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

March 11, 2010

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: TECHNICAL SPECIFICATION CHANGE TO MODIFY THE MODE OF APPLICABILITY FOR EMERGENCY FEEDWATER ACTUATION SIGNAL (TAC NO. ME0779)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 289 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 2, 2009, as supplemented by letter dated June 24, 2009.

The amendment modifies TS 3.3.1.1, "Reactor Protective Instrumentation," and TS 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation," specifically, Table 3.3-1, Table 4.3-1, and Table 3.3-3, to adopt a mode of applicability for the Logarithmic Power Level - High, Pressurizer Pressure - Low, Steam Generator [SG] Pressure - Low, and the SG Differential Pressure and Level Low functions. These changes are consistent with NUREG-1432, Revision 3.0, "Standard Technical Specifications, Combustion Engineering Plants," dated June 2004.

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,

N. Kaly Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 289 to NPF-6
2. Safety Evaluation

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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. 289
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 March 2, 2009, as supplemented by letter dated June 24, 2009, 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) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 289, 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 within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
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: March 11, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 289

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 3-2	3/4 3-2
3/4 3-7	3/4 3-7
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15

(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. 289, 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

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1,2 3*,4*,5*	5 8
2. Linear Power Level – High	4	2	3	1,2	2,3
3. Logarithmic Power Level – High					
a. Startup	4	2(a)(d)	3	2,3*,4*,5*	2,3
b. Shutdown	4	0	2	3*,4*,5*	4
4. Pressurizer Pressure – High	4	2	3	1,2	2,3
5. Pressurizer Pressure – Low	4	2(b)	3	1,2	2,3
6. Containment Pressure – High	4	2	3	1,2	2,3
7. Steam Generator Pressure – Low	4/SG	2/SG	3/SG	1,2	2,3
8. Steam Generator Level – Low	4/SG	2/SG	3/SG	1,2	2,3
9. Local Power Density – High	4	2(c)(d)	3	1,2	2,3

TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U (1)	N.A.
2. Linear Power Level – High	S	D (2,4) M (3,4) Q (4)	TA (10)	1,2
3. Logarithmic Power Level – High	S	R (4)	TA (10) S/U (1)	1,2,3*,4*,5*
4. Pressurizer Pressure – High	S	R	TA (10)	1,2
5. Pressurizer Pressure – Low	S	R	TA (10)	1,2
6. Containment Pressure – High	S	R	TA (10)	1,2
7. Steam Generator Pressure – Low	S	R	TA (10)	1,2
8. Steam Generator Level – Low	S	R	TA (10)	1,2
9. Local Power Density – High	S	D (2,4) R (4,5)	TA (10) R (6)	1,2
10. DNBR – Low	S	S (7), D (2,4), R (4,5)	TA (10) R (6)	1,2
11. Reactor Protection System Logic	N.A.	N.A.	TA (10)	1,2,3*,4*,5*
12. Reactor Trip Breakers	N.A.	N.A.	M	1,2,3*,4*,5*
13. Core Protection Calculators	S	D (2,4) R (4,5)	TA (9,10) R (6)	1,2
14. CEA Calculators	S	R	TA (10) R (6)	1,2

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	1/Bus	2/Bus	1,2,3	9
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	1/Bus	1/Bus	1/Bus	1,2,3	9
8. EMERGENCY FEEDWATER (EFAS)					
a. Manual (Trip Switches)	2 sets of 2 per S/G	2 sets of 2 per S/G	2 sets of 2 per S/G	1,2,3	9
b. SG Level and Pressure (A/B) – Low and ΔP (A/B) – High	4/SG	2/SG	3/SG	1,2,3	10,11
c. SG Level (A/B) – Low and No S/G Pressure – Low Trip (A/B)	4/SG	2/SG	3/SG	1,2,3	10,11
d. ESFAS Logic					
1. Matrix Logic	6	1	3	1,2,3	12
2. Initiation Logic	4	2	4	1,2,3	9
e. Automatic Actuation Logic	2	1	2	1,2,3	13

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 be in at least HOT STANDBY within the next 6 hours and exit the MODE(s) of Applicability within the following 30 hours.

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)
4. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
5. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 11 – With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, operation in the applicable MODES may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
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 (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
3. Steam Generator 2 Pressure	Steam Generator 2 Pressure – Low Steam Generator 1 ΔP (EFAS 1) Steam Generator 2 ΔP (EFAS 2)
4. Steam Generator 1 Level	Steam Generator 1 Level – Low Steam Generator 1 ΔP (EFAS 1)
5. Steam Generator 2 Level	Steam Generator 2 Level – Low Steam Generator 2 ΔP (EFAS 2)

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.

