50-313

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

December 31, 1998

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NO. 194 TO FACILITY OPERATING LICENSE NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M95703)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 194 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 May 31, 1996.

The amendment revises the surveillance test interval for the reactor protection system control rod drive trip breakers, protective channel coincidence logic, and electronic trip relays from a monthly interval to a quarterly interval. Your application of May 31, 1996, requested a six month surveillance test interval and referenced Babcock and Wilcox Owners Group (B&WOG) Topical Report BAW-10167, Supplement 3, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," dated January 1995, for much of the supporting analyses. A B&WOG submittal dated November 5, 1997 (subsequent to your request to revise the TSs for ANO-1), amended the topical report to propose a three month interval instead of the six month surveillance test interval included in the original topical and your proposed TS change. In its safety evaluation dated January 7, 1998, the Nuclear Regulatory Commission (NRC) staff accepted the revised topical report, including the three month test interval, for use by the licensees participating in the B&WOG program.

Following discussions with your staff regarding the most efficient means to proceed with your request, the NRC is issuing this amendment approving the three month surveillance test interval, as supported by the accepted topical report, BAW-10167, Supplement 3. You may, if you choose, attempt to justify further extensions to the surveillance test intervals on a plant specific basis or as part of another B&WOG project. The NRC staff will treat a request to extend the surveillance test interval beyond the three month interval approved in this amendment as a new request and considers its activities pertaining to TAC Number M95703 to be completed.

DF01

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:
Nicholas D. Hilton, 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. 194 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

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Document Name: AR195703.AMD

OFC	PM/PD4-1	LA/PD4-1	HICB	OGC	PD/PD4-1
NAME	NHilton/vw	CHawes	JWermiel	S. Martz	JHannon
DATE	11/9/98	11/16/98	12/10/98	12/16/98	12/31/98
COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

OFFICIAL RECORD COPY

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,

Nicholas D. Hilton

Nicholas D. Hilton, 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. 194 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

cc:

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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. 194
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 May 31, 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. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 194 , 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

Nicholas D. Hilton, 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: December 31, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 194

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
68	68
69	69
72d	72d

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup.

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

Table 4.1-1
Instrument Surveillance Requirements

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic	NA	Q	NA	
2. Control Rod Drive Trip Breaker	NA	Q(1)	NA	(1) To include independent testing of the shunt and undervoltage trip attachments.
3. Power Range Amplifier	NA	NA	T/W(1)	(1) Heat balance calibration twice weekly under steady state operating conditions, daily under non-steady state operating conditions.
4. Power Range Channel	S M(1)	M	M(1)(2)	(1) Using core instrumentation. (2) Axial offset upper and lower chambers monthly and after each startup if not done previous week.
5. Intermediate Range Channel	S	P/M	NA	
6. Source Range Channel	S(1)	P	NA	(1) When in service.
7. Reactor Coolant Temperature Channel	S	M	R	
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	R	
11. Reactor Coolant Pressure Temperature Comparator	S	M	R	
12. Pump Flux Comparator	S	M	R	

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
61. Core-exit Thermocouples	M	NA	R	
62 Electronic (SCR) Trip Relays	NA	Q	NA	
63 RVLMS	M	NA	R	
64 HLLMS	M	NA	R	

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.194 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 May 31, 1996, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit No. 1 (ANO-1) Technical Specification (TS). The requested changes were to revise the surveillance test interval (STI) for the reactor protection system reactor trip breakers (RTBs), protective channel coincidence logic/reactor trip modules (RTMs), and electronic trip relays from a monthly interval to a six month interval. The application of May 31, 1996, referenced Babcock and Wilcox Owners Group (B&WOG) Topical Report BAW-10167, Supplement 3, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," dated January 1995, for much of the supporting analyses. A B&WOG submittal dated November 5, 1997 (subsequent to the licensee's request to revise the TSs for ANO-1), amended the topical report to propose a three-month interval instead of the six-month STI included in the original topical and the proposed TS change for ANO-1. In its safety evaluation dated January 7, 1998, the Nuclear Regulatory Commission (NRC) staff accepted the revised topical report, including the three-month test interval, for use by the licensees participating in the B&WOG program.

Following discussions with the licensee regarding the most efficient means to proceed with the TS amendment request for ANO-1, the NRC staff agreed to issue this amendment approving the three-month STI, as supported by the accepted topical report, BAW-10167, Supplement 3. Although the approved amendment differs (in terms of the length of the STI) from the licensee's application, the underlying information provided in the initial proposed no significant hazards consideration determination remains valid.

