January 13, 2006

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

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 (ANO-1) - ISSUANCE OF AMENDMENT RE: REVISION OF THE ALLOWABLE VALUE FOR EMERGENCY FEEDWATER INITIATION AND CONTROL FUNCTION (TAC NO. MC9437)

Dear Mr. Forbes:

The Commission has issued the enclosed Amendment No. 227 to Renewed Facility Operating License No. DPR-51 for ANO-1. The amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated January 3, 2006, as supplemented by letters dated January 6 and 10, 2006.

This amendment revises TS 3.3.11, "Emergency Feedwater [EFW] Initiation and Control System Instrumentation." Entergy Operations, Inc. (Entergy) requests a revised TS allowable value of $9.34 inches and a limiting trip setpoint value of $10.42 inches in order to achieve and maintain 100 percent power. An actuation time delay of #10.4 seconds is also provided to minimize the possibility of inadvertant actuations during anticipated transients such as main feedwater transients, main turbine trips, etc. Operation at 100 percent power with the current allowable value involves an increased risk of spurious EFW initiation. Entergy requested this amendment as an emergency license amendment to allow for full, rated power operation (100 percent) power at ANO-1 while preventing unnecessary plant transients.

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

Sincerely,

/RA/

Drew Holland, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 227 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

January 13, 2006

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

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 (ANO-1) - ISSUANCE OF AMENDMENT RE: REVISION OF THE ALLOWABLE VALUE FOR EMERGENCY FEEDWATER INITIATION AND CONTROL FUNCTION (TAC NO. MC9437)

Dear Mr. Forbes:

The Commission has issued the enclosed Amendment No. 227 to Renewed Facility Operating License No. DPR-51 for ANO-1. The amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated January 3, 2006, as supplemented by letters dated January 6 and 10, 2006.

This amendment revises TS 3.3.11, "Emergency Feedwater [EFW] Initiation and Control System Instrumentation." Entergy Operations, Inc. (Entergy) requests a revised TS allowable value of $9.34 inches and a limiting trip setpoint value of $10.42 inches in order to achieve and maintain 100 percent power. An actuation time delay of #10.4 seconds is also provided to minimize the possibility of inadvertant actuations during anticipated transients such as main feedwater transients, main turbine trips, etc. Operation at 100 percent power with the current allowable value involves an increased risk of spurious EFW initiation. Entergy requested this amendment as an emergency license amendment to allow for full, rated power operation (100 percent) power at ANO-1 while preventing unnecessary plant transients.

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

Sincerely,
/RA/
Drew Holland, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 227 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION: RidsNrrDorlLplg (DTerao) RidsOgcRp
PUBLIC RidsNrrPMDHolland RidsAcrsAcnwMailCenter RidsNrrDorlDpr
LPLIV Reading YHHsii GHill(4) PRebstock
RWDeese (ANO-1 Sr. Res) JDixon (ANO-1 Res) ECrowe (ANO-1 Res)
RidsNrrDorl (CHaney/CHolden) RidsNrrPMJDonohew RidsNrrLADJohnson
RidsNrrDirsItsb (TBoyce) AHowe JNakoski
RidsRgn4MailCenter (DGraves)

Package No.: ML060130643 TS: ML060170321 NRR-100
ADAMS Accession No.: ML060130639 NRR-058

OFFICE	LPL4/PM	LPL4/LA	EICB/BC	SPWB/BC	OGC Nlo	LPL4/BC
NAME	DHolland	LFeizollahi	AHowe	JNakoski	MWoods	DTerao
DATE	1/13/06	1/13/06	1/13/06	1/13/06	1/13/06	1/13/06

OFFICIAL RECORD COPY

E:\Filenet\ML060130639.wpd

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 227
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated January 3, 2006, as supplemented by letters dated January 6 and 10, 2006, 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. DPR-15 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 227, are hereby incorporated in the license. EOI 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 7 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 13, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 227

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of 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.

