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

November 1, 1990

Docket No. 50-368

Mr. Neil S. Carns
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Carns:

SUBJECT: ISSUANCE OF AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. 74579)

The Commission has issued the enclosed Amendment No. 110 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 August 22, 1989, as supplemented July 5, 1990.

This amendment changes the allowable minimum setpoint range value on the Pressurizer Code Safety Valves as specified in TSs 3.4.2 and 3.4.3 from 1% to 3%. It would also change the minimum setpoint range value for the Main Steam Line Code Safety Valves from 1% to 3% as specified in TS Table 3.7-5. Under these TS revisions, if the setpoint for either type of safety valve was found outside a plus or minus 1% tolerance band, the setpoint would be adjusted to within plus or minus 1% of the lift setting specified in the TS. The effective date of this change is 30 days after issuance of this amendment to allow for distribution and procedure revisions by the licensee.

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.

DF01
1/1

DISTRIBUTION
Docket File
T. Quay
ARM/LFMB MNBB4503
C. Poslusny (2)
NRC PDR
L. Berry
OGC MS15B18
ACRS(10) P-315
Local PDR
PD4-1 Reading
PD4-1 Plant
R. Mullikin, RIV
J. Calvo MS11F22
G Hill(4) P1 37
D. Hagan MNBB 3206
M. Virgilio 13E4
GPA/PA 2G5
Wanda Jones MNBB 7103

Mr. Neil S. Carns
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 2

cc:

Mr. Donald C. Hintz
Executive Vice President
and Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, Mississippi 39286

Mr. Jerry Yelverton
Director Nuclear Operations
Arkansas Nuclear One
Route 3 Box 137G
Russellville, Arkansas 72801

Mr. Nicholas S. Reynolds
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
1700 Rockville Pike, Suite 525
Rockville, Maryland 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
1 Nuclear Plant Road
Russellville, Arkansas 72801

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Honorable Joe W. Phillips
County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

Ms. Greta Dicus, Director
Division of Environmental Health
Protection
Arkansas Department of Health
4815 West Markam Street
Little Rock, Arkansas 72201

Mr. Gerald Muench
Vice President Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, Mississippi 39286

Mr. Robert B. McGehee
Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, Mississippi 39205

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
Combustion Engineering, Inc.
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Admiral Kinnaird R. McKee, USN (Ret)
Post Office Box 41
Oxford, Maryland 21654

Mr. Tom Nickels
Arkansas Nuclear One
Route 3, Box 137G
Russellville, Arkansas 72801

Also enclosed with this amendment package is a change to the TS Basis 3/4.3.3.3 to correct a reference to Regulatory Guide 1.12, "Instrumentation for Earthquakes." This change is being made in response to your letter dated July 2, 1990. We reviewed the proposed change and found it to be acceptable. The revised TS page is enclosed.

Sincerely,

Chester Poslusny Jr.

Chester Poslusny, Jr., Project Manager
Project Directorate IV-1
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 110 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

Also enclosed with this amendment package is a change to the TS Basis 3/4.3.3.3 to correct a reference to Regulatory Guide 1.12, "Instrumentation for Earthquakes." This change is being made in response to your letter dated July 2, 1990. We reviewed the proposed change and found it to be acceptable. The revised TS page is enclosed.

Sincerely,

ORIGINAL SIGNED BY:

Chester Poslusny, Jr., Project Manager
Project Directorate IV-1
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 110 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	NRC PDR	Local PDR	PD4 Reading
M. Virgilio	ADR4	T. Quay	L. Berry
C. Poslusny (2)	OGC	D. Hagan	G. Hill (4)
Wanda Jones	PD4 Plant File	J. Calvo	ACRS (10)
GPA/PA	ARM/LFMB		

OFC	:PDIV-1/LA	:PDIV-1/PM	:EMEB	:SRXB	:OGC	:PDIV-1/D	:
NAME	:LBerry	:CPoslusny:	:TMarsh	:RJones	:E Holler	:TQuay	:
DATE	:10/3/90	:10/9/90	:10/18/90	:10/22/90	:10/26/90	:11/1/90	:

OFFICIAL RECORD COPY
Document Name: ANO2 AMEND 74579

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT No. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-6

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated August 22, 1989, and as supplemented on July 5, 1990, 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.110 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective 30 days after the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Acting Project Director
Project Directorate IV-1
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 1, 1990

ATTACHMENT TO LICENSE AMENDMENT NO.110

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	INSERT PAGES
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
3/4 7-4	3/4 7-4
B 3/4 3-2	B 3/4 3-2
B 3/4 7-1	B 3/4 7-1

REACTOR COOLANT SYSTEM

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA +1,-3%*.

APPLICABILITY: MODES 4 AND 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2. No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ±1% tolerance band, the setting shall be adjusted to within ±1% of the lift setting shown.

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting 2500 psia +1,-3%*.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

b. The provisions of Specification 3.0.4 may be suspended for one valve at a time for up to 18 hours for entry into and during operation in MODE 3 for the purpose of setting the pressurizer code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ±1% tolerance band, the setting shall be adjusted to within ±1% of the lift setting shown.