Operation in the applicable MODES may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent operation in the applicable MODES may continue if one channel is restored to OPERABLE status and the provisions of ACTION 10 are satisfied.

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 be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

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 be in at least HOT STANDBY within the next 6 hours and exit the MODE(s) of Applicability within the following 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 289 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 application dated March 2, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090630163), as supplemented by letter dated June 24, 2009 (ADAMS Accession No. ML091800331), Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit No. 2 (ANO-2).

The proposed changes would revise:

1) TS 3.3.1.1, Reactor Protective Instrumentation, Table 3.3-1, to adopt a mode of applicability for the Logarithmic Power Level - High, Pressurizer Pressure - Low, Steam Generator [SG] Pressure - Low;

2) TS 3.3.1.1, Table 4.3-1, surveillance requirements modes for the Logarithmic Power Level - High, Pressurizer Pressure - Low, and Steam Generator Pressure - Low;

3) TS 3.3.2.1, Engineered Safety Feature Actuation System Instrumentation (ESFAS), Table 3.3-3, Functional Unit 8a (Emergency Feedwater, Manual (Trip Switches)) and Functional Unit 8b (SG level and Pressure (A/B) – Low and ΔP (A/B) – high, Action Statement 9 and Action Statement 10.

These changes are consistent with NUREG-1432, Volume 1, Revision 3, "Standard Technical Specifications, Combustion Engineering Plants" (ADAMS Accession No. ML041830597), dated June 2004. The NRC staff reviewed the proposed changes for compliance with 10 CFR 50.36 and agreement with the precedent as established in NUREG-1432. In general, licensees cannot justify TS changes solely on the basis of adopting the Standard Technical Specification (STS) model. Licensees may revise the TSs to adopt the improved STS format and content, provided that a plant specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or

(3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards

The supplemental letter dated June 24, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 2, 2009 (74 FR 26433).

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The TSs ensure the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The NRC regulatory requirements related to the content of the TSs are contained in Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) that requires that the TSs include items in the following categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TSs.

ANO-2 was originally designed to comply with the 70 "Proposed General Design Criteria for Nuclear Power Plant Construction Permits," published in July 1967. The ANO-2 Updated Final Safety Analysis Report, Sections 3.1.1 through 3.1.6 provide a comparison with the Atomic Energy Commission General Design Criteria (GDC) published as Appendix A to 10 CFR 50 in 1971. The current version of the updated Final Safety Analysis Report (FSAR), as required by 10 CFR 50.71, is Amendment 22.

In Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 13, 20, 21, 22, 23, and 24 describe requirements associated with automatic actuation systems for reactor protection. GDC 34 describes requirements associated with primary system heat removal methods that may include secondary heat removal capability.

> *Criterion 13--Instrumentation and control.* Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

> *Criterion 20--Protection system functions.* The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences

and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21--Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22--Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23--Protection system failure modes. The 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.

Criterion 24--Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 34--Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

3.0 TECHNICAL EVALUATION

3.1 Background

The safety-related emergency feedwater (EFW) system is designed to provide a means of supplying water to the intact SGs following a postulated main steam line rupture or loss of main feedwater (MFW) event to remove reactor decay heat and provide for cooldown of the reactor coolant system (RCS) to within the temperature and pressure at which the shutdown cooling (SDC) system can be placed in operation. The EFW system is designed so that the non-safety-related auxiliary feedwater (AFW) pump may be used to supply water to the SGs during non-emergency conditions to avoid challenging the safety-related EFW pumps.

The AFW pump (non-safety related) is capable of supplying sufficient water to the SGs for heat load of approximately 4 percent of full plant power at maximum SG pressure. During an emergency condition, the safety-related EFW pumps are designed to automatically supply water to the SGs upon the initiation of an emergency feedwater actuation signal (EFAS) or a diverse emergency feedwater actuation signal. In addition, in the unlikely failure of both safety-related EFW pumps during an emergency condition, the AFW pump can be manually actuated to supply water to the SGs.

