

October 5, 1992

Docket No. 50-368

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Yelverton:

SUBJECT: ISSUANCE OF AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE
NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M84039)

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. NPF-6 for the Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 9, 1992, as supplemented by letter dated September 14, 1992.

The amendment revises TS Table 2.2-1 Reactor Protective Instrumentation Trip Setpoint Limits and TS Table 3.3-4 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Trip Values, to allow for the replacement of the narrow range containment building pressure transmitters.

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

Sincerely,

Original signed by:

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:

- 1. Amendment No. 137 to NPF-6
- 2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	NRC/Local PDR	PD4-1 Reading
S. Peterson (2)	M. Virgilio	J. Larkins
P. Noonan	ACRS(10)(MSP315)	OGC(MS15B18)
D. Hagan(MS3206)	G. Hill(4)	Wanda Jones(MS7103)
C. Grimes(MS11E22)	PD4-1 Plant File	OPA(MS2G5)
OC/LFMB(MS4503)	A. Beach, RIV	RYoung
TAlexion	S. Athavale	

OFC	LA:PD4-1	I:PD4-1	PM:PD4-1	OGC	D:PD4-1
NAME	PNoonan	RYoung	TAlexion	CBachmann	JLarkins
DATE	9/21/92	9/21/92	9/21/92	9/24/92	10/5/92

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Sincerely,

Original signed by:

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 5, 1992

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Vice President, Operations ANO
Energy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

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

Sincerely,

A handwritten signature in cursive script that reads "Thomas W. Alexion".

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 137 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 2

cc:

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Honorable Joe W. Phillips
County Judge of Pope County
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Ms. Greta Dicus, Director
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Little Rock, Arkansas 72205-3867



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

ENERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
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 July 9, 1992, as supplemented by letter dated September 14, 1992, 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.

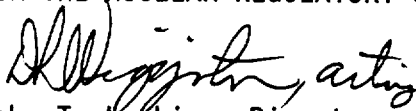
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

2. Technical Specifications

- The Technical Specifications contained in Appendix A, as revised through Amendment No. 137, are hereby incorporated in the 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.

FOR THE NUCLEAR REGULATORY COMMISSION



John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 5, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 137

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES

2-5
3/4 3-16
3/4 3-17

INSERT PAGES

2-5
3/4 3-16
3/4 3-17

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
b. Three Reactor Coolant Pumps Operating	*	*
c. Two Reactor Coolant Pumps Operating - Same Loop	*	*
d. Two Reactor Coolant Pumps Operating - Opposite Loops	*	*
3. Logarithmic Power Level - High (1)	≤ 0.75% of RATED THERMAL POWER	≤ 0.819% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1766 psia (2)	≥ 1712.757 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 729.613 psia (3)
8. Steam Generator Level - Low	≥ 23% (4)	≥ 22.111 (4)

* These values left blank pending NRC approval of safety analyses for operation with less than four reactor coolant pumps operating.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
9. Local Power Density - High	≤ 21.0 kw/ft (5)	≤ 21.0 kw/ft (5)
10 DNBR - Low	≥ 1.25 (5)	≥ 1.25 (5)
11. Steam Generator Level - High	$\leq 93.7\%$ (4)	$\leq 94.589\%$ (4)

TABLE NOTATION

- (1) Trip may be manually bypassed above $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\leq 10^{-4}$ of RATED THERMAL POWER.
- (2) Value may be decreased manually, to a minimum value of 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is ≥ 500 psia.
- (3) Value may be decreased manually during a planned reduction in steam generator pressure provided the margin between the steam generator pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.
- (4) % of the distance between steam generator upper and lower level instrument nozzles.
- (5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}\%$ of RATED THERMAL POWER.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure -Low	≥ 1766 psia (1)	≥ 1712.757 psia (1)
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 23.3 psia	≤ 23.490 psia
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed or placed in the tripped condition for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP VALUE</u>	<u>ALLOWABLE VALUES</u>
4. MAIN STREAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 751 psia (2)	≥ 729.613 psia (2)
5. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons)	Not Applicable	Not applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure - Low	≥ 1766 psia (1)	≥ 1712.757 psia (1)
6. RECIRCULATION (RAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	54,400 \pm 2,370 gallons (equivalent to 6.0 \pm 0.5% indicated level)	between 51,050 and 58,600 gallons (equivalent to between 5.111% and 6.889% indicated level)
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3120 volts (4)	3120 volts (4)
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	423 \pm 2.0 volts with an 8.0 \pm 0.5 second time delay	423 \pm 4.0 volts with an 8.0 \pm 0.8 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP VALUE</u>	<u>ALLOWABLE VALUES</u>
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level-Low	$\geq 23\%$ (3)	$\geq 22.111\%$ (3)
c. Steam Generator ΔP -High (SG-A > SG-B)	≤ 90 psi	≤ 99.344 psi
d. Steam Generator ΔP -High (SG-B > SG-A)	≤ 90 psi	≤ 99.344 psi
e. Steam Generator (A&B) Pressure - Low	≥ 751 psia (2)	≥ 729.613 psia (2)

-
- (1) Value may be decreased manually, to a minimum of > 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is ≥ 500 psia.
- (2) Value may be decreased manually during a planned reduction in steam generator pressure, provided the margin between the steam generator pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.
- (3) % of the distance between steam generator upper and lower level instrument nozzles.
- (4) Inverse time relay set value, not a trip value. The zero voltage trip will occur in 0.75 ± 0.075 seconds.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENERGY OPERATIONS, INC.,

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated July 9, 1992, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit No. 2 (ANO-2) Technical Specifications (TSs). The requested changes would revise TS Table 2.2-1 Reactor Protective Instrumentation Trip Setpoint Limits and TS Table 3.3-4 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Trip Values, to allow for the replacement of the existing narrow range containment building pressure transmitters (PTs) of the 0-70 psia range with new PTs of the 0-27 psia range. The September 14, 1992, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

2.0 BACKGROUND

Containment building pressure at the ANO-2 is monitored using four narrow range PTs which send signals to the Reactor Protection System (RPS) and ESFAS. Upon sensing a high pressure in the containment building, the PT loops generate the following trip and actuations:

- (1) High containment pressure trip of the RPS,
- (2) Actuation of the safety injection (SI) function of the ESFAS,
- (3) Actuation of the containment isolation function of the ESFAS,
- (4) Actuation of the containment cooling function of the ESFAS, and
- (5) Actuation of the containment spray function of the ESFAS.