Remove Page	Insert Page
3.3.11-2	3.3.11-2
3.3.11-3	3.3.11-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated January 3, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060050448), Entergy Operations, Inc. (Entergy/the licensee) requested an emergency Technical Specification (TS) amendment for (Arkansas Nuclear One, Unit 1 (ANO-1). By letters dated January 6 (ADAMS Accession No. ML060110060) and January 10, 2006 (ADAMS Accession No. ML060120152), Entergy supplemented its January 3, 2006, license amendment request with additional information requested by the Nuclear Regulatory Commission (NRC) staff.

The licensee indicates that the change is needed because phenomena not related to once-through steam generator (OTSG) level tend to reduce the measured value of the level during normal operation by an amount greater than anticipated. The request would lower the OTSG level - low allowable value of Limiting Condition for Operation (LCO) 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation," from $11.1 inches to $9.34 inches with the limiting trip setpoint $10.42 inches. In addition, an actuation time delay of #10.4 seconds is also proposed. The licensee indicated the lower allowable value is intended to alleviate its concern that operating at 100 percent power with the current allowable value involves an increased risk of spurious emergency feedwater (EFW) initiation, and the actuation time delay is intended to minimize the possibility of inadvertent actuations during anticipated transients such as main feedwater (MFW) transients and main turbine trip.

2.0 REGULATORY EVALUATION

In promulgating Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR), the NRC established its regulatory requirements related to the content of TSs. In doing so, the NRC emphasized those matters related to the prevention of accidents and mitigation of consequences of such accidents. As recorded in the Statements of Consideration accompanying the rule, "Technical Specifications for Facility Licenses; Safety Analysis Reports" (33 FR 18612, December 17, 1968), the NRC noted that licensees are expected to incorporate into their plant TSs those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity. Pursuant to 10 CFR 50.36, TSs are required to include items in five specific categories related to station operation. Specifically, those categories include: (1) safety limits, limiting safety system settings (LSSSs), and limiting control settings; (2) LCOs; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TSs.

Nuclear power plants are licensed to operate at a specified core thermal power. The instrument measurement uncertainty should be considered to avoid exceeding the power level assumed in the design-basis transient and accident analysis. The safety-related instrument trip setpoints are calculated to ensure that sufficient allowance exists between the trip setpoint and the safety limit to account for instrument uncertainties. The Commission's regulatory requirements related to this review can be found in 10 CFR 50.36(c)(1)(ii)(A) which requires that, where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before they exceed a safety limit. LSSSs are for automatic protective devices related to variables having significant safety functions. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary (RCPB), containment, and associated safety systems. Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," is used to evaluate the conformance of setpoints with 10 CFR 50.36.

3.0 TECHNICAL EVALUATION

3.1 Description of the Proposed Changes

In its application, the licensee proposed the following changes to TS 3.3.11, "EFIC System Instrumentation:"

1. Add two footnotes to Surveillance Requirements (SRs) 3.3.11.2 and 3.3.11.3 on performing the channel functional test and channel calibration, respectively.
2. Revise the allowable value for the EFW initiation-SG level - low function in Table 3.3.11-1, "Emergency Feedwater Initiation and Control System Instrumentation."
3. Add two footnotes (c) and (d) to Table 3.3.11-1 to the allowable value for the EFW initiation-SG level - low function.

The NRC staff's evaluation of these TS changes is given below:

