October 3, 1996

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NO. 186 TO FACILITY OPERATING LICENSE NO. DPR-51 ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M95368)

Dear Mr. Yelverton:

The Commission has issued the enclosed Amendment No. 186 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 29, 1996.

The amendment relocates cycle specific operating parameters from the TSs to the Core Operating Limits Report per Generic Letter 88-16. The parameters being relocated by this amendment include the variable low reactor coolant system pressure trip and the variable low reactor coolant system pressure-temperature protective limits.

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
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 186 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

DFO1/1

DISTRIBUTION

Docket File	PUBLIC	PD4-1 r/f	OGC (15B18)
P. Tressler	T. Alexion (2)	G. Hill (2)	J. Dyer, RIV
J. Roe	L. Hurley, RIV	J. Kilcrease, RIV f/r	C. Grimes
E. Adensam (EGA1)	L. Kopp		

Document Name: AR195368.AMD

OFC	PM/PD4-1	LA/PD4-1	SRXB	OGC
NAME	TAlexion:sp	PTressler	LKopp	
DATE	9/23/96	9/23/96	9/23/96	10/2/96
COPY	YES/NO	YES/NO	YES/NO	YES/NO

OFFICIAL RECORD COPY

TWA signed on 10/3/96

9610080391 961003
PDR ADOCK 05000313
P PDR

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

October 3, 1996

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NO. 186 TO FACILITY OPERATING LICENSE NO. DPR-51 ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M95368)

Dear Mr. Yelverton:

The Commission has issued the enclosed Amendment No. 186 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 29, 1996.

The amendment relocates cycle specific operating parameters from the TSs to the Core Operating Limits Report per Generic Letter 88-16. The parameters being relocated by this amendment include the variable low reactor coolant system pressure trip and the variable low reactor coolant system pressure-temperature protective limits.

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,

Thomas W. Alexion

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

Docket No. 50-313

Enclosures: 1. Amendment No. 186 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-199

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
1400 L Street, N.W.
Washington, DC 20005-3502

Manager, Rockville Nuclear Licensing
Framatone Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

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

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
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 April 29, 1996, 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. DPR-51 is hereby amended to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. **186** , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas W. Alexion

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

Attachment: Changes to the Technical Specifications

Date of Issuance: **October 3, 1996**

ATTACHMENT TO LICENSE AMENDMENT NO. 186

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace 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.

REMOVE PAGES	INSERT PAGES
iv	iv
7	7
8	8
9	9
9a	-
9b	-
9c	-
13	13
14a	-
14b	-
15	15
142	142

LIST OF FIGURES

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F) for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.

The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.

The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

pumps(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Table 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure (COLR) trip setpoint shown in Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

To account for the calibration and instrumentation errors, the accident analysis used the protective limit specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Table 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620 F.

E. Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

Table 2.3-1

Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)	One Reactor Coolant Pump Operating in Each Loop[d] (Nominal Operating Power, 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig. min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

(a) Automatically set when other segments of the RPS (as specified) are bypassed.

(b) Reactor coolant system flow, %

(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

- 2.1 Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits
- 2.3.1 Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure
- 3.1.8.3 Minimum Shutdown Margin for Low Power Physics Testing
- 3.5.2.1 Allowable Shutdown Margin limit during Power Operation
- 3.5.2.2 Allowable Shutdown Margin limit during Power Operation with inoperable control rods
- 3.5.2.4 Quadrant power Tilt limit
- 3.5.2.5 Control Rod and APSR position limits
- 3.5.2.6 Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 186 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated April 29, 1996, Entergy Operations, Inc. requested changes to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to expand the current Core Operating Limits Report (COLR). Specifically, the variable low reactor coolant system (RCS) pressure-temperature protective limits and the variable low RCS pressure trip (VLPT) setpoint would be relocated to the COLR.

The staff's evaluation of the proposed changes follows.

2.0 EVALUATION

The licensee requested TS changes in accordance with 10 CFR 50.90. The specific changes are as follows:

1) Specification 2.1.3

Specification 2.1.3, including the associated Bases, will be revised to relocate Figures 2.1-1 and 2.1-3 and design nuclear power peaking factors from the TSs to the COLR. These figures will be combined in the COLR.

2) Specification 2.3.1

Table 2.3-1 will be revised to relocate the VLPT setpoint from the TSs to the COLR. The associated Bases Figure 2.3-1 will be relocated to the COLR. Items C (RCS Pressure) and D (Coolant Outlet Temperature) of the Bases will also be revised to reflect the relocation of the VLPT setpoint and Figure 2.3-1.

3) Specification 6.12.3.1

Specification 6.12.3.1 will be revised to add the variable low RCS pressure-temperature protective limits and the VLPT setpoint as being documented in the COLR.

Although there have only been a few previous revisions to the ANO-1 VLPT setpoint, it is anticipated that an increasing number of future changes will be made in order to accommodate advanced core designs. For example, the licensee anticipates that a more restrictive VLPT setpoint will be required for Cycle 14 in order to prevent the control rod insertion limits from severely impacting plant operation. In addition, a further revision may be required for Cycle 15 in order to credit the margin allowed by the use of a statistical core design methodology. Therefore, due to these expected frequent changes to the VLPT setpoint for future cycles of ANO-1, the staff considers the VLPT setpoint an appropriate cycle-specific COLR item.

The licensee has requested that the cycle-specific variable low RCS pressure-temperature protective limits and the VLPT setpoint, be relocated from the TSs to the COLR. The staff has reviewed the proposed change and has determined that these cycle-specific parameters may be modified by the licensee, without affecting nuclear safety, provided that such changes are determined using the NRC-approved methodologies specified in TS 6.12.3.2. The approved calculational basis for the variable low RCS pressure-temperature protective limits and the VLPT setpoint exists in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," which is referenced in TS 6.12.3.2. NRC approval and a license amendment would be required prior to using a methodology other than one approved and specified in TS 6.12.3.2. Because plant operation will continue to be limited in accordance with the values of the cycle-specific variable low RCS pressure-temperature protective limits and the VLPT setpoint, using NRC-approved methodologies, the staff finds the proposed changes acceptable and consistent with NRC guidance contained in Generic Letter 88-16 on modifying cycle-specific parameters.

The staff has concluded that the relocation of the variable low RCS pressure-temperature protective limits and the VLPT setpoint to the COLR is acceptable for the following reasons: (1) these parameters are cycle-specific and, therefore, meet the intent of Generic Letter 88-16, (2) reference to and the requirement for conformance to these limits remains in the TSs, assuring conformance with 10 CFR 50.36, and (3) plant operation continues to be limited in accordance with the values of these parameter limits that are established using NRC-approved methodologies specified in the TSs and will ensure that operation will be consistent with applicable limits of the safety analysis. The proposed TS changes appropriately reflect relocation of the specified parameters to the COLR and, therefore, are acceptable.

3.0 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.

4.0 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 (61 FR 28613). The amendment also changes reporting and recordkeeping requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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.

5.0 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: L. Kopp

Date: October 3, 1996