable Regulatory Requirements

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in Title 10 of the *Code of Federal Regulations* (CFR) Section 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation, (3) SRs, (4) design features, and (5) administrative controls.

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

ANO-2 was licensed for construction prior to the issuance of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," and so the ANO-2 operating license was issued based on compliance with the draft General Design Criteria (GDC) (*Federal Register* notice dated July 11, 1967 (32 FR 10213)). However, Section 3.1.1 through 3.1.6 of the ANO-2 Safety Analysis Report (SAR) provide an analysis against those GDC contained in 10 CFR Part 50, Appendix A. The GDCs of Section 3.1.1 through 3.1.6 of the ANO-2 SAR applicable to this proposed change are:

- GDC 4, "Environmental and dynamic effects design bases," states, in part, that "[s]tructures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents...."

- GDC 10, "Reactor design," states that "[t]he reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

- GDC 23, "Protection system failure modes," states that "[t]he protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced."

- GDC 25, "Protection system requirements for reactivity control malfunctions," states that "[t]he protection system shall be designed to assure that specified

acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods."

- GDC 26, "Reactivity control system redundancy and capability," states that "[t]wo independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions."

- GDC 27, "Combined reactivity control systems capability," states that "[t]he reactivity systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained."

- GDC 28, "Reactivity limits," states that "[t]he reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in the reactor coolant temperature and pressure, and cold water addition."

- GDC 29, "Protection against anticipated operational occurrences," states that "[t]he protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences."

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

The SR 4.1.3.1.2 currently states the following:

> Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days. (Note 1)
>
> Note 1 – Movement of CEA #43 is not required for the remainder of cycle 15. If an outage of sufficient duration occurs prior to the end of Cycle 15, maintenance activities will be performed to restore the CEA.

The SR 4.1.3.1.2 would be revised to state the following:

> Each CEA not fully inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 92 days. (Note 1)
>
> Note 1 – Movement of CEA 18 is not required for the remainder of Cycle 24. If an outage of sufficient duration occurs prior to the end of Cycle 24, maintenance activities will be performed to restore the CEA.

3.2 Summary of Technical Information Provided by the Licensee

The licensee requested to revise the ANO-2 TS in order to eliminate exercising CEA 18 for the remainder of operating Cycle 24, currently scheduled to end September 20, 2015. The proposed amendment would modify a Note to SR 4.1.3.1.2 such that CEA 18 may be excluded from SR performance for the remainder of Cycle 24.

In its letter dated February 6, 2015, the licensee stated that during the October 2, 2014, quarterly exercise, in accordance with TS SR 4.1.3.1.2, the CEA 18 UGC current trace indicated more "noise" than expected. The current draw was measured to be higher than the normal current. CEA 18 was transferred to the lower gripper coil (LGC), thus de-energizing the UGC. The licensee stated that this change in current was considered an indication of UGC degradation. The subsequent CEA exercise performed in January 2015 indicated further degradation of the CEA 18 UGC with a measured current draw higher than the September exercise, even though the UGC had not been in service during the SR interval. Despite the progressive degradation, CEA 18 was exercised successfully. Following the testing, the licensee energized the LGC "holding" coil for CEA 18 instead of the UGC, which is degrading.

In its letter dated February 6, 2015, the licensee stated that the purpose of SR 4.1.3.1.2 is to verify that the CEAs are moveable and trippable (i.e., otherwise free from mechanical binding). The licensee evaluated the degrading condition to avoid a rod drop accident or plant transient. The licensee concluded that CEA 18 remains trippable and is expected to remain available to contribute to shutdown margin (SDM).

In its letter dated February 6, 2015, the licensee provided the following evaluation regarding SDM:

> TSs 3.1.1.1 (Modes 1 through 4) and 3.1.1.2 (Mode 5) specify that SDM shall be greater than or equal to that specified in the Core Operating Limits Report (COLR). The Cycle 24 COLR SDM operating limit is 5.0% Δk/k in Modes 1 through 5. Calculations were performed at various Effective Full Power Days (EFPDs) throughout the remainder of Cycle 24 operation, and Entergy has determined that a minimum SDM of 5.5142% Δk/k would exist following a reactor trip assuming both CEA 18 and the single CEA of highest reactivity worth fail to insert, well above the 5.0% Δk/k SDM requirement of the COLR. This value was calculated at End of Cycle conditions which were determined to bound operation through the remainder of Cycle 24.