The existing PTs must be replaced because of their approaching obsolescence, which leads to a lack of spare parts, and because they are approaching the end of their environmentally qualified life. The existing PTs are Rosemount 1153 series A, range code 6 type and have a calibrated range of 0-70 psia. The new PTs will be Rosemount 1153 series D, range code 5 type and will have a smaller calibrated range of 0-27 psia. The licensee wanted to procure new PTs in the

same calibrated span of 0-70 psia, but due to a large increase in calculated errors of the new PT loops, the licensee procured the PTs with a smaller calibrated span.

In the past, the licensee calculated setpoints for existing PT loops using the methodology accepted at the time. The past method of calculating setpoints did not account for all possible sources of errors and all possible environmental effects on the setpoint. The current methodology as prescribed by the Instrument Society of America (ISA) Standard 67.04, 1982 and endorsed by Regulatory Guide (RG) 1.105 Revision 2, "Instrument Setpoints for Safety-Related Systems," is comprehensive and addresses more possible factors of errors. Therefore, the present methodology results in a total loop error larger than the error resulting from the past methodology. The larger span also results in larger values of instrument errors, because the errors for most of the instruments are specified in proportion to the spans of these instruments.

The licensee noted that, if the current methodology is applied to new PTs having a span of 0-70 psia, which is also the span of the existing PTs, the sum of resulting uncertainties would be larger than the existing allowable margin between the nominal setpoint and the TS limit. Therefore, to reduce the total of the PT loop uncertainties, the licensee selected replacement PTs of the 0-27 psia span, which is smaller than the 0-70 psia span of the existing PTs. However, the sum of calculated loop errors even for the smaller span was slightly larger than the allowable margin between the nominal setpoint and TS limit of the process variable. Therefore, the licensee needed to reduce the TS setpoints for the above-listed RPS trip and ESFAS actuations, by the difference between the margin and the new error total, so that the analytical limit of the controlled variable could be maintained at its present value.

3.0 EVALUATION

The NRC staff reviewed the setpoint calculation for the setpoints for the above-listed RPS trip and ESFAS actuations. The staff reviewed Engineering Calculation 91-EQ-2002-02, Revision 0, "Loop Error, Setpoint, and Time Response Analysis for Narrow Range Containment Building Pressure ESFAS and RPS Trip Functions," provided by letter dated September 14, 1992. The licensee performed this calculation to determine the uncertainties, setpoints, allowable values, and response times of the narrow range pressure loops of containment building pressure at ANO-2. The licensee calculated the total of uncertainties in the loops for reference, abnormal, and accident conditions. The statistical method of the square root of the sum of squares (SSRS) was used to determine the sum of random errors in individual components, and in the complete loop. Non random errors were combined algebraically with the sum of random errors to establish total uncertainty of the PT loop. The licensee used a currently accepted methodology for this calculation. The staff found this practice acceptable.

The staff also reviewed the licensee's explanation (provided during telephone conversations and as documented in the September 14, 1992, letter) for the following conditions in Section 3.0, "Assumptions and Given Conditions" of Engineering Calculation 91-EQ-2002-02:

- (a) Accuracy of the measurement and test equipment (M&TE) used for calibration and testing devices of instrument loop was assumed to be twice as good as the accuracy of the device or the loop being tested.
- (b) Seismic and post seismic errors were not considered with any design basis events.
- (c) The drift value for the signal converter device was assumed to be equal to the instrument's reference accuracy without specifying any time interval.

For item (a) above the licensee explained that it has implemented plant procedures that provide guidance for test technicians to verify the accuracy requirements of M&TE before starting any calibration or test activity. The licensee stated that their technicians are adequately trained and plant procedures are implemented rigorously. The staff found this explanation acceptable. For item (b), the licensee has implemented comprehensive procedures at the plant to assess the effects of seismic activity immediately after it occurs. The licensee stated that the plant operators are trained for these procedures. The staff found this explanation acceptable. For item (c), the licensee informed the staff that it had previously not been able to obtain values of instrument drift from the instrument vendor. Therefore, the licensee evaluated historical data from the past calibrations of these instruments. Intervals of past calibrations were varied from 16 to 24 months. The licensee found that the worst case difference noted between the as-found and as-left setpoints was nearly equal to the reference accuracy of the instrument. Based on an evaluation of the historical calibration data, the licensee made an engineering judgement that assumed the drift value of the instrument would be equal to its reference accuracy. The staff believes that taking the difference between the as-found and the as-left settings equal to drift is reasonable and conservative, although reason for such difference may not be just the drift, but rather a combination of drift added with changes in instrument characteristics due to environmental effects during intervals between successive calibrations. The staff found the licensee's explanation acceptable.

4.0 SUMMARY

Based upon our review as presented above, the NRC staff finds the licensee's proposed TS amendment acceptable.

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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 34581). 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) 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: S. Athavale, SICB

Date: October 5, 1992