The EFW system instrumentation and controls, (Reference: Section 10.4.9 of the UFSAR) are designed for operation during all phases of plant operation. The EFW system is designed to provide makeup to the SGs following a feedwater line break (FWLB) or main steam line break (MSLB) during Modes 1, 2, or 3. In Mode 4, the SDC system is placed in operation and used for decay heat removal purposes. EFW may be maintained available in the Mode 4 as a backup, but it is not required by plant design or by TSs. SG temperature and pressure is relatively low during Mode 4 operation. Since SDC is available, an FWLB will not prevent a means of continued primary system cooling in Mode 4, even with EFW unavailable. MSLB in Mode 4 would cause a cooldown of the primary system until the affected SG blows dry. At this point, SDC again will be the primary means to maintain post cooldown temperature of the primary system. Thus, ANO-2's plant design illustrates that EFW makeup is not required in Mode 4.

If SDC was lost and EFW is unavailable, feedwater may remain available via the AFW pump or the non-safety condensate pump contained within the MFW system. Although the AFW and MFW sources are lost on a loss-of-offsite power event, ANO has an alternate alternating current (AC) diesel generator that can be quickly started and connected to non-vital electrical buses to restore power to AFW or MFW components as needed. In addition, procedures are available that provide adequate guidance for restoring AFW/MFW capability following an MSLB, if needed. Thus, ANO-2's systems design would provide adequate makeup capability during a FWLB or MSLB.

3.2 Evaluation

3.2.1 TS Table 3.3-1 and Table 4.3-1

The Pressurizer Pressure - Low and Steam Generator Pressure - Low reactor trip functions mode of applicability in TS 3.3.1.1, Tables 3.3-1 and 4.3-1 (Functions 5 and 7) will be revised from "Modes 1, 2, 3*, 4*, and 5*" to "Modes 1, 2" ("*" indicates that the reactor Trip Circuit Breakers (TCBs) are closed and capable of Control Element Assembly (CEA) withdrawal).

In Table 3.3-1, the "and *" in Functional Unit 3a under Functional Unit 3, Logarithmic Power Level – High, is deleted. In addition, a "*" is added to the modes of applicability shown in the Applicable Modes column for Functional Unit 3.b.

The Logarithmic Power Level – High trip function mode of applicability in TS 3.3.1.1, Table 4.3-1 (Functions 3) will be revised from "Modes 1, 2, 3, 4, 5 and *" to "Modes 1, 2, 3*, 4*, 5*" (where "*" indicates that the TCBs are closed and capable of CEA withdrawal).

In Amendment No. 159, dated April 3, 1995 (ADAMS Accession No. ML021550345), the NRC staff approved the changes proposed by the licensee to revise the Reactor Protective System (RPS) and ESFAS instrumentation Limiting Condition for Operation (LCO) actions and to add a new administrative control requirement in the plant TSs. Based on the administrative and TS controls, including other trip functions that protect against inadvertent criticality from a subcritical condition that prevent an unintended approach to criticality in Modes 3, 4, and 5, with the "*" (relating to the TCB position) is no longer required.

In addition, the TSs require declaration of Mode 2 conditions prior to withdrawal of Regulator Group 3, which means that regulating CEA groups and Group P CEAs are fully inserted in Mode 3 or below. This ensures that a reduction in RCS temperature following an MSLB in Mode 3 cannot bring the reactor critical. Even if criticality were achieved, the Log Power Level High reactor trip, which occurs at 10^{-4} percent power, will ensure that safety limits are not exceeded. The production of a reactor trip signal, initiated by a Pressurizer Pressure - Low or Steam Generator Pressure - Low function, is not necessary since the reactor is already shut down in Modes 3, 4, and 5.

For the Logarithmic Power Level - High trip in Table 3.3-1, the "*" in Functional Unit 3a refers to conditions when the reactor TCBs are closed and the control rods are, therefore, capable of withdrawal. This is redundant to the "*" next to the modes of applicability shown in the Applicable Modes column. It is noted that Mode 2 does not contain a "*" because criticality of the reactor (in Mode 2) cannot be physically achieved with reactor trip circuit breakers open (power source to control rods); therefore, it is not necessary for the "*" to be tied to Mode 2 conditions.

In addition, a "*" is added to the modes of applicability shown in the Applicable Modes column for Functional Unit 3.b. When reactor TCBs are open (control rods inserted), boron dilution monitors are used to monitor changes in reactivity. In addition, significant shutdown margin is maintained during shutdown operations by the elevated boron concentration of the RCS up to the point an approach to criticality is to be initiated. Prior to initiating an approach to criticality, reactor TCBs are closed and control rod Shutdown Banks A and B are withdrawn as backup to the Boron Dilution Monitors. The proposed change to Functional Unit 3.b will ensure Log Power channels are operable for reactivity monitoring purposes under this condition.