3.3 NRC Staff Evaluation

In a request for additional information (RAI) dated February 23, 2015 (ADAMS Accession No. ML15055A091), the NRC staff requested that the licensee provide additional information regarding any other possible causes, other than UGC degradation, for the higher current draws encountered while performing SR 4.1.3.1.2 (RAI 1). The licensee stated, in its letter dated February 24, 2015, that the wide current trace for the UGC could be caused by the following: (1) degradation in the gripper coil; (2) degradation in the firing circuit; or (3) the power switch or coil driver card could also fire the circuit improperly, generating a higher current draw and improper traces. In its RAI response, the licensee also provided the NRC staff with other possible causes for difficulties encountered while performing SR 4.1.3.1.2 other than UGC degradation, and provided information regarding whether it appeared that these other causes were the most probable cause. Based on the information provided by the licensee, the NRC staff has determined the licensee has adequately identified the possible causes for the degradation observed while exercising CEA 18, and, therefore, RAI 1 is resolved.

In its RAI dated February 23, 2015, the NRC staff also requested that the licensee identify any degradation mechanisms which could prevent the CEA from dropping into the core when the reactor trip breakers are open (RAIs 2, 3, and 4). By letter dated February 24, 2015, the licensee stated the following:

> All postulated failure modes are associated with the coil and associated control circuitry. The reactor trip circuit breakers (RTCBs) open upon an automatic or manual reactor trip signal which removes power from both control and holding circuitry. All coils on each CEA de-energize, resulting in all CEAs inserting into the core.
>
> ...Any postulated failure mechanism that could prevent rod insertion (such as mechanical binding of the CEA itself) is not influenced or impacted by coil failure, or, control or holding circuitry failures. Therefore, there are no postulated failure mechanisms where the coil or associated circuitry could physically prevent rod insertion once the RTCBs have opened. In addition, the postulated failure

modes would not affect the RTCBs. Although not directly related to the identified condition of CEA 18, the system is designed in accordance with single failure criteria such that all circuitry will be de-energized even if one RTCB fails to open upon a reactor trip.

Also in its letter dated February 24, 2015, the licensee identified the degradation mechanisms that could prevent the CEA from dropping into the core when the reactor trip breakers are open. The identified degradation mechanisms still allow for the CEA to insert into the core in a loss of power scenario, failure of the RTCB to open, failure of a reactor trip signal, or any physical limitation. Based on the above, the NRC staff has determined the licensee has adequately identified the failure modes and established that the rod will insert into the core with reasonable certainty, and, therefore, RAIs 2, 3, and 4 are resolved.

Additionally, in its RAI dated February 23, 2015, the NRC staff requested that the licensee confirm that SDM calculations were performed using NRC-approved methodologies currently used by ANO-2 to generate the COLR (RAI 5). In its response dated February 24, 2015, the licensee stated that it used the methodology contain in Entergy Topical Report ENEAD-01-P-A, "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," approved by the NRC staff by letter dated September 29, 1995 (ADAMS Legacy Accession No. 9510040333). The calculations performed assured adequate existing SDM assuming both CEA 18 and the single highest reactivity worth CEA failed to insert. These models and methods utilized for the SDM calculation are also used to perform the TS surveillances. The NRC staff determined that the SDM calculation was performed using an NRC-approved method. Based on the above, the NRC staff has determined the licensee has adequately calculated the SDM available if CEA 18 fails to trip, and that the SDM available in that case would be greater than the SDM required by the COLR, and, therefore, RAI 5 is resolved.

The NRC staff concludes that the licensee identified the possible causes of the problems identified with CEA 18, identified the failure modes, established that CEA 18 remains trippable with reasonable certainty, and verified that there is adequate SDM (given the postulated event that CEA 18 and the single CEA of highest reactivity worth fail to insert). Additionally, the NRC staff concludes that the elimination of SR 4.1.3.1.2 for CEA 18 for the remainder of Cycle 24 will reduce the likelihood of a rod drop event and resulting plant transient until CEA 18 can be repaired. Based on the above, and since the licensee has provided information to assure that the requirements of 10 CFR 50.36(c)(3) and the ECCS performance criteria of 10 CFR 50.46 will continue to be met, the NRC staff concludes that the elimination of exercising CEA 18 for SR 4.1.3.1.2 for the remainder of Cycle 24 is acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

As mentioned above, the NRC has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* March 3, 2015 (80 FR 11475). At the time of issuance of this amendment, the public comment period has expired, however, the 60-day period to request a hearing does not expire until May 4, 2015. The NRC's regulations in 10 CFR 50.92 state that the NRC may make a final determination that a license amendment involves no

significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluation; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), an evaluation of the issue of no significant hazards consideration is presented below, as provided by the licensee in its license amendment request dated February 6, 2015:

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

   Response: No.