TABLE 3.7-5

STEAM LINE SAFETY VALVES

	VALVE NUMBER		LIFT SETTING (+1,-3%)*	ORIFICE SIZE
	Line No. 1	Line No. 2		
a.	2 PSV 1002	2 PSV 1052	1078 psig	TT (26.0 $in.^2$)
b.	2 PSV 1003	2 PSV 1053	1105 psig	TT (26.0 $in.^2$)
c.	2 PSV 1004	2 PSV 1054	1105 psig	TT (26.0 $in.^2$)
d.	2 PSV 1005	2 PSV 1055	1132 psig	TT (26.0 $in.^2$)
e.	2 PSV 1006	2 PSV 1056	1132 psig	TT (26.0 $in.^2$)

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ±1% tolerance band, the setting shall be adjusted to within ±1% of the lift setting shown.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Safety Guide 12, "Instrumentation for Earthquakes," March, 1971.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," Feburary 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY CYCLES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1000 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The "as-found" requirements are conservative with respect to Section XI of the ASME Boiler and Pressure Vessel Code, 1986 Edition, and Addenda through 1987. The total relieving capacity for all valves on all of the steam lines in 14,799,360 lbs/hr which is 118.7 percent of the total secondary steam flow of 12, 463,439 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (125)$$

For single loop operation (two reactor coolant pumps operating in the same loop)

$$SP = \frac{(X) - (Y)(U)}{X} \times (**)$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 110 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

INTRODUCTION

By letter dated August 22, 1989, as supplemented July 5, 1990, Arkansas Power and Light Company requested changes to the Technical Specifications (TSs) appended to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). The proposed amendment would change the allowable minimum setpoint value on the Pressurizer Code Safety Valves as listed in TSs 3.4.2 and 3.4.3, and would change the allowable minimum setpoint range value on the Main Steam Line Code Safety Valves as listed in TS Table 3.7-5. The change would specify an operability range for the safety valves from -3% to +1% of the required setpoint. (Previously it was ±1%.) If the "as found" setpoint exceeded the ±1% band, then adjustment of the setpoint to within the ±1% band would be required. A change would also be made to the corresponding Bases section.

In addition, the licensee, in a letter dated July 2, 1990, requested a change to the TS Bases Section 3/4.3.3.3 to correct a reference included therein.

EVALUATION

The Code of Federal Regulations, 10 CFR 50.55a, requires, in part, that certain safety valves be tested in accordance with the American Society of Mechanical Engineers (ASME) Section XI Code requirements.

The current Technical Specifications 3.4.2 and 3.4.3 require that the Pressurizer Code Safety Valves have a lift setpoint of 2500# plus or minus 1%. Technical Specification Table 3.7-5 requires that the main steam line safety valves have a lift setpoint as specified within a range of from plus to minus 1%. When a surveillance identifies drift in the "as found" setpoints outside the TS limits for any of these code safety valves, ASME Code Section XI requires resetting, repair, or replacement of the valve and further requires that additional valves in the system be tested. In addition, the licensee is required to submit a Licensee Event Report in accordance with 10 CFR 50.73 whenever the TS requirements have not been met.

The licensee, in its submittal, indicated that the specified tolerance for these valves is occasionally exceeded during surveillance testing, usually in the minus direction and results in the additional testing of other valves, beyond that normally required. This results in an impact on plant schedule and additional radiation exposure to plant staff performing the tests. The ASME has recognized the potential for code safety valves to experience setpoint drift; i.e., ASME/ANSI OM-1987 requires that a valve be repaired or replaced and the cause of failure be determined and corrected only if the valve exceeds its set pressure by 3% or greater. Thus, the staff feels that the licensee's proposed change in the specified operability range of from -3% to +1% of the specific setpoint is not inconsistent with available ASME Code guidance.

Further, in the licensee's design basis analyses, the pressurizer code safety valves and the main steam line safety valves both were assumed to open at a pressure 1% above the setpoint. Hence, if the valves lifted at a setpoint less than this analyzed value, then the resultant peak pressure would be bounded by the limiting case established by the plus 1% tolerance. The licensee also performed an analysis of the minus 3% tolerance and determined that there would be no impact on the system response results derived by the original analyses.

The licensee did determine, however, that for one postulated event, the steam generator tube rupture, with a loss of offsite power, the offsite dose could increase by as much as 10% above previous estimates, but the result would still remain within 10 CFR Part 100 limits and would be below the estimated offsite doses calculated for the bounding case of a Loss of Coolant Accident. The licensee indicated that the increased dose would be caused by the increased release of steam through the main steam line code safety valves associated with the affected steam generator prior to its isolation. Following the isolation, only steam from the intact steam generator would be released and this steam would have a lower assumed concentration of radioactive material.

The staff has reviewed the licensee's justification for the proposed TS revision as discussed above and has determined that this revision indicates a change in the definition of operability of the valves but not in the design basis requirement to have the setpoints maintained within plus or minus 1%. This TS revision of the pressurizer code safety valve and main steam line code safety setpoint tolerance, as requested, would have acceptably low safety significance and would not exceed the limits of any of the accident analyses. Therefore, this revision is considered acceptable.

The staff has also reviewed the licensee's proposed change to the ANO-2 TS Basis 3/4.3.3.3 to correct the reference for Regulatory Guide (RG) 1.12, "Instrumentation for Earthquakes." The current TS Basis lists the date of the RG as April 1974. The proposed modification changes the date to March 1971, which is the revision date of the RG cited in the ANO-2 Safety Analysis Report, and more significantly, the plant was designed in accordance with the 1971 revision of the RG. Therefore, the staff finds this change to TS Basis 3/4.3.3.3 to be acceptable.

ENVIRONMENTAL CONSIDERATION

The amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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 exposures. 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. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 1, 1990

Principal Contributor: Chester Poslusny, Jr.