3.2 Thermal Hydraulics Evaluation

The EFW system is used to protect the core against the consequences of overheating conditions upon a loss of MFW or a loss of the reactor coolant system (RCS) forced circulation. TS Table 3.3.11-1 specifies the EFIC system instrumentation for four plant conditions that trigger the EFW system initiation as: loss of MFW pumps, Steam Generator level - low, OTSG pressure - low, and reactor coolant pumps (RCP) status (loss of all four RCPs). In addition, an Emergency Safeguard Actuation System actuation or a Diverse Reactor Overpressure Protection System/Anticipated Transient Without Scram Mitigation System Actuation Circuitry actuation will also initiate the EFW system. The licensee proposed to reduce the OTSG level - low instrumentation allowable value from 11.1 inches to 9.34 inches, to introduce other setpoint-related constraints and include a limiting setpoint of 10.42 inches and an actuation time delay of #10.4 seconds, while the other actuation settings remain intact. Although many transients and accidents analyzed in the ANO-1 Safety Analysis Report (SAR), Chapter 14, such as loss of alternating current power, loss of reactor coolant flow, and steam line break, require the EFW system for overpressure mitigation, these events result in plant conditions other than low OTSG level that actuate the EFW. The loss of feedwater (LOFW) transient is the only event that results in the actuation of the EFW due to low OTSG level, and is, therefore, the only transient affected by a change to the OTSG level - low actuation setpoint.

Since the OTSG level - low actuation setpoint is credited in the mitigation of the LOFW event, the licensee performed an analysis of the LOFW event to demonstrate that acceptance criteria associated with the LOFW are met with the revised OTSG level - low setpoint value and time delay. This analysis is performed with the assumption that the EFW system will be actuated when the OTSG level falls below 6 inches above the lower tube sheet plus a time delay of 10.4 seconds. This safety analysis actuation setpoint of 6 inches corresponds to the reduced allowable value of 9.34 inches and limiting as-left setpoint value of 10.42 inches (the evaluation of the instrument settings is addressed later in this section).

The analysis of the LOFW event is performed with the RELAP5/MOD2-B&W [Babcock and Wilcox] code and the methodology described in the NRC-approved topical report BAW-10193P-A, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors," dated January 2000. The analysis follows the guidance outlined in Appendix A of BAW-10193P-A, including the use of the "large detail" RELAP5/MOD2 plant model, and conservative initial and boundary conditions for the heatup transients. The large detail model is used for those transients such as LOFW in which the performance of the OTSG and/or secondary plant dominates the transients. Tables 1 and 2 in Attachment 1 to Entergy's January 3, 2006, letter list the input assumptions of initial and boundary conditions. These initial and boundary conditions are consistent with the guidance of BAW-10193P-A regarding the incorporation of uncertainties associated with initial power, reactor coolant pressure, temperature and flow, pressurizer level, and OTSG mass inventory, and boundary conditions regarding reactor trip setpoints, core decay heat, safety valve setpoints, reactivity coefficients, and EFW actuation setpoint and time delay, and the EFW flow rate. The analysis assumes 102 percent of rated thermal power plus the addition of reactor coolant pump heat, lower SG mass inventory at 62 percent operating range, high pressurizer level, pressurizer safety valve lift setpoint of 2500 pounds per square inch-gage (psig) plus 3 percent tolerance, moderator temperature coefficient (MTC) of $^{+}1.3 \times 10^{-5}$ Δk/k/°F. A failure of the motor-driven EFW pump is assumed, and the remaining EFW flow is conservatively assumed to be 500 gpm, versus the design flow of 700 gpm, equally split between the two OTSGs. The time delay for the EFW delivery to the OTSG is assumed to be 80 seconds to account for the EFW actuation delay time of 10.4 seconds pump start and time to reach full speed conditions and valve opening. No loss of offsite power is assumed consistent with the ANO-1 licensing basis.

Figures 5 through 12 in Attachment 1 to Entergy's January 3, 2006 letter, depict the analysis results, and Table 3 summarizes the LOFW event sequence. With the loss of MFW flow that ramps down at 7 seconds, resulting in boil-off of OTSG inventory and heat-up and pressurization of the primary side, the reactor trips at about 17 seconds due to high RCS pressure. After the reactor trip, the reactor power is reduced to the decay heat and the RCP heat. The RCS pressure continues to increase, resulting in the safety valve lifting in about 20 seconds. The OTSG secondary inventory continues to boil off, and at 83 seconds, the

OTSG level reaches the EFW actuation setpoint of 6 inches. Assuming an 80 seconds delay time for the EFW to reach the OTSGs, EFW starts to deliver 250 gallons per minute (gpm) to each OTSG at 183 seconds. This flow is initially not sufficient to remove the primary side heat until approximately 10 minutes when the decay heat has decreased sufficiently. Subsequently, the EFW is sufficient to remove the heat and pressure begins to decrease.