   One function of the CEAs is to provide a means of rapid negative reactivity addition into the core. This occurs upon receipt of a signal from the Reactor Protection System. This function will continue to be accomplished with the approval of the proposed change. Typically, once per 92 days each CEA is moved at least five inches to ensure the CEA is free to move. CEA 18 remains trippable (free to move) as illustrated by the last performance of SR 4.1.3.1.2 in January 2015. However, due to abnormally high coil voltage and current measured on the CEA 18 Upper Gripper Coil (UGC), future exercising of the CEA could result in the CEA inadvertently inserting into the core, if the UGC were to fail during the exercise test. The mis-operation of a CEA, which includes a CEA drop event, is an abnormal occurrence and has been previously evaluated as part of the ANO-2 accident analysis. Inadvertent CEA insertion will result in a reactivity transient and power reduction, and could lead to a reactor shutdown if the CEA is deemed to be unrecoverable. The proposed change would minimize the potential for inadvertent insertion of CEA 18 into the core by maintaining the CEA in place using the Lower Gripper Coil (LGC), which is operating normally. The proposed change will not affect the CEAs ability to insert fully into the core upon receipt of a reactor trip signal.

   No modifications are proposed to the Reactor Protection System or associated Control Element Drive Mechanism Control System logic with regard to the ability of CEA 18 to remain available for immediate insertion. The accident mitigation features of the plant are not affected by the proposed amendment. Because CEA 18 remains trippable, no additional reactivity considerations need to be taken into consideration. Nevertheless, Entergy has evaluated the reactivity consequences associated with failure of CEA 18 to insert upon a reactor trip in accordance with TS requirements for Shutdown Margin (SDM) and has determined that SDM requirements would be met should such an event occur at any time during the remainder of Cycle 24 operation.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   CEA 18 remains trippable. The proposed change will not introduce any new design changes or systems that can prevent the CEA from perform its specified safety function. As discussed previously, CEA mis-operation has been previously evaluated in the ANO-2 accident analysis. Furthermore, SDM has been shown to remain within limits should an event occur at any time during the remainder of operating Cycle 24 such that CEA 18 fails to insert into the core upon receipt of a reactor trip signal.

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

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

   Response: No.

   SR 4.1.3.1.2 is intended to verify CEAs are free to move (i.e., not mechanically bound). The physical and electrical design of the CEAs, and past operating experience, provides high confidence that CEAs remain trippable whether or not exercised during each SR interval. Eliminating further exercising of CEA 18 for the remainder of Cycle 24 operation does not directly relate to the potential for CEA binding to occur. No mechanical binding has been previously experienced at ANO-2. CEA 18 is contained within a Shutdown CEA Group and is not used for reactivity control during power maneuvers (the CEA must remain fully withdrawn at all times when the reactor is critical). In addition, Entergy has concluded that required SDM will be maintained should CEA 18 fail to insert following a reactor trip at any point during the remainder of Cycle 24 operation.

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

Based on the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 STATE CONSULTATION

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

6.0 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 made a final finding that the amendment involves no significant hazards consideration. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

Principal Contributor: M. Hardgrove, NRR/DSS/SRXB

Date: April 29, 2015

April 29, 2015

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISE TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENT TO ELIMINATE MOVEMENT OF CONTROL ELEMENT ASSEMBLY 18 FOR THE REMAINDER OF OPERATING CYCLE 24 (TAC NO. MF5698)

Dear Sir or Madam:

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

Sincerely,
/RA/

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 302 to NPF-6
2. Safety Evaluation

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ADAMS Accession No. ML15096A381 * per SE memo **via email

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NAME	LRegner**	AGeorge	JBurkhardt	RElliott (MChernoff for)	CJackson*
DATE	4/7/15	4/7/15	4/7/15	4/8/15	4/1/15
OFFICE	NRR/DE/EICB/BC	OGC – NLO	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM	
NAME	JThorp	DRoth	MMarkley	AGeorge	
DATE	4/10/15	4/20/15	4/29/15	4/29/15	

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