2.0 BACKGROUND

The rod drive control system (RDCS) at ANO-1 provides for withdrawal and insertion of control rod assemblies (CRAs) to control reactivity within the reactor core. The RDCS consists of three basic components: (1) control rod drive (CRD) motor power supplies, (2) system logic, and (3) trip breakers and the electronic trip relays (SCR). The RDCS trip breakers and electronic trip relays are provided to interrupt power to the control rod drive motors. When power is removed, the roller nuts (within the control rod drive mechanism) disengage from the leadscrew (attached to the CRAs) and a gravity free-fall of the CRAs occurs.

The proposed change to the STIs for the RTBs, RTMs, and electronic trip relays is based on Supplement 3 to B&WOG Topical Report BAW-10167. The topical report includes analyses to justify the proposed changes to the STI for the RTMs, RTBs, and electronic trip relays. In a safety evaluation dated January 7, 1998, the NRC staff accepted BAW-10167, Supplement 3 (revised to propose a three month test interval) for use by the licensees participating in the B&WOG program, including the licensee for ANO-1.

3.0 EVALUATION

The NRC staff reviewed topical report BAW-10167, Supplement 3, and found that, as revised, it was acceptable and that the STI for the reactor trip devices, consisting of RTBs, RTMs, and electronic trip relays, could be extended for those B&W plants that participated in the B&WOG program. The licensee participated in the program and therefore, the staff's generic findings regarding the extension, from a one month test interval to a three month interval, is applicable to the reactor trip devices at ANO-1.

The reliability models used in the analyses in BAW-10167, Supplement 3, were representative of the Oconee-type reactor trip system (RTS) design group that includes ANO-1. The unavailability of the RTS trip devices was modeled in the report using reliability block diagrams for both the current one-month STI and the originally proposed six-month STI. The proposed STI extension was analyzed for its potential effect on core melt frequency and RTS unavailability to demonstrate that the proposed STI change did not significantly increase plant risk when compared to the current TS requirements. The submittals pertaining to the topical report and the licensee's proposed TS change and the NRC safety evaluation for BAW-10167, Supplement 3, were prepared prior to the issuance of Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998. The overall methodology used by the staff and the staff's findings pertaining to the acceptability of the topical report are, however, generally consistent with the positions defined in the later regulatory guides for the employing risk-informed insights into the regulation of nuclear power plants.

The methodology and models presented in BAW-10167, Supplement 3, were the same as those in the NRC staff-approved Supplement 1 of BAW-10167 including time-dependent, common mode failure and uncertainty analyses. Emphasis was placed on the use of operating experience for the data source in the derivation of both random and common mode failure rates. The RTB portions of the reliability models included evaluation of failure mechanisms associated with cyclic stresses and time-in-service stresses. The RTB failure data reflected reliability improvements and reduction in the potential for common mode failures due to the licensee's implementation of the guidance provided in Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events."

The NRC staff reviewed the information provided in B&WOG submittals and found that the proposed change to the STI from one month to six months was acceptable in terms of its impact on RTS failure probability and overall plant risk. However, during the staff's review of BAW-10167, Supplement 3, the B&WOG revised the proposed extension from a six-month STI to a three-month STI due to the lack of operating history data for the extended test intervals. The B&WOG stated that the performance of the RTS trip devices would be monitored to ensure that degradation does not occur as a result of the STI extension. The staff notes that such performance monitoring can reasonably be expected as part of the requirements contained in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants." Given the requirements of 10 CFR 50.65, the staff does not consider it necessary to have additional requirements or regulatory commitments for monitoring or reporting the performance of the RTS trip devices.

In its submittal of May 31, 1996, the licensee included supporting information pertaining to the applicability of BAW-10167, Supplement 3 to ANO-1. The data from ANO-1 was included in the evaluations reported in the topical report and the staff finds that the site-specific experience for ANO-1 is consistent with the results reported in BAW-10167, Supplement 3. The licensee's submittal proposing TS revisions for ANO-1 was made prior to the revision of the topical report which reduced the proposed STI extension from six months to three months. Following discussions with the licensee regarding the most efficient means to proceed with the proposed TS changes, the NRC staff agreed to issue this amendment approving the three-month STI, as supported by the accepted topical report, BAW-10167, Supplement 3.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 44356). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Reckley

Date: December 31, 1998