For the transient event, the peak RCS pressure is calculated to be 2704 pounds per square inch-absolute (psia), below the 100 percent design pressure of 2500 psig (i.e., 2750 psig). The analysis did not explicitly calculate the departure from nucleate boiling ratio (DNBR) for the LOFW transient. This is because the core thermal conditions during the transient do not approach the limiting conditions for departure from nucleate boiling (DNB). Prior to the reactor trip, the RCS temperature and pressure increase. At the time of reactor trip, the core inlet temperature has increased by less than 3 $^{\circ}$F while the RCS pressure has increased more than 200 pounds per square inch (psi), and the core power increases slightly due to a slightly positive MTC. However, the RCPs continue to maintain forced circulation flow through the core and keep the core in subcooled condition. The DNBR would decrease from the initial value but would still be above the DNBR limit. After the reactor trip at 17 seconds, the reactor power drops very quickly resulting in an increase in the DNBR. Therefore, the specified acceptable fuel design limit is met and there would be no fuel failure. Therefore, the safety analysis EFW OTSG level - low actuation setpoint of 6 inch and 10.4 seconds actuation time delay are acceptable.

3.3 Initial Control Transient

Entergy was asked to confirm that the increased OTSG level control error upon initiation of EFW would not adversely affect the operation of the system. The increased initial control error would be due to the increased difference between the (reduced) initiation setpoint and the (unchanged) control setpoint. Specifically, the licensee was asked to show that the control overshoot magnitude and period, or the additional time required to approach the control setpoint, if the system is underdamped, does not compromise the conclusion that acceptable system behavior is maintained. The license replied that the OTSGs are expected to be empty by the time EFW arrives, and that all of the EFW introduced into them would also be evaporated until after the RCS pressure and temperature had already begun to decrease. Once the water level begins to recover and level control becomes important, the heat load from the reactor core will already have begun to diminish. Therefore, the greatest heat load to the OTSG will occur when the water level is zero. The OTSG does not depend upon a fixed EFW inventory to accomplish its safety function, and so fluctuations in water level due to level control considerations are not important to the successful operation of the OTSG under EFW conditions. The licensee also pointed out that under normal operation the EFIC level error signal is artificially biased, so that the integral function in the Proportional-Plus-Integral controllers are already saturated when a need for EFW occurs. As a result, the control response is already saturated, and the specific water level in existence at the time of control initiation has little influence over the initial control response. Therefore, the original concern expressed above is resolved.

3.4 Setpoint-Related TS Considerations

A. Total Loop Uncertainty

The Total Loop Uncertainty calculation was provided in the licensee's January 6, 2006, supplement. This calculation accounts for process-induced errors, inherent instrument errors, environmental effects, errors in test and measurement equipment, drift over the interval between setpoint measurements, and other effects. The calculation also derives the other quantities applied in the TS as discussed below.

B. Limiting Setpoint

The licensee indicates that the limiting setpoint (10.42 inches) is computed by adding the total loop uncertainty to the OTSG level assumed in the safety analysis for level-based initiation of EFW, with the addition of a small arbitrary safety margin.

C. As-Found Setpoint Evaluation

1) Deviation Limit

The proposed TSs require that the as-found setpoint be within the acceptable tolerance band about the previous as-left setting. The associated bases establish the tolerance band as ±1.08 inches. The 1.08 inch limit is derived in the uncertainty calculation as the combination of instrument reference accuracy, calibration tolerance (M&TE uncertainty), and limiting anticipated drift over the interval between tests.