In TS Table 4.3-1, the Logarithmic Power Level - High trip function is designed to prevent exceeding safety limits should an inadvertent criticality occur due to an uncontrolled CEA withdrawal event. This event cannot reasonably occur when TCBs are opened since the CEAs have no means of receiving electrical power to their drive systems in this condition. The mode of applicability for this function is changed from "Modes 3, 4, 5" to "Modes 3*, 4*, 5*" (* indicates TCBs are closed and the CEAs are capable of withdrawal). As in the trip functions above, this results in a change to Table 3.3-1 relating to operability and Table 4.3-1 relating to surveillance testing of these instruments.

Based on the above, the NRC staff concludes that the changes to Tables 3.3-1 and 4.3-1 are acceptable.

3.2.2 Table 3.3-3 Mode of Applicability

Currently, the TS mode of applicability associated with automatic actuation and operation of the EFW systems required operable EFW-related actuation channels in Modes 1, 2, 3, and 4. Since the EFW components that are expected to be actuated are not required to be operable in Mode 4, requiring operable EFW-related actuation channels in Mode 4 is found to be unnecessary. As stated in Section 3.2.1 of this safety evaluation, SG temperature and pressure are relatively low during Mode 4 operation and with the SDC available, an FWLB will not prevent a means of continued primary system cooling in Mode 4, even with the EFW unavailable. A Mode 4 MSLB would cause a cooldown of the primary system until the affected SG blows dry. At this point, SDC again will be the primary means to maintain post cooldown temperature of the primary system. This design illustrates that EFW makeup is not required in Mode 4

Based on the above, the NRC staff concludes that the changes to Functional Units 8a and 8b in Table 3.3-3 are acceptable.

3.2.3 Table 3.3-3 Action Statements 9, 10, 11, 12, and 13

The Action statements under Table 3.3-3 are revised in light of the proposed mode of applicability change described above. Currently, some Action notes associated with equipment required to be operable in Modes 1, 2, and 3 also required the operator to place the plant in Mode 5 if an inoperable component was not restored to an operable status within the allotted time period. In such cases, the Action note should only require exiting the mode of applicability, which is Mode 4. Those Action notes that are associated with components where some are required in Modes 1, 2, and 3, while others are required in Modes 1, 2, 3, and 4, were revised to simply state that to "exit the MODE(s) of Applicability" within the currently specified time frame. This is consistent with TS LCO 3.0.1 which states, in part:

> Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification.

The NRC staff concludes that this change is acceptable since it is in accordance with the plant design basis in Section 10.4.9 of the UFSAR, "Emergency Feedwater System," and matches the TS-required mode of applicability of the EFW system and the instrumentation actuation channels that automatically initiate and control the EFW system. Additionally, the proposed mode of applicability for the EFAS channels is also consistent with the STS.

3.3 Summary

The NRC staff has reviewed the proposed TS changes to TS 3.3.1.1, Reactor Protective Instrumentation, and TS 3.3.2.1, Engineered Safety Feature Actuation System. The EFW system will actuate automatically and support a plant cooldown to Mode 4, where the SDC system may be placed in service for decay heat removal purposes. The licensee's TS complies with the regulatory requirements of 10 CFR 50.36.

The proposed TS changes will reconcile a difference between the current ANO-2 TSs and the STS in relation to RPS or ESFAS functions.

In addition, the staff concludes that the licensee has provided adequate justification to support the requested changes and reasonable assurance that ANO-2 complies with the regulatory requirements and, therefore, meets 10 CFR 50.36. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

4.0 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.

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, and there has been no public comment on such finding published in the *Federal Register* on June 2, 2009 (74 FR 26433). 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) 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: K. Desai
N. Kalyanam

Date: March 11, 2010

March 11, 2010

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: TECHNICAL SPECIFICATION CHANGE TO MODIFY THE MODE OF APPLICABILITY FOR EMERGENCY FEEDWATER ACTUATION SIGNAL (TAC NO. ME0779)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 289 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 2, 2009, as supplemented by letter dated June 24, 2009.

The amendment modifies TS 3.3.1.1, "Reactor Protective Instrumentation," and TS 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation," specifically, Table 3.3-1, Table 4.3-1, and Table 3.3-3, to adopt a mode of applicability for the Logarithmic Power Level - High, Pressurizer Pressure - Low, Steam Generator [SG] Pressure - Low, and the SG Differential Pressure and Level Low functions. These changes are consistent with NUREG-1432, Revision 3.0, "Standard Technical Specifications, Combustion Engineering Plants," dated June 2004.

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/

N. Kaly Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 289 to NPF-6
2. Safety Evaluation

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