2) Allowable Value

The licensee indicates that the allowable value is computed by means of ISA67.04, "Setpoints for Nuclear Safety-Related Instrumentation," Part 2, Method 3. The proposed TS require that the channel be declared inoperable if the as-found setpoint is nonconservative relative to the allowable value.

3) As-Left Requirement

The proposed TS requires that the setpoint be reset to a value not less conservative than the established limiting setpoint.

The licensee has proposed various footnotes to address the setpoint limits and the associated calculational methodology associated with SRs 3.3.11.2 and 3.3.11.3 and with the EFW initiation-SG level - low function in Table 3.3.11-1. The NRC staff concludes that the setpoint limits and the methodology which has been used by the licensee to determine those limits will provide reasonable assurance that the setpoint-related assumptions of the safety analysis will be met.

3.5 Conclusions

The NRC staff has reviewed Entergy's proposed TS changes concerning allowable values and limiting setpoints, and the notes proposed to be added to the TS relating to testing of instrumentation associated with the EFIC initiation function. The addition of the notes and other TS changes provides reasonable assurance that the plant will operate in accordance with the design and licensing basis, and also that the operability of the instrumentation is ensured. The NRC staff has evaluated the proposed TS values with respect to the thermal-hydraulic performance of the OTSGs, the RCS, and the reactor fuel for the LOFW event. The LOFW is the only licensing basis event potentially impacted by the proposed TS values. The evaluation by the NRC staff of RCS and OSTG performance using the new setpoint and time delay have been determined to be acceptable. In addition, core thermal limits are met. Therefore, the establishment of the new level setpoint limits and time delay has been evaluated and is acceptable to the NRC staff based on our review of the licensee's submittals and the addition of the TS notes.

Based on (1) the proposed EFW OTSG level - low actuation setpoint of 6 inches and 10.4 seconds actuation time delay is acceptable to assure no fuel failure (the proposed actuation setpoint limit of 10.42 inches with 10.4 seconds time delay provides reasonable assurance that the analytical limit of 6 inches will not be exceeded), (2) the original concern has been resolved, and (3) the setpoint methodology assures that the LSSS shall be met, the NRC staff concludes that the proposed TS changes meet 10 CFR 50.36. Based on this, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATEMENT OF EMERGENCY CIRCUMSTANCES

10 CFR 50.91(a)(5) states:

> Where the Commission finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. In such a situation, the Commission will not publish a notice of proposed determination on no significant hazards consideration, but will publish a notice of issuance under § 2.106 of this chapter, providing for opportunity for a hearing and for public comment after issuance. The Commission expects its licensees to apply for license amendments in timely fashion. It will decline to dispense with notice and comment on the determination of no significant hazards consideration if it determines that the licensee has abused the emergency provision by failing to make timely application for the amendment and thus itself creating the emergency. Whenever an emergency situation exists, a licensee requesting an amendment must explain why this emergency situation occurred and why it could not avoid this situation, and the Commission will assess the licensee's reasons for failing to file an application sufficiently in advance of that event.

The original OTSG orifice plates contained 3/4" gaps and were only adjustable between fully closed (0 Percent open) and 25 percent open. Operating at greater than 25 percent open resulted in excessive OTSG (EOTSG) instability. In an effort to provide more adjustability of the EOTSG orifice plates while maintaining flow stability, the design value of the radial gap between the plate outside diameter and shell inside diameter was reduced to 5/8". It was also desired to set the EOTSG orifice plates such that the operating range was similar to that of the original OTSG. The OTSG operating ranges (unfouled) are on the order of 55 percent to 60 percent. The minimum required operating range is 50 percent, and the maximum inventory from flooding of the aspirator ports is 80 percent. Based on reducing the potential of being less than 50 percent or greater than 80 percent, a target operating range of 65 percent was selected. The 65 percent target value also provides calculated stability ratios that are improved from those of the original OTSG.

The RELAP5 thermal hydraulic computer code was used to model both the original OTSGs and EOTSGs using plant startup and operating level data. Data taken at 98 percent power during startup after outage 1R19 agreed well with these predictions. However, RELAP5 predictions of the OTSG EFIC low level response are not as easily predictable. This is due to difficulties in computing localized void distributions in unheated (non-boiling) regions (i.e., EFIC instrument span) rather than large heated regions that occur over the startup and operating ranges.

As a result, EFIC low level predictions for the EOTSG were performed on a comparative basis. OTSG EFIC level data was adjusted by the difference in EOTSG and orifice plate resistance. This adjustment assumed design dimensions, for the orifice plates and shell dimensions, and also assumed that the OTSG orifice plate was fully closed. Based on a relatively small difference in orifice plate pressure losses, only about a 10-inch difference in level was predicted at 100 percent power versus the approximately 50 inches experienced. Therefore, the reduced operating margin for the EFIC low level trip was not able to be readily foreseen. Because the instrument inaccuracy was unexpected, the licensee could not have anticipated the need for a license amendment that would allow for a 30-day comment period.

Therefore, the licensee concluded that an emergency TS change is appropriate to avoid continued derate of ANO-1 and to allow for an expeditious resumption of power to 100 percent. Based on the above, Entergy believes the conditions for an emergency TS change are met. Additionally, the proposed amendment involves no significant hazards as the licensee provided.

On the basis of the above discussion, the NRC staff has determined that, in accordance with 10 CFR 50.91(a)(5), emergency circumstances exist and that the licensee used its best efforts to make a timely application and could not avoid the emergency situation.

5.0 <u>FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION</u>

The Commission's regulations in 10 CFR 50.92(c) state that the Commission 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 evaluated; or

(2) Create the possibility of a new or different kind of accident from any previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following analysis was provided by the licensee in its January 3, 2006, letter.

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

   Response: No.

   The allowable value for actuation of the EFIC system is not an accident initiator and, therefore, cannot increase the probability of an accident. The EFIC and associated Emergency Feedwater (EFW) systems are components credited to mitigate the consequences of an accident. However, the small reduction in the SG Level – Low allowable value in conjunction with the addition of an actuation time delay still affords ample volume in the SGs [steam generators] to remove decay heat in a timely manner from the Reactor Coolant System (RCS) following a design[-]basis accident described in the ANO-1 Safety Analysis Report (SAR). The revised allowable values for both the SG Level- Low and the delay time will continue to enable the EFW system to maintain plant parameters within SAR limits for the previously evaluated accidents. The analysis results do not impact the dose consequences of any accidents previously analyzed.

   Therefore, the proposed 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.

   The proposed change to the TSs does not require any physical alteration to the plant or alter plant design, other than the slight reduction in SG Level – Low allowable value and the addition of an actuation time delay, as associated with EFIC actuation. The proposed change does not present a significant adverse impact on the EFIC function or EFW systems as credited in any safety analyses for the prevention or mitigation of any accident. The proposed change is associated with mitigating systems and the change cannot, in itself, initiate an accident of any type.

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

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

   Response: No.

   The proposed change does not significantly impact a margin of safety for any accident previously evaluated. Based on the revised safety analysis, the proposed change in EFIC low level initiation and the addition of an EFW actuation delay time will still assure adequate margin for EFW actuation under a Loss of Feedwater event, but will minimize inadvertent EFW actuation due to other plant transients.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, has concluded that operation of the facility in accordance with the proposed amendment satisfies the three standards of 10 CFR 50.92(c). Therefore, the NRC staff determines that the proposed amendment involves no significant hazards consideration.

6.0 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.

7.0 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 no significant hazards finding with respect to this amendment.

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.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated, or (b) create the possibility of a new or different kind of accident from any previously evaluated, or (c) involve a significant reduction in a margin of safety and, therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Yi-Hsiung Hsii
P. Rebstock

Date: January 13, 2006

Arkansas Nuclear One

cc:

Senior Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1700 K Street, N.W.
Washington, DC 20006-